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ENGINEERING SERVICES FOR THE NEXT GENERATION NUCLEAR PLANT (NGNP) WITH HYDROGEN PRODUCTION

NGNP Technology Development Road Mapping Report

Prepared by General Atomics
For the Battelle Energy Alliance, LLC

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EXECUTIVE SUMMARY

This report presents the work that the General Atomics (GA) NGNP team has performed on the HTGR Component Test Facility (CTF) initial conceptual design task (WBS element #CTF.000.ICD) under Subcontract 75309 with the Battelle Energy Alliance (BEA). Although an important objective of this task was to identify component testing that will require a test facility such as the CTF that is currently planned to be built at the INL to support the Next Generation Nuclear Plant (NGNP) Project, the primary effort was to systematically define the current technology readiness level (TRL) for the critical systems, structures, and components (SSCs) in GA's reference NGNP configuration[1] and to define the activities necessary to advance the TRLs to the level required for installation and operation of the SSCs in the NGNP. Consequently, the task is more appropriately referred to as the NGNP technology development road mapping task.

This report covers the entire scope of work performed by the GA team on the NGNP technology road mapping task. The scope included the following subtasks:

- Prepare a technology development road map (TDRM) and the supporting TRL rating sheets for each critical SSC
- Prepare a Test Plan for each critical SSC that identifies the activities necessary to advance the TRL from the initial (baseline) level to TRL 8 and that provides ROM cost estimates and a schedule for these activities
- Prepare an integrated technology development schedule that supports NGNP startup in 2021
- Perform a survey to assess the international gas-cooled reactor community's interest in the planned CTF at the INL and to obtain input with respect to the functional and operational requirements (F&ORs) for the CTF
- Review and comment on the preliminary CTF F&ORs prepared by INL and provide recommendations with respect to potential changes to the F&ORs
- Prepare a final report that includes the TDRMs and the supporting TRL ratings sheets, the integrated SSC test schedule, and comments and recommendations on the CTF F&ORs

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[1] The NGNP configuration, which served as the basis for this technology road mapping task, was the reference configuration as of June 2008 when the technology road mapping task began. GA's reference configuration has since changed as a result of the NGNP Project's decision to reduce the reactor outlet gas temperature objective for the NGNP from 950°C into the range of 750°C to 800°C. The NGNP Project has also decided to make co-generation of process steam and electricity the primary mission of the NGNP.

Because the NGNP design process is at a very early stage, adequate design details to precisely define design data needs (DDNS) and the testing required to satisfy the DDNs are not currently available. Consequently, the TDRMs, TRL rating sheets, and test plans reflect GA's engineering judgment at this time based on the results of the NGNP preconceptual and conceptual design studies performed by the GA NGNP team and the design data needs (DDNs) and engineering development plans developed for other GA MHR designs including the MHTGR, the NP-MHTGR, the GT-MHR, and the PC-MHR, none of which have the same reactor operating conditions as the reference NGNP configuration. Consequently, GA views the TDRMs, TRL rating sheets, and test plans as preliminary documents that will need to be continually updated as the design and technology development efforts progress. Further, it is assumed that DDNs specific to the NGNP design will be prepared during NGNP conceptual design and that the specific requirements for the tests needed to satisfy the DDNs will be defined in Test Specifications, which will also be prepared by GA during conceptual design. The details of the tests will be provided in test plans and test procedures to be prepared by the testing organizations.

The technology development road mapping effort was based on the NGNP configuration shown in Figure E-1, which General Atomics (GA) selected as its preferred configuration for the NGNP during the FY08-1 Conceptual Design Studies in early 2008. This plant configuration is consistent with the high-level requirements for the NGNP that existed at that time, and it was selected at the onset of the NGNP technology development road mapping task as the basis for the technology development road mapping effort.

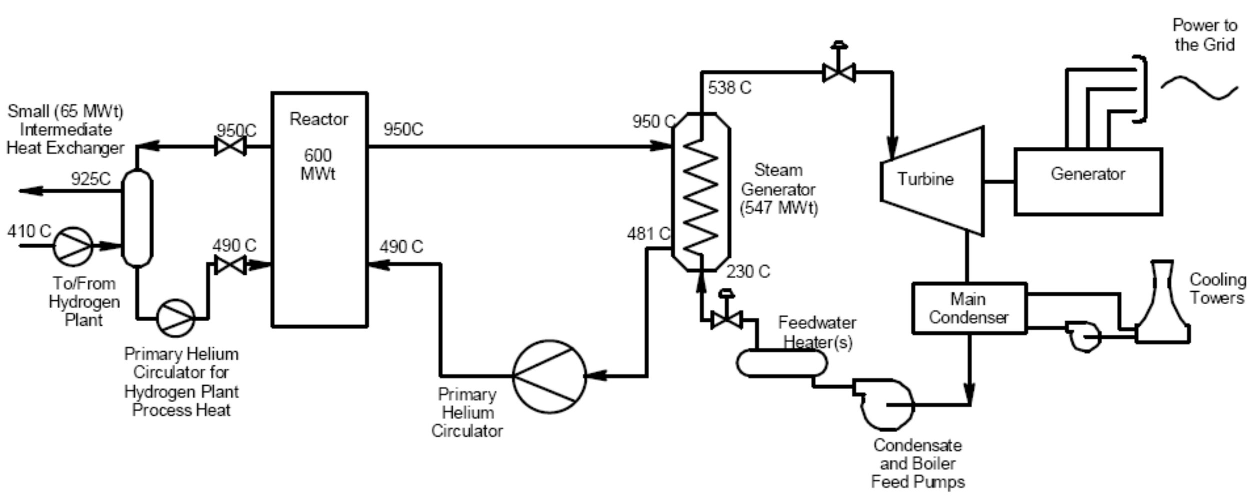

Figure E-1. NGNP Configuration for Technology Development Road mapping

The SSCs and the baseline technology readiness levels (TRLs) for the SSCs are based on the above NGNP configuration, and the TDRMs and Test Plans reflect this NGNP configuration and these assumptions.

For the purposes of the technology road mapping task, critical SSCs are defined as SSCs that are not commercially available or that do not have proven industry experience. Based primarily on the design data needs (DDNs) listed in Table 5 of the NGNP Technology Development Plan prepared by GA during the NGNP preconceptual design phase, GA identified the following critical SSCs to be considered in this study:

- Reactor control equipment
- Reactor internals (control rods)
- High temperature ducting (hot duct)
- Reactor core assembly
- Reactor graphite elements
- Reactor pressure vessel/reactor vessel cooling system
- Helium circulators (PHTS, SCS, SHTS)
- Intermediate heat exchanger (IHX)
- Shutdown cooling heat exchanger (SCHE)
- Reactor cavity cooling system (RCCS)
- Steam generator (SG)
- Turbomachinery (for direct combined-cycle PCS)
- High temperature valves
- S-I hydrogen production system
- Fuel handling and storage system
- Primary circuit and balance of plant instrumentation
- RPS, IPS, and PCDIS

GA's reference NGNP design as shown in Figure E-1 does not include turbomachinery; however, GA developed a TDRM and Test Plan for this PCS option because the GA team believes that a combined-cycle PCS (either direct or indirect) has the potential to improve the performance and economics of commercial gas-cooled reactor plants for electricity production and cogeneration.

Fuel, which is clearly a critical SSC for GA's NGNP design, was not addressed in this study. This is because the NGNP/AGR Fuel Development and Qualification Program already has a detailed technical program plan (that GA helped prepare as a participant in this Program) that defines the necessary technology development for fuel and fission products.

Table E-1 lists the initial (baseline) TRL rating that GA has assigned to each critical SSC.

Table E-1. Initial TRL Ratings for NGNP Critical SSC

SSC #	SSC	Initial TRL Rating
1	Reactor control equipment	4
2	Reactor internals	
	a. Control rods	2
	b. Control rod and RSM guide tubes	2
	c. Metallic core support structure (insulation)	3
	d. Upper core restraint	2
	e. Upper plenum shroud (thermal barrier)	2
3	Hot duct	2
4	Reactor core and graphite	
	a. Reactor core	5
	b. Graphite	6
5	Reactor pressure vessel/vessel cooling system	5
6	Helium circulator	6
7	Intermediate heat exchanger	2
8	Shutdown cooling system heat exchanger	4
9	Reactor cavity cooling system	4
10	Steam generator	
	750C gas inlet temperature	4
	950C gas inlet temperature	3
11	Turbomachinery (for combined cycle PCS)	4
12	High temperature isolation valves and pressure relief valves	3
13	S-I hydrogen production system	3
14	Fuel handling and storage system	4
15	Primary circuit and BOP protection inst.	3
16	RPS, IPS, PCDIS	4
N/A	Fuel	4

As noted above, fuel was not addressed in this study because the NGNP/AGR Fuel Development and Qualification Program already has a detailed technical program plan that defines the necessary technology development for fuel. However, it is GA's view that the current TRL for TRISO-coated UCO fuel is 4. This TRL rating is based on the excellent performance to date of experimental-scale fuel made at BWXT (UCO kernels) and ORNL (TRISO-coated particles and compacts) in irradiation test AGR-1, as indicated by the very-low fission-gas release from all six capsules in the test train. The AGR-1 test is scheduled to complete irradiation in the June – September 2009 time frame and post-irradiation examination (PIE) and safety-testing of the irradiated fuel will start shortly thereafter. A TRL rating of 5 will be achieved for the fuel when PIE results confirm satisfactory performance of the fuel during irradiation (i.e., with respect to retention of metallic fission products) and the results of safety-testing demonstrate acceptable fuel performance during simulated accident conditions (i.e., conduction cool down events).

It is important to note that a decision was made in October 2008 by the NGNP Project to reduce the nominal reactor outlet helium temperature for the NGNP from 950°C into the range of 750°C to 800°C with a corresponding reduction in the reactor inlet helium temperature. Because the current technology road mapping task covered by this report was started and largely completed while the reactor outlet helium temperature objective for NGNP was still 950°C, the technology road mapping effort continued to focus on defining the technology development activities required for a reactor operating at that temperature. However, the decision to reduce the reactor outlet helium temperature will have a significant impact on the technology development effort required to support the NGNP. Generally speaking, much of the technology development required for an NGNP operating with a reactor outlet helium temperature of 950°C will no longer be needed (for example, development and qualification of high-temperature metal alloys for the IHX and ceramic composites for several reactor internals components, design and verification of a reactor vessel cooling system, etc.). This reduction in required technology development could significantly impact the cost vs. benefit analysis for the planned NGNP Component Test Facility. Consequently, GA recommends that a follow-on study be performed in the near term to re-evaluate the TRLs and necessary technology maturation activities for an NGNP operating within the lower reactor outlet helium temperature range.

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ACRONYMS

ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing and Materials
BEA	Battelle Energy Alliance
CCD	Conduction Cool Down (event)
CTF	Component Test Facility
DDN	Design Data Need
DOE	U.S. Department of Energy
EHGA	Element Hoist and Grapple Assembly
F&ORs	Functional and Operational Requirements
FHEP	Fuel Handling Equipment Positioner
FHESS	Fuel Handling Equipment Support Structure
FHM	Fuel Handling Machine
FHSS	Fuel Handling and Storage System
FSIF	Fuel Sealing and Inspection Facility
FSV	Fort Saint Vrain
FTC	Fuel Transfer Cask
GA	General Atomics
GT-MHR	Gas Turbine Modular Helium Reactor
HTE	High Temperature Electrolysis
HTGR	High-Temperature, Gas-Cooled Reactor
HTTR	High Temperature Test Reactor
IHX	Intermediate Heat Exchanger
INL	Idaho National Laboratory
IFMU	In-core Flux Monitor
IPS	Investment Protection System
JAEA	Japan Atomic Energy Agency
KAERI	Korea Atomic Energy Research Institute
LWR	Light Water Reactor
NCA	Neutron Control Assembly
NGNP	Next Generation Nuclear Plant
MHR	Modular Helium Reactor
MHTGR	Modular HTGR
NP-MHTGR	New Production Modular HTGR
ORNL	Oak Ridge National Laboratory
PCDIS	Plant Control, Data, and Instrumentation System
PCHE	Printed Circuit Heat Exchanger
PC-MHR	Plutonium Consumption – Modular Helium Reactor

PCS	Power Conversion System
PHTS	Primary Heat Transport System
PIE	Post-Irradiation Examination
PRD	Power Range Detector
RCE	Reactor Control Equipment
RCCS	Reactor Cavity Cooling System
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSM	Reserve Shutdown Material
SCHE	Shutdown Cooling Heat Exchanger
SG	Steam Generator
SCS	Shutdown Cooling System
SHTS	Secondary Heat Transport System
SRD	Source Range Detector
S-I	Sulfur-Iodine
SSC	System, Structure, and Components
TDRM	Technology Development Road Map
TRL	Technology Readiness Level
UCO	Uranium Oxycarbide (fuel)
VCS	Vessel Cooling System

1 INTRODUCTION

1.1 Scope

This report presents the work that the General Atomics (GA) NGNP team has performed on the HTGR Component Test Facility (CTF) initial conceptual design task (WBS element #CTF.000.ICD) under Subcontract 75309 with the Battelle Energy Alliance (BEA). Although an important objective of this task was to identify component testing that will require a test facility such as the CTF that is currently planned to be built at the INL [INL 2008a] and [INL 2007] to support the Next Generation Nuclear Plant (NGNP) Project, the primary effort was to systematically define the current technology readiness level (TRL) for the critical systems, structures, and components (SSCs) in GA's reference NGNP configuration[2] and to define the activities necessary to advance the TRLs to the level required for installation and operation of the SSCs in the NGNP. Consequently, the task is more appropriately referred to as the NGNP technology development road mapping task.

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[2] As discussed in Section 1.2, the NGNP configuration, which served as the basis for this technology road mapping task, was the reference configuration as of June 2008 when the technology road mapping task began. GA's reference configuration has since changed as a result of the NGNP Project's decision to reduce the reactor outlet gas temperature objective for the NGNP from 950°C into the range of 750°C to 800°C. The NGNP Project has also decided to make co-generation of process steam and electricity the primary mission of the NGNP.

- Prepare a final report that includes the TDRMs and the supporting TRL ratings sheets, the integrated SSC test schedule, and comments and recommendations on the CTF F&ORs

Sections 1.2 and 1.3 present the reference NGNP configuration and the SSCs that were considered in this technology development road mapping task. Section 2 discusses the methodology used to develop the TDRMs. Section 3 provides the baseline TRL levels assigned to the SSC. Section 4 includes the TDRMs and supporting TRL rating sheets for each of the critical SSC. Section 4 also provides references to the test plans, which have been issued as separate documents and are not included herein (with the exception of two test plans prepared by GA team member URS – Washington division, which are included as appendices to this report). Section 5 provides both an integrated schedule for all of the technology maturation testing identified in the TRL rating sheets, TDRMs, and the test plans. Section 5 also includes a schedule just for the testing that can potentially be performed in the CTF. Section 6 presents the results of the survey that GA performed to assess the interest of the international gas-cooled reactor community in the planned CTF at the INL and to obtain input with respect to the functional and operational requirements (F&ORs) for the CTF. Section 7 provides comments and recommendations with respect to the CTF F&ORs defined in [INL 2008b].

Because the NGNP design process is at a very early stage, adequate design details to precisely define design data needs (DDNS) and the testing required to satisfy the DDNs are not currently available. Consequently, the TDRMs, TRL rating sheets, and test plans reflect GA's engineering judgment at this time based on the results of the NGNP preconceptual and conceptual design studies performed by the GA NGNP team and the design data needs (DDNs) and engineering development plans developed for other GA MHR designs including the MHTGR, the NP-MHTGR, the GT-MHR, and the PC-MHR, none of which have the same reactor operating conditions as the reference NGNP configuration. Consequently, GA views the TDRMs, TRL rating sheets, and test plans as preliminary documents that will need to be continually updated as the design and technology development efforts progress.

Further, it is assumed that DDNs specific to the NGNP design will be prepared during NGNP conceptual design and that the specific requirements for the tests needed to satisfy the DDNs will be defined in Test Specifications, which will also be prepared by GA during conceptual design. The details of the tests will be provided in test plans and test procedures to be prepared by the testing organizations. This approach is consistent with the approach shown in Figure 1-1, which GA has used historically to integrate design and technology development to maximize the benefit of the technology development programs in terms of supporting a plant design and minimizing the technical risk of the design. This model is based on successful Engineering Development and Demonstration (ED&D) programs conducted and managed by GA for DOE projects, including Accelerator Production of Tritium, the Salt Waste Processing Facility, the commercial GT-MHR, and the New Production Reactor.

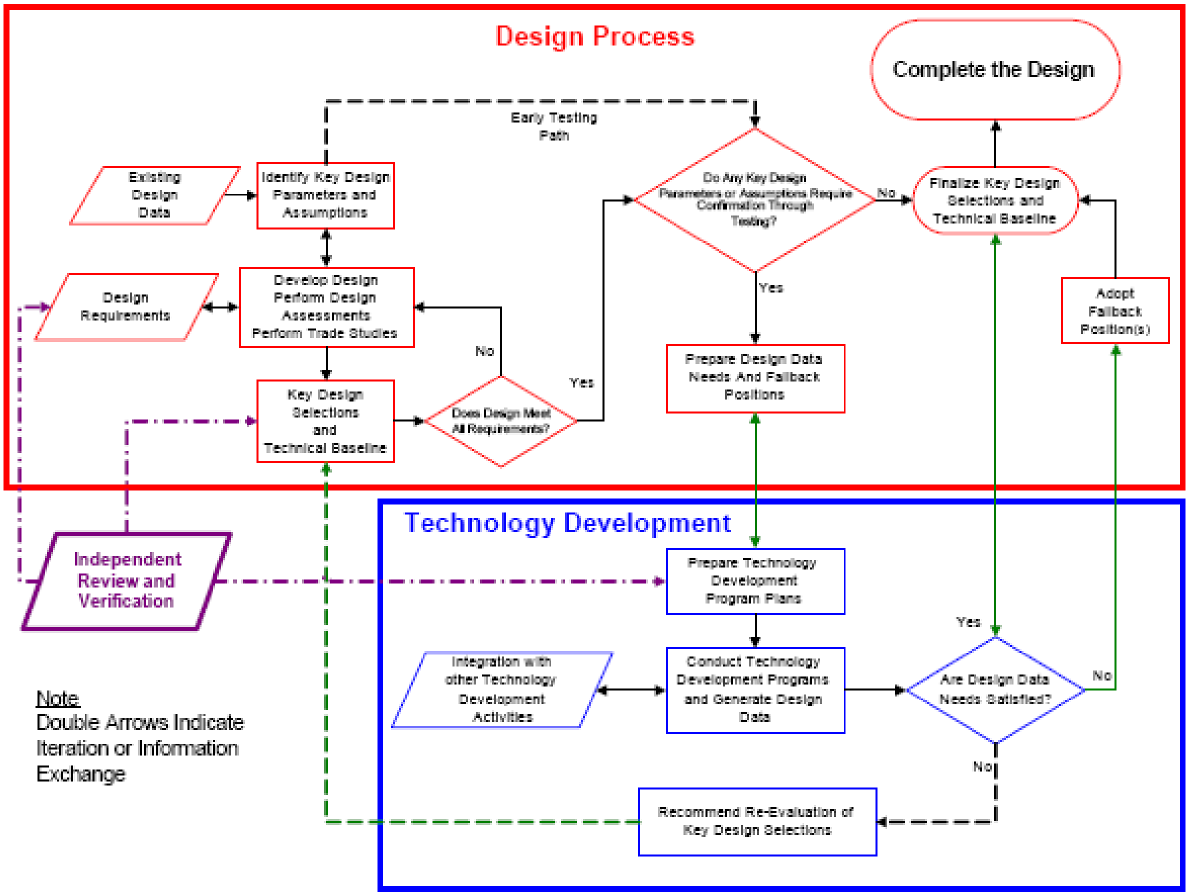


Figure 1-1. Approach for Integration of Design with Technology Development

The members of the GA NGNP team that participated in this task included GA, URS Washington Division (URS-WD) and Fuji Electric Systems. JAEA also participated as a subcontractor to Fuji. GA was responsible for all of the work scope with the exception of the TDRMs, TRL rating sheets, and Test Plans for two of the seventeen SSCs for which technology development road maps were prepared. JAEA reviewed and provided comments to GA on many of the TDRMs, TRL rating sheets, and Test Plans. Fuji and JAEA also provided recommendations with respect to testing in the CTF and the CTF F&ORs. Input concerning potential uses of the CTF was also provided by two other GA Team members, KAERI and Rolls-Royce.

1.2 Reference NGNP Configuration

This technology development road mapping effort is based on the NGNP configuration shown in Figure 1-2, which General Atomics (GA) selected as its preferred configuration for the NGNP during the FY08-1 Conceptual Design Studies in early 2008 [GA 2008a]. This plant configuration is consistent with the high-level requirements for the NGNP that existed at that time, and it was selected at the onset of the NGNP technology development road mapping task as the basis for the technology development road mapping effort.

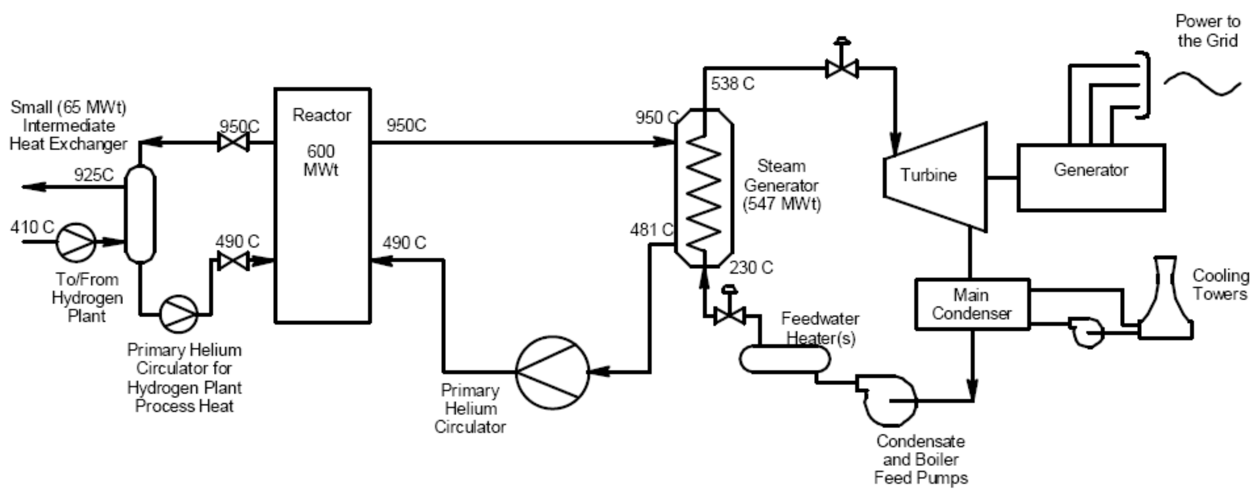


Figure 1-2. NGNP Configuration for Technology Development Road mapping

In the absence of a conceptual design, the following assumptions were made with respect to the NGNP design to provide a basis for this technology development road mapping effort. These assumptions are based on the various NGNP conceptual design studies that have been performed to date by the GA team.

- The working fluid for both the primary and secondary heat transport loops will be helium.
- All vessels will be made out of LWR steel (i.e., SA-508/533). A vessel cooling system will be used to keep reactor pressure vessel maximum temperatures below ASME code limits for SA-508/533
- The 65-MWt IHX will be a printed-circuit-type compact heat exchanger (PCHE); however, a helically-coiled tube-and-shell heat exchanger of similar design to the IHX used in the HTTR in Japan should be developed in parallel as a backup to the compact IHX

The SSCs and the current technology readiness levels (TRLs) for the SSCs are based on the above NGNP configuration and assumptions, and the TDRMs and Test Plans reflect this NGNP configuration and these assumptions. Although it is not needed for the NGNP configuration shown in Figure 1-2, a technology option that the GA NGNP team believes should be pursued because it has the potential to improve the performance and economics of commercial gas-cooled reactor plants for electricity production and cogeneration is a combined-cycle power conversion system (PCS) [GA 2008b]. Consequently, a TDRM and Test Plan was also prepared for the turbomachinery for a direct combined-cycle PCS.

It is important to note that a decision was made in October 2008 by the NGNP Project to reduce the nominal reactor outlet helium temperature for the NGNP from 950°C into the range of 750°C to 800°C with a corresponding reduction in the reactor inlet helium temperature. Because the current technology road mapping task covered by this report was started and largely completed while the reactor outlet helium temperature objective for NGNP was still 950°C, the technology road mapping effort continued to focus on defining the technology development activities required for a reactor operating at that temperature. However, the decision to reduce the reactor outlet helium temperature will have a significant impact on the technology development effort required to support the NGNP. Generally speaking, much of the technology development required for an NGNP operating with a reactor outlet helium temperature of 950°C will no longer be needed (for example, development and qualification of high-temperature metal alloys for the IHX and ceramic composites for several reactor internals components, design and verification of a reactor vessel cooling system, etc.). This reduction in required technology development could significantly impact the cost vs. benefit analysis for the planned NGNP Component Test Facility. Consequently, GA recommends that a follow-on study be performed in the near term to re-evaluate the TRLs and necessary technology maturation activities for an NGNP operating within the lower reactor outlet helium temperature range.

1.3 Critical SSCs

For the purposes of the technology road mapping task, critical SSCs have been defined by BEA as components that are not commercially available or that do not have proven industry experience. Based primarily on the design data needs (DDNs) listed in Table 5 of the NGNP Technology Development Plan prepared by GA during the NGNP preconceptual design phase [GA TDP 2007], GA identified the following critical SSCs to be considered in this study

- Reactor control equipment
- Reactor internals (control rods)
- High temperature ducting (hot duct)
- Reactor core assembly
- Reactor graphite elements
- Reactor pressure vessel/reactor vessel cooling system
- Helium circulators (PHTS, SCS, SHTS)
- Intermediate heat exchanger (IHX)
- Shutdown cooling heat exchanger (SCHE)
- Reactor cavity cooling system (RCCS)
- Steam generator (SG)
- Turbomachinery (for direct combined-cycle PCS)*
- High temperature valves
- S-I hydrogen production system
- Fuel handling and storage system
- Primary circuit and balance of plant instrumentation
- RPS, IPS, and PCDIS

*GA's reference NGNP design as shown in Figure 1-2 does not include turbomachinery; however, GA developed a TDRM and Test Plan for this PCS option because the GA team believes that a combined-cycle PCS (either direct or indirect) has the potential to improve the performance and economics of commercial gas-cooled reactor plants for electricity production and cogeneration [GA 2008b].

Fuel, which is clearly a critical SSC for GA's NGNP design, was not addressed in this study. This is because the NGNP/AGR Fuel Development and Qualification Program already has a detailed technical program plan (that GA helped prepare as a participant in this Program) that defines the necessary technology development for fuel and fission products.

2 METHODOLOGY

2.1 Establish Baseline TRLs

The TRLs are an input to inform NGNP project decision makers of the readiness of a particular technology or component. TRLs are associated with the entire NGNP or the applicable area, system, subsystem (structure), component, or technology (ASSCT). For TRLs 1 through 5, assessment typically occurs on a technology or component basis with a roll-up TRL for the areas, systems, and subsystems. TRLs 6 through 8 generally involve integrated subsystem or system testing, which allows TRL assessments directly against subsystems and systems.

Table 2-1 provides the TRL definitions that GA used as the basis for assigning a baseline TRL to each critical SSC. These TRL definitions are basically the same as in [INL 2008c], but GA has made some minor modifications for clarification purposes. These changes were reviewed and accepted by BEA.

As an aid to understanding the context under which TRLs are applied, Figure 2-1 depicts the interrelationship among the TRLs, their abbreviated definitions, and the increasing amount of integration as the TRL levels advance.

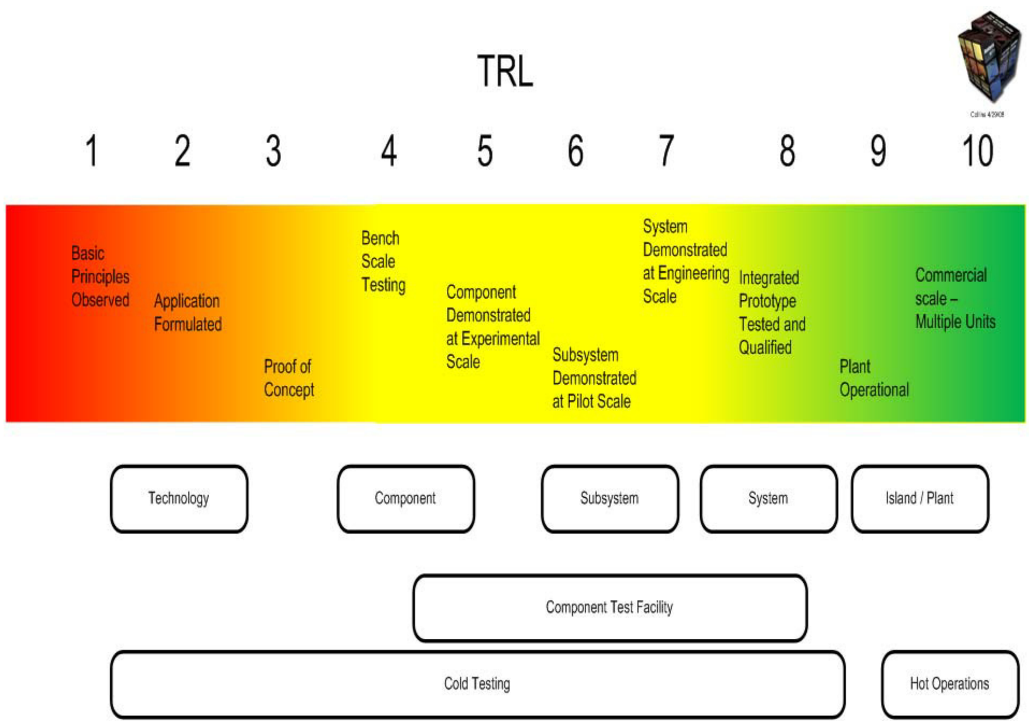


Figure 2-1. Comparison between TRL number, integration, and testing

Table 2-1. Technology Readiness Level (TRL) Definitions

TRL	Technology Readiness Level Definition	Abbreviated Definition
1	**Basic principles observed** and reported in white papers, industry literature, lab reports, etc. Scientific research without well-defined application.	Basic principles observed
2	**Technology concept and application formulated.** Issues related to performance identified. Issues related to technology concept have been identified. Issues related to materials of construction have been identified. Paper studies indicate potentially viable system operation	Application Formulated
3	**Proof-of concept:** Related industrial experience and/or technology, component, and/or material testing at laboratory scale provide proof of potential viability in anticipated service. Although analysis of performance of SSC gives favorable results, testing is required to provide the data needed to support design of key features. Materials property data may be incomplete, but sufficient traceable material properties data are available for material selection.	Proof of Concept
4	**Technology or Component bench-scale testing has been performed to** demonstrate technical feasibility and functionality. Alternately, equivalent relevant operating or test data from similar applications of the technology or component are available to demonstrate technical feasibility and functionality. For analytical modeling, use generally recognized benchmarked computational methods and traceable material properties.	Component Verified at Bench Scale
5	**Component demonstrated at less-than-full scale (experimental scale) in relevant environment**. Experimental scale testing provides the necessary design data or component demonstration, but the test article may not be a model of the final component design. Experimental-scale demonstration may also be satisfied by equivalent operating or test data from similar applications of the component. At this TRL, sufficient data is available to completely define the component and identify any technology issues that must be resolved before the component can be integrated into a system or subsystem for pilot scale testing. Demonstration methods include analyses, verification, tests, and inspection.	Component Verified at Experimental Scale
6	**Components have been integrated into a subsystem** and demonstrated at a pilot scale in a relevant environment. The test article used in pilot-scale testing will likely not be identical to the final version, but should be sufficiently representative to serve as a basis for performance demonstration. Pilot-scale demonstration may also be satisfied by equivalent operating or test data from similar applications, but a high degree of component/subsystem similarity is necessary to achieve this TRL based on such data.	Subsystem Verified at Pilot Scale
7	**Subsystem integrated into a system** for integrated engineering scale demonstration in a relevant environment.	System Verified at Engineering Scale
8	**Integrated prototype of the system is demonstrated** in its operational environment with the appropriate number and duration of tests and at the required levels of test rigor and quality assurance. Analyses, if used support extension of demonstration to all design conditions. Analysis methods verified and validated. Technology issues resolved pending qualification (for nuclear application, if required). Demonstrated readiness for hot startup.	System Tested and Qualified
9	**The project is in final configuration** tested and demonstrated in operational environment.	Plant Operational
10	**Commercial-scale demonstration** is achieved. Technological risks minimized by multiple units built and running through several years of service cycles – Multiple Units	Commercial Scale – Multiple Units

2.2 Preparation of TDRMs and TRL Rating Sheets

2.2.1 Technology Development Road Maps (TDRMs)

Based on the BEA SOW and discussions held with BEA, the information needed for each TDRM (much of which is to be provided in the TRL rating sheets) is as follows:

- Description of the SSC under consideration
- Current TRL for the SSC
- Identification of technology options, if any, for the SSC
- The decision discriminators to be used in technology down selection, if applicable
- The tasks (e.g., studies, tests, modeling, and analyses) required to obtain the discriminating information for technology down selection, if applicable
- The tasks necessary to achieve the next TRL level
- The tasks necessary to achieve all TRL levels up to TRL 8
- The validation requirements for each TRL level – parameters and, to the extent possible, acceptance values

Most technology option selections have already been made for the steam-cycle MHR and the GT-MHR based on past trade studies, but the higher reactor outlet helium temperature imposes more stringent demands on the NGNP that will require additional technology development, selection, and maturation. Key design and technology selection issues for the NGNP include, but are not limited to, those summarized in Table 2-2. In most cases, GA has already made a preliminary selection with respect to these issues based on the results of preconceptual and conceptual design studies for the NGNP and trade studies performed for previous MHR reactor designs. The TDRMs and test plans prepared under this NGNP technology road mapping task reflect these selections. These selections will need to be confirmed during NGNP conceptual design.

Table 2-2. Technology Options for NGNP

Critical System, Structure, or Component	Technology Options
Hydrogen production system	- S-I, HTE, or hybrid sulfur process
Intermediate heat exchanger (IHX)	- Heat exchanger type (tube & shell, PCHE, etc.) - Material of construction
Reactor pressure vessel	- VCS or no VCS - Material of construction
Helium circulators	- Bearing type - Impeller type - Motor type - Motor cavity seal type
High temperature valves	- Type of valve - Material of construction
Graphite	- Graphite grade(s) for fuel elements - Graphite grade(s) for reflector and core support elements
Reactor cavity cooling system	Air or water cooled system
Reactor internals	Material of construction (composites needed?)
Neutron control system	Material of construction (composites needed?)
High temperature ducting and insulation	- Material of construction for ducts - Type of insulation - Material of construction for cover plates
Steam generator	Materials of construction (if SG is to be located in primary loop and have a helium inlet temperature > 760°C)

2.2.2 TRL Rating Sheets

TRL rating sheets were developed for each TRL from the baseline TRL to TRL 8 for each critical SSC using the TRL rating sheet form provided by BEA (and slightly modified by GA). GA prepared TRL rating sheets for the yet-to-be-achieved TRLs as requested by BEA although it is clearly difficult to define a basis for the yet-to-be-achieved TRLs and the actions needed to reach the next level before reaching the previous rating level. The primary purpose of the TRL rating sheets for the higher-than-baseline TRL levels is therefore to provide an outline of the actions needed to advance to the next level. To reach a given TRL, all of the actions identified (to reach the next TRL level) in the TRL rating sheet for the previous TRL level must be successfully completed. Clearly, it will be necessary to update these TRL rating sheets as the technology development effort progresses and new information becomes available.

2.2.3 Test Plans

Test Plans were prepared for each of the SSCs identified in Section 1.3. These Test Plans are SSC specific and define and describe the activities required to advance the TRL from the baseline TRL to TRL 8. For the most part, the activities described are tests, but design and computer modeling activities are also identified and described in several of the Test Plans. As requested by BEA, the descriptions of the tests are generally organized under the following headings:

- Test objective
- Test description
- Test conditions
- Test configuration
- Required data
- Test location
- Data requirements
- Test evaluation criteria
- Test deliverables
- Cost, schedule, and risk

Also, as required by BEA, the Test Plans are organized by TRL level, with a section for each TRL step (i.e., 3 to 4, 4 to 5, etc.).

3 BASELINE TRL RATINGS

Table 3-1 lists the initial (baseline) TRL rating that GA has assigned to each critical SSC.

Table 3-1. Initial TRL Ratings for NGNP Critical SSC

SSC #	SSC	Initial TRL Rating
1	Reactor control equipment	4
2	Reactor internals	
	a. Control rods	2
	b. Control rod and RSM guide tubes	2
	c. Metallic core support structure (insulation)	3
	d. Upper core restraint	2
	e. Upper plenum shroud (thermal barrier)	2
3	Hot duct	2
4	Reactor core and graphite	
	a. Reactor core	5
	b. Graphite	6
5	Reactor pressure vessel/vessel cooling system	5
6	Helium circulator	6
7	Intermediate heat exchanger	2
8	Shutdown cooling system heat exchanger	4
9	Reactor cavity cooling system	4
10	Steam generator	
	750C gas inlet temperature	4
	950C gas inlet temperature	3
11	Turbomachinery (for combined cycle PCS)	4
12	High temperature isolation valves and pressure relief valves	3
13	S-I hydrogen production system	3
14	Fuel handling and storage system	4
15	Primary circuit and BOP protection inst.	3
16	RPS, IPS, PCDIS	4
N/A	Fuel	4

As noted in Section 1.3, fuel, which is clearly a critical SSC for GA's NGNP design, was not addressed in this study because the NGNP/AGR Fuel Development and Qualification Program already has a detailed technical program plan that defines the necessary technology development for fuel. However, it is GA's view that the current TRL for TRISO-coated UCO fuel is 4. This TRL rating is based on the excellent performance to date of experimental-scale fuel made at BWXT (UCO kernels) and ORNL (TRISO-coated particles and compacts) in irradiation test AGR-1, as indicated by the very-low fission-gas release from all six capsules in the test train. The AGR-1 test is scheduled to complete irradiation in the June – September 2009 time frame and post-irradiation examination (PIE) and safety-testing of the irradiated fuel will start shortly thereafter. A TRL rating of 5 will be achieved for the fuel when PIE results confirm satisfactory performance of the fuel during irradiation (i.e., with respect to retention of metallic fission products) and the results of safety-testing demonstrate acceptable fuel performance during simulated accident conditions (i.e., conduction cool down events).

4 TDRMS, TRL RATING SHEETS, AND TEST PLANS

Using the methodology discussed in Section 2, TDRMs and the supporting TRL rating sheets were developed for each critical SSC identified in Section 1.3. The TRL rating sheets and TDRM for each critical SSC are presented below. The TRL rating sheets provide a brief description of the SSC and the basis for the TRL rating assigned to the SSC. They also outline the actions required to advance the TRL to the next level. More detailed descriptions of the SSCs can be found in [PCDSR 2007] and in the Test Plans. The Test Plans for the critical SSCs have been issued as separate documents with the exception of the Test Plans for the hot duct (SSC-3) and the high-temperatures valves (SSC-12), which were prepared by URS – Washington Division and are included in this report as Appendix A and Appendix B. Table 4-1 identifies the document numbers for the Test Plans prepared by GA.

Table 4-1. Test Plans for NGNP Critical SSC

SSC #	SSC Description	Originating Org.	Report #
1	Reactor control equipment	GA	911133
2	Control rods	GA	911134
3	Hot duct	URS-WD	Appendix A
4a	Reactor core assembly	GA	911135
4b	Graphite elements	GA	911136
5	Reactor pressure vessel	GA	911137
6	Helium circulator	GA	911138
7	Intermediate heat exchanger	GA	911139
8	Shutdown cooling heat exchanger	GA	911140
9	Reactor cavity cooling system	GA	911141
10	Steam generator	GA	911142
11	PCS turbomachinery	GA	911143
12	High-temperature valves	URS-WD	Appendix B
13	S-I hydrogen production system	GA	911144
14	Fuel handling and storage system	GA	911145
15	Primary circuit and BOP instrumentation	GA	911146
16	RPS, IPS, and PCDIS	GA	911147

4.1 SSC-1 Reactor Control Equipment

TRL Rating Sheets, TRL 4 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-1.1	**Revision:** 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Reactor Control and Protection, Reactor Control Equipment (SSC-1)

Description:
SSC-1 contains equipment associated with control and measurement of reactor processes. This includes the Neutron Control Assembly (NCA), which contains control rod drive equipment and instrumentation. Reserve Shutdown Control Equipment (RSCE) is contained in the NCAs that operate the outer control rods. SSC-1 also includes other nuclear instrumentation – the in-core Flux Mapping Units (IFMU), the Source Range Neutron Detectors (SRD), and the Power Range neutron detectors. The Power Range neutron detectors are located in six wells, equally spaced around the Reactor Vessel, in the Reactor Building concrete structure behind the RCCS. (Cont.)

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Proof of concept	Components verified at bench scale	Components verified at experimental scale
TRL	**3**	**4**	**5**

Basis for Rating (Check box if continued on additional sheets) ☒

The initial level 4 technical rating for SSC-1 relies on experience gained at Fort St. Vrain and considerable conceptual design effort in both the commercial MHR program and the NPR program. Later, the GT-MHR program at General Atomics continued this work, all of which is applicable to the NGNP design and justifies a level 4 rating and continuation of the NGNP Conceptual Design (CD) at this technical rating. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Complete preliminary NHSS conceptual design of SRD, Power Range ex-vessel neutron detector, IFMU, NCA, and NCA with Reserve Shutdown equipment. Provide assembly views of each system. Document design issues. (Cont.)	GA	CD 0-36mo	2,000

DDN(s) Supported: C.11.10.01, C.11.01.03, C.11.01.04, C.11.01.05, C.11.01.06, C.11.02.01	**Technology Case File:**

Subject Matter Expert Making Determination: Dale Pfremmer

Date: 10/24/08	**Originating Organization:** General Atomics

Additional Description Sheet(s)

Description:

Component level development testing parallels Reactor Internals testing to develop control rod materials, guide tubes, and other Reactor Internals components. Likewise, the nuclear instrumentation design requires considerable interaction with Reactor System, Reactor Internals, and Vessel System development efforts. SSC-1 testing at the component level requires interaction with all these development efforts. SSC-1 testing of the installed NCA, SRD, and IFMU systems will be needed to confirm hot startup readiness. These tests involve operation of the equipment from the control room, utilizing the Plant Control, Data and Instrumentation System (PCDIS). Reactor Control and Protection development testing therefore includes testing of NCA, SRD, and IFMU operational functions for hot startup readiness, as well as testing at the component design level.

Additional Basis Sheet(s)

Basis:

Design issues remain from these efforts, but these require conceptual design analysis in order to make equipment selections and proceed with the technical development. For instance, the location of the SRDs in the lower reactor requires conduction cooldown temperature analysis, as well as calculation of the neutron flux levels at the SRDs following a reactor trip. These in turn are affected by vessel design considerations. In earlier designs, it was concluded that reflector material temperatures adjacent to the SRDs were too high for fission chamber detectors available at the time. There were solutions, such as the use of pyrolytic liners to protect the SRDs. This might require bench scale material testing if the design issue remains in the NGNP design. However, both the NGNP conceptual design temperature analysis and available fission chamber detector design improvements must be considered first. A level 5 TRL rating requires completion of conceptual design calculations, completion of component selections and mechanism designs, and review of the effects of all bench scale component data (obtained from manufacturers) on critical design issues. Inability to operate the SRDs at the required temperatures would be one of these issues.

Component data for the SRDs includes; neutron detection range, maximum operating temperature and pressure, duty-cycle and lifetime, etc. SRD neutron detectors are withdrawn through the lower vessel to prevent premature burnup, and remain withdrawn during all but startup, normal shutdown or reactor trip, and refueling operations. They must also remain inserted and operating in the event of a conduction cooldown event. They require drive mechanisms and controls, as well as supporting structures, pressure seals, insulation, etc. Range, maximum operating temperature and pressure, duty-cycle and lifetime data is also required for the Power Range neutron detectors. However, the Power Range detectors have different temperature requirements and remain in place for all operations except maintenance. The Reactor Building design must assure that the neutron detectors and instrumentation cabling are not exposed to undesirable temperature transients during a conduction cooldown event. The IFMUs are also movable neutron detection devices. They include neutron detector assemblies, drive mechanisms and controls, thermocouples, cabling, etc. They are lowered into the reactor by a weight and retracted by the drive mechanism, and include a support structure for the movable detector and guide tube equipment, gamma shielding to protect personnel during retrieval of the IFMU, pressure seals, insulation, and flow restrictions to suppress flow of hot core inlet gas into the vessel penetration and to minimize air in-leakage during handling operations. The Neutron Control Assembly (NCA) contains and operates the control rods. The NCAs also have drive mechanisms and controls, thermocouples, cabling, etc., but in addition will be instrumented to obtain the control rod position, rod full-in or full-out indication, control rod motor start stop indication, and control rod support cable tension for each control rod. The NCAs will also be instrumented to obtain temperatures in the lower portion of the NCA control rod drive mechanism area. These temperatures will be processed by

the PCDIS to provide excess temperature operator alarms, indicating control rod location, and to provide graphic displays for the operator to observe during events such as conduction cooldown. Additionally, the RSCEs (which are included in the outer NCAs, but not the startup NCAs) are instrumented to provide measurement of fuse link continuity and hopper gate open close status for display on the Reactor Protection System (RPS) operator console. Testing will be completed to verify manufacturers data for some of the components selected in the CD, and to confirm level 5 technical readiness.

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Coordinate with interfacing design areas – SRD with Reactor System, Reactor Internals, and Vessel System; Power Range detectors with Reactor System and Reactor Building; IFMU and NCA with Reactor System, Reactor Internals, and Vessel System – to provide supporting design analysis of component operating conditions, to complete interrelated design efforts (e.g. NCA control rod, guide-tubes etc. development under Reactor Internals), and to assure compatibility of interrelated components, such as consideration that fluid flow through the guide tubes and around the control rods for cooling is also adequate to protect the control rod drive mechanism at a different location in the NCA. Resolve design issues which do not require component testing.	GA	CD 0-36mo	500
3. List all design issues which do require component testing and determine tests required. (NOTE: For design issues from pre-existing conceptual design work see DDN(s) Supported, on page 1.) For the components being used, or considered for use, in the SRD, IFMU, Power Range neutron detector, and NCA/RSCE designs, identify all data which is marginal or questionable, and requires verification testing at the Experimental Scale (ES). From the list of considerations below, applying experience gained during CD to modify the list, develop a verification process and prepare test facilities to resolve design issues at the component test level, and to verify or extend component data which was supplied by manufacturers. <u>Neutron Detector Assemblies:</u> Fission chamber devices used in the SRD, IFMU, and Power Range neutron detectors require design specific range, response time, maximum operating temperature and pressure margins, duty-cycle and lifetime capabilities, etc. In addition, SRD and IFMU designs require movement to operate and movement during operation. Manufacturers may not supply this	GA Vendor(s) Facility	CD 12-36mo CD 12-24mo CD 12-24mo	300 500 700

information, and ES testing will be needed to provide the data. All the detectors require handling operations for maintenance and inspection, as well. Instrument cabling and electronics, associated with each of the detectors, must also meet the handling and operating requirements of the detector itself. For example, IFMU instrumentation cabling (for neutron detectors and thermocouples located in the movable detector assemblies) must be extended and retracted while the detector assembly travels through the reactor. <u>Drive Mechanisms and Controls:</u> The SRD, IFMU, and NCA motor driven operating mechanisms require design specific torque, speed, minimal stop start travel increments, maximum operating temperature and pressure margins, duty-cycle and lifetime capabilities, etc. In addition, all may require testing of particular motor loading extremes associated with guide-tube misalignment, caused by temperature effects or vessel and core misalignment, including misalignment of individual core components under various conditions of reactor operation. Attached motor power and controller cabling and electronics, switches, etc. must meet the requirements of the motor itself. In addition, the NCA requires instrumentation to indicate motor start stop status, cable tension for each control rod, rod full-in or full-out indication, operating temperature, etc. Instrumentation placed near the motors may require testing to evaluate the effects of electrical noise. Most of the SSC-1 equipment is safety-related, and consideration of this should be included in all test planning. This requires Safe Shutdown Earthquake (SSE) seismic testing prior to installation of the systems. CD recommendations to verify some component reliability issues might be included in ES component testing; otherwise, reliability testing will occur at the pilot scale. <u>Support Structures, Movement Guidance Structures, Pressure Seals, Insulation, and Shielding</u>: Some SRD, IFMU, Power Range detector, or NCA components, which fall into the above categories, may require additional test data to achieve a level 5 technical rating. However, it is assumed that most CD component selections will specify documentation assuring qualification of materials and small components, such as pressure seals, to operate in the neutron flux environment at the required operating conditions. In the pre-existing design work, mentioned above, only the IFMU appears to require gamma shielding to protect personnel. The other systems may need this requirement as well. Also, support cables,	GA	CD 24-36mo	500

rods, tubing, pressure seals, structures, etc. which experience changes in temperature, pressure, alignment, etc. associated with movement of devices, during performance of the specific SRD, IFMU, and NCA functions, must be tested at the most extreme conditions, with consideration of the required operating lifetime. 4. Complete experimental-scale testing as determined above, make design adjustment and repeat testing, if required. Also, repeat testing of other components where inter-dependencies might occur. Document results to confirm level 5 technical rating. Provide recommendations for testing at next technical rating level.	Vendor(s) Facility	CD 12-36mo CD 12-36mo	1,000 1,500

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-1.2	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Reactor Control and Protection, Reactor Control Equipment (SSC-1)

Description:
SSC-1 contains equipment associated with control and measurement of reactor processes. This includes the Neutron Control Assembly (NCA), which contains control rod drive equipment and instrumentation. Reserve Shutdown Control Equipment (RSCE) is contained in the NCAs that operate the outer control rods. SSC-1 also includes other nuclear instrumentation - in-core Flux Mapping Units (IFMU), the Source Range Neutron Detectors (SRD), and the Power Range neutron detectors. The Power Range neutron detectors are located in six wells, equally spaced around the Reactor Vessel, in the Reactor Building concrete structure behind the RCCS. (Cont.)

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Verified at bench scale	Verified at experimental scale	Verified at pilot scale
TRL	**4**	**5**	**6**

Basis for Rating (Check box if continued on additional sheets) ☒

The level 5 technical rating for SSC-1 is based on completion of activities required to achieve a level 5 technical rating. This work included testing of components in the SRD, IFMU, Power Range neutron detectors, NCA, and RSCE systems. These components are contained in Reactor Control and Protection systems which are included in the reactor control equipment design. Prominent subsystems are neutron detector assembly subsystems; drive mechanism, controls, instrumentation, and support structure subsystems; and movement guidance subsystems. (Cont)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Complete NHSS preliminary Final Design (FD) of SRD, Power Range ex-vessel neutron detector, IFMU, NCA, and NCA with Reserve Shutdown equipment. Provide subsystem and final assembly views and supporting analysis to determine operating conditions for each subsystem. Document design issues. (Cont)	GA	FD 0-42mo	3,000

DDN(s) Supported: C.11.10.01, C.11.01.02, C.11.01.07, C.11.01.03, C.11.01.04, C.11.01.05, C.11.01.06, C.11.02.01	**Technology Case File:**

Subject Matter Expert Making Determination: Dale Pfremmer

Date: 10/24/08	**Originating Organization:** General Atomics

Additional Description
Description: Component level development testing parallels Reactor Internals testing to develop control rod materials, guide tubes, and other Reactor Internals components. Likewise, the nuclear instrumentation design requires considerable interaction with Reactor System, Reactor Internals, and Vessel System development efforts. SSC-1 testing at the subsystem level requires interaction with all these development efforts. SSC-1 testing of the installed NCA, SRD, and IFMU systems will be needed to confirm hot startup readiness. These tests involve operation of the equipment from the control room, utilizing the Plant Control, Data and Instrumentation System (PCDIS). Reactor Control and Protection development testing therefore includes testing of NCA, SRD, and IFMU operational functions for hot startup readiness, as well as testing at the subsystem design level.

Additional Basis Sheet(s)

Basis:

Component testing was done in conjunction with interfacing system groups, including Reactor System, Reactor Internals, Vessel System, and the Reactor Building System. Design issues which could not be resolved at the component testing level may require testing of pilot scale configurations to resolve issues of operability, reliability, and failure effects to achieve a level 6 technical rating.

Operability, reliability, and failure effects issues for the SRD, IFMU, and NCA-RSCE subsystems are resolved during preliminary Final Design (PFD), and if necessary include pilot scale testing. System by system test planning considerations are mentioned below;

The SRDs operate through the lower vessel and must be mounted in a fashion which allows removal and replacement of the entire assembly. SRD neutron detectors are withdrawn through the lower vessel to prevent premature burn-up of fissile material contained in the detector, and remain withdrawn during all but startup, normal shutdown or reactor trip, and refueling operations. They require drive mechanisms and controls to operate, and these rely on alignment considerations affecting both the vessel design and lower reflector. They are required to operate during conduction cooldown events. Normal life expectancy is approximately 5 years. Both operability and reliability should be verified at the subsystem level and effects of various drive mechanism failure on overall reactor operability should be considered as well.

The Power Range neutron detectors are permanently mounted, and may not require pilot scale subsystem testing.

The IFMUs are also movable neutron detection devices (but also contain temperature instrumentation, whereas the SRDs do not) and contain a drive mechanism subsystem which lowers the detector assemblies into the reactor. However, these operate through the top-head, as do the NCAs. The IFMUs operate only periodically, but the NCAs operate the control rods, and have a more severe duty-cycle. Both the IFMU and NCA systems have drive mechanisms. The NCA contains instrumentation in the drive mechanism enclosure. This includes temperature instrumentation and possibly contact switches or other devices to determine and verify full out or full in positioning of individual control rods. The IFMU has instrumentation cabling attached through the drive mechanism enclosure to the detector assembly, which travels axially through the guide system in the reactor. This, in turn, requires extension/retraction of instrumentation cabling. RSCEs are included in the outer NCAs, but not the startup NCAs, so these NCAs contain a different subsystem. The mounting structure for these subsystems interfaces with the vessel top-head. Associated instrumentation and power cabling, entering the enclosure into the drive motor area is also a consideration in the vessel top-head design. Both drive mechanisms require suppression of hot core inlet gas heating effects, and this is a concern to other parts of the system (such as the guide-tubes) as well. Subsystem operability must be verified. Sub-assembly drawings and accompanying analysis from the preliminary Final Design provide operating conditions and the arrangement of each subsystem. Component testing and analysis contribute, but subsystem testing may also be required to verify certain operability considerations such as the requirement that drive mechanisms must maintain movement of the control rods, or the IFMU detector assembly, by gravity force through guide-tubes under abnormal, as well as normal, conditions of reactor operation. (Loss of flow, over-temperature, conduction cooldown scenarios, etc.) Accelerated life testing may also be needed to verify reliability. Failure effects also may be needed for NCA and IFMU drive subsystems.

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Coordinate with interfacing design areas – SRD with Reactor System, Reactor Internals, and Vessel System; Power Range detectors with Reactor System and Reactor Building; IFMU and NCA with Reactor System, Reactor Internals, and Vessel System – to provide supporting data and stress analysis to verify drive mechanism and detector assembly operating integrity. Complete interrelated subsystem development efforts. Resolve design issues which do not require specific subsystem testing, using analysis or test data from qualified similar applications. (For example, some aspects of NCA movement guidance structure analysis/testing might be applicable to IFMUs as well.)	GA	FD 0-12mo	500
3. List all design issues which do require subsystem testing and determine tests required. Coordinate with Reactor Internals, Vessel System, and Reactor System interrelated design areas. (NCA development, under Reactor Internals, includes control rod guide-tubes and the control rods.) Identify the SRD, IFMU, and NCA/RSCE subsystems which require reliability verification testing at the Pilot Scale. Prepare test facilities for drive mechanisms, detector subsystems, etc. using representative versions of final design. The following consideration should be included in preparing facilities for testing operability, reliability, and failure modes. <u>Detector Assembly Subsystems:</u> SRD, IFMU, and Power Range neutron detectors which require design specific range, response time, maximum operating temperature and pressure margins, duty-cycle and lifetime capabilities, etc. will have been tested, or verified, at the component level and should not require subsystem testing to verify these capabilities. However, SRD and IFMU detector subsystems require movement to operate and movement during operation. These features can be confirmed at the subsystem level to assure subsystem reliability, within design lifetime, operating conditions, etc. Failure modes affecting plant operation or which cause effects in interfacing design areas (Vessel, Reactor Internals, etc.) should also be considered. <u>Drive Mechanisms Subsystems:</u> The SRD, IFMU, and NCA motor driven operating mechanisms, at the subsystem level, require a representative version of the final design including	GA Vendor(s) Facility	FD 12-30mo FD 24-30mo FD 12-30mo	300 500 700

gearing, cables and pulleys, pushrods, motor and instrumentation support structures, etc sufficient to test torque, speed, minimal stop start travel increments, etc. under maximum operating temperature and pressure conditions, with cables, etc. attached. Duty-cycle and lifetime capabilities, etc. may incorporate additional testing of particular motor loading extremes associated with guide-tube misalignment, core misalignment, etc. Testing should include attached motor power and controller cabling and electronics, switches, etc., as well as instrumentation included in the NCA to measure control rod and motor enclosure parameters. It is expected that testing to evaluate the effects of electrical noise on instrumentation can be done better at the subsystem level. Test documentation should support safety-related qualification of SSC-1 equipment. Support Structures, Movement Guidance Structures, Pressure Seals, Insulation, and Shielding: Most, if not all, SRD, IFMU, or NCA components, which fall into the above category, may require no additional testing at the subsystem level, since it is assumed that most small components will achieve a level 6 rating by inclusion in subsystem testing. Movement guidance structures may be one exception. It may be necessary to separate testing of this portion of the NCA or IFMU from drive mechanism testing, for example. In this case, that portion of the subsystem must be tested at the most extreme conditions, with consideration of the required operating lifetime, etc. just as would be the case were it included as part of the drive mechanism subsystem testing. Also considered at the subsystem level, are various equipment handling systems. While other features of the handling systems probably don't require testing below level 7, it may be desirable to verify attachment/pick-up features of handling systems at the subsystem level. 4. Complete Pilot Scale testing as determined above. Make design adjustment and repeat testing, if required. Document results to confirm level 6 technical rating. Provide recommendations for pre-installation integrated system level testing of SRD, Power Range ex-vessel neutron detector, IFMU, NCA, and NCA with Reserve Shutdown equipment.	GA Vendor(s) Fabricators Facility	FD 30-42mo FD 30-36mo FD 30-36mo FD 30-42mo	500 300 1,400 1,800

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-1.3	**Revision:** 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Reactor Control and Protection, Reactor Control Equipment (SSC-1)

Description:
SSC-1 contains equipment associated with control and measurement of reactor processes. This includes the Neutron Control Assembly (NCA), which contains control rod drive equipment and instrumentation. Reserve Shutdown Control Equipment (RSCE) is contained in the NCAs that operate the outer control rods. SSC-1 also includes other nuclear instrumentation - in-core Flux Mapping Units (IFMU), the Source Range Neutron Detectors (SRD), and the Power Range neutron detectors. The Power Range neutron detectors are located in six wells, equally spaced around the Reactor Vessel, in the Reactor Building concrete structure behind the RCCS. (Cont.)

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Verified at experimental scale	Verified at pilot scale	Verified at engineering scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☒

The level 6 technical rating for SSC-1 is based on completion of activities required to achieve a level 6 technical rating. This work included testing of subsystems in the SRD, IFMU, Power Range neutron detectors, NCA, and RSCE systems. These subsystems are contained in Reactor Control and Protection systems which are included in the reactor control equipment design. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Complete NHSS Final Design (FD) of SRD, Power Range ex-vessel neutron detector, IFMU, NCA, and NCA with Reserve Shutdown equipment. Fabricate equipment and provide as-built drawings showing final assembly views, sub-assembly views, control and instrumentation diagrams, etc. and supporting documentation to allow assembly, installation, test-point hookup procedures for test instruments, etc. Document pre-installation issues. (Cont.)	GA Fabricators	FD 43-84mo FD 48-54mo	2,500 33,000

DDN(s) Supported: C.11.01.02, C.11.01.09, C.11.02.01	**Technology Case File:**

Subject Matter Expert Making Determination: Dale Pfremmer

Date: 10/24/08	**Originating Organization:** General Atomics

Additional Description
Description: Component level development testing parallels Reactor Internals testing to develop control rod materials, guide tubes, and other Reactor Internals components. Likewise, the nuclear instrumentation design requires considerable interaction with Reactor System, Reactor Internals, and Vessel System development efforts. SSC-1 testing at the subsystem level requires interaction with all these development efforts. SSC-1 testing of the installed NCA, SRD, and IFMU systems will be needed to confirm hot startup readiness. These tests involve operation of the equipment from the control room, utilizing the Plant Control, Data and Instrumentation System (PCDIS). Reactor Control and Protection development testing therefore includes testing of NCA, SRD, and IFMU operational functions for hot startup readiness, as well as testing at the subsystem design level.

Additional Basis Sheet(s)

Basis:

Prominent subsystems are neutron detector assembly subsystems, drive mechanism subsystems, and movement guidance subsystems. Subsystem testing was done in conjunction with interfacing system groups, including Reactor System, Reactor Internals, Vessel System, and the Reactor Building System. Issues such as reliability, failure effects, etc. have been resolved by pilot scale testing using representative configurations to test operability, perform accelerated life testing, and determine failure modes at a level 6 technical rating. The final design is completed under the level 6 technical rating, but demonstration of installation readiness requires further testing. For example, an integrated engineering scale demonstration of specific features such as SRD, IFMU or NCA extension and retraction operations requires facilities and procedures to perform the testing. The actions below address design efforts and testing to achieve a level 7 technical rating for SSC-1 equipment. Related handling equipment will also be evaluated at level 7. Seismic testing for safety-related qualification of SSC-1 equipment is also completed at level 7.

Specific test planning considerations are mentioned below:

Since the SRDs operate the SRD neutron detectors through the lower vessel, an integrated test configuration must be devised to assure alignment, retrieval, etc. The vessel and lower reflector are involved. It may be desirable to coordinate test activities with these design areas to verify the alignment aspects prior to installation. The IFMUs and NCAs also require alignment verification. This can probably be accomplished with checkout of the handling machines. The SRDs, IFMUs, and NCAs all interface with the vessel and therefore must maintain all requirements for vessel integrity, including leakage tightness. Testing to verify this may previously exist under the vessel design scope, but this should be verified and documented as part of the SSC-1 installation readiness process.

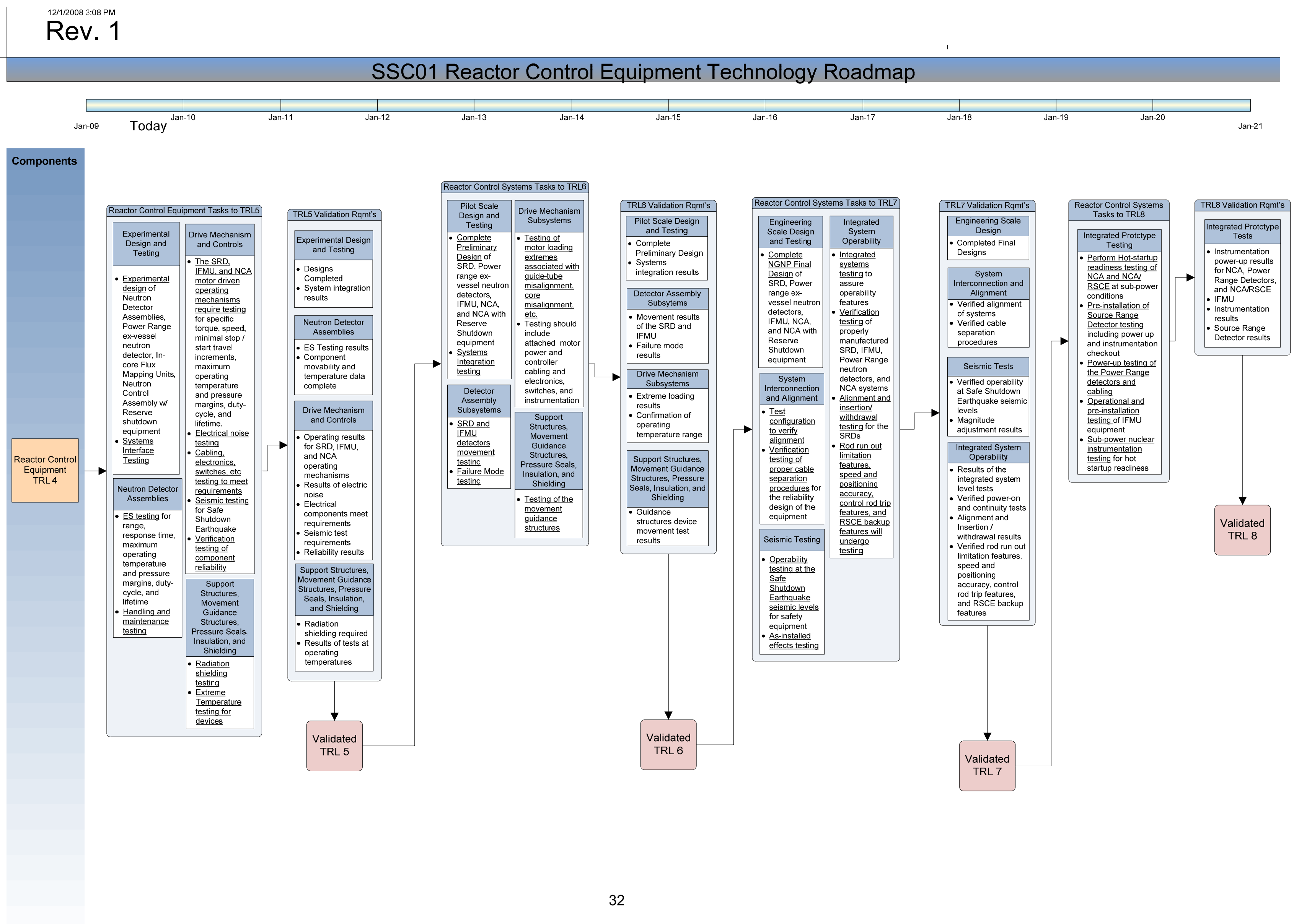
All SSC-1 systems, including the Power Range neutron detectors, have power and instrumentation wiring. Test procedures to verify power cable and instrumentation readiness are needed as well. These would include testing to verify subsystem power-up, at the integrated system level. (Including operation from the actual or representative control consoles.) In particular, the NCA equipment has rigorous safety-related design requirements. Verification of rod runout limitation features, power cable and channel separation features, drive mechanism failure-detection features, etc. must be provided. Tests requiring end-to-end power cable and control access to simulate NCA operational and failure protection features, which cannot or should not be tested prior to installation of NCA equipment, must be completed beforehand. Examples of this include testing the control rod trip operation (under simulated controller failures resulting in a rod runout, reactor exit over-temperature, etc.), and RSCE release of the boron balls. An above-reactor test rig (possibly on the refueling floor) may be required to accomplish this testing.

The IFMUs also contain movable equipment and may require testing similar to the NCAs. However, the IFMU may require only minimal verification of operation functions. It may be reasonable to verify IFMU operation more fully after installation, prior to hot startup. Verification of IFMU handling equipment functions will be required before installation.

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Coordinate with interfacing design areas – SRD with Reactor System, Reactor Internals, and Vessel System; Power Range detectors with Reactor System and Reactor Building; IFMU and NCA with Reactor System, Reactor Internals, and Vessel System – to provide an all-design-area test requirement summary for pre-installation checkout of each system, with design area responsibility included.	GA	FD 43-84mo	1,000
3. Resolve issues which do not require testing, using all available information. Document resolution of issues for advancement to level 7.	GA	FD 43-54mo	500
4. List all issues which do require testing and determine tests required, with participation from Reactor Internals, Vessel System, Reactor Building, Reactor System and BOP engineering design areas.	GA	FD 54-60mo	200
5. Prepare test facilities for SRD, IFMU, and NCA equipment, and Power Range neutron detector equipment, if necessary. All associated handling equipment should be checked. The following primary areas of verification and testing are expected in preparing the SSC-1 systems for installation: <u>System Interconnection and Alignment:</u> Since the SRD neutron detectors operate through the lower vessel, a test configuration must be devised to verify proper alignment during installation. This may require observation of the SRD insertion/withdrawal process with the lower vessel and internals partially assembled. This might be accomplished during checkout of SRD removal and handling equipment, to verify that no binding or bending of the overall SRD assembly could impair the operating function. The IFMU and NCA assemblies have similar considerations, but pre-installation testing may not be required. However, checkout of the handling equipment is required, as well as checkout of maintenance equipment associated with the Power Range neutron detector assemblies. The SRD, IFMU and NCA systems each seal their respective vessel penetrations to prevent primary coolant leakage during operation. It is assumed that this requirement will be verified during Vessel System checkout. The Vessel System, Reactor Building, and AE (Architect Engineer) checkout of electric power wiring must verify cable harnesses, cable tray attachments, etc for each of these systems. This also includes verification of proper cable separation	GA Facility	FD 54-60mo FD 54-72mo	300 1,000

procedures for the reliability design of the equipment. <u>Integrated System Operability:</u> The SRD, IFMU, and NCA operating mechanisms, powered instrumentation, etc. which were tested at the subsystem level, will require additional testing at the integrated system level to assure operability features which could not be demonstrated fully at the subsystem level. Integrated system level testing is better suited to fully check power and power transfer/control mechanisms, instrumentation and power cabling, etc. Also, systems can be connected to allow activation of system functions from the actual command consoles. It is assumed that simple point-of-fabrication procedures will have been completed to verify proper manufacturing of the SRD, IFMU, Power Range neutron detectors, and NCA systems. These will include equipment power-on tests, continuity checks, etc. However, minimal, special purpose testing equipment may be required for these tests as well. After delivery of prototype units, more testing is required. The SRDs require alignment and insertion/withdrawal tests (see above). The NCAs require verification of rod runout limitation features, speed and positioning accuracy, control rod trip features, and RSCE backup features (release of boron balls). An above-reactor test rig (possibly on the refueling floor) will be required to accomplish this testing. The normal features of control rod withdrawal and insertion should be demonstrated also. IFMU testing may be required as well, but some IFMU prototype testing could be accomplished with checkout of IFMU handling equipment (assuming IFMU placement in and removal from the reactor will be included), or this testing could be deferred to level 8. NCA testing is, however, limited after installation and prior to hot startup, so the above testing is required outside the reactor. Test documentation should be provided to support safety-related qualification of SSC-1 equipment. <u>Seismic Testing</u>: Seismic testing of SSC-1 systems is required to achieve a level 7 technical rating. These tests will be accomplished in a nuclear qualified facility. Special test structures to attach equipment and produce as-installed seismic effects, or amplification of the seismic effects to represent the as-installed effects, will be required. Operability at Safe Shutdown Earthquake (SSE) seismic levels must be demonstrated for safety-related equipment. The SSE magnitude is twice the Operational Basis Earthquake (OBE) magnitude, but	GA	FD 54-84mo	500

the OBE requirement applies to all equipment, and requires that all equipment needed to operate the reactor must continue to operate. Therefore, temporary relocation of supporting test equipment to seismic test facilities will be necessary. Test documentation from seismic testing must be provided to support SSE and OBE qualification of the equipment. 6. Determine the Engineering Scale testing, prepare test facilities, and complete testing. If equipment adjustments are necessary, repeat testing after adjustments are completed. Provide equipment change information to manufacturing, modify as-built drawings, and assure that all levels of Quality Assurance are repeated in the process. Document results to confirm level 7 technical rating. Provide recommendations for after-installation-testing of SRD, Power Range ex-vessel neutron detector, IFMU, NCA, or NCA with Reserve Shutdown equipment, which should be completed prior to hot startup.	Facility Fabricator s Seismic	FD 54-78mo FD 54-78mo FD 78-84mo	4,000 1,000 3,000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Rev. 1
SSC01 Reactor Control Equipment Technology Roadmap
Jan-09
Today
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Components
Reactor Control Equipment TRL 4
Reactor Control Equipment Tasks to TRL5
Experimental Design and Testing
• Experimental design of Neutron Detector Assemblies, Power Range ex-vessel neutron detector, In-core Flux Mapping Units, Neutron Control Assembly w/ Reserve shutdown equipment
• Systems Interface Testing
Neutron Detector Assemblies
• ES testing for range, response time, maximum operating temperature and pressure margins, duty-cycle, and lifetime
• Handling and maintenance testing
Drive Mechanism and Controls
• The SRD, IFMU, and NCA motor driven operating mechanisms require testing for specific torque, speed, minimal stop / start travel increments, maximum operating temperature and pressure margins, duty-cycle, and lifetime.
• Electrical noise testing
• Cabling, electronics, switches, etc testing to meet requirements
• Seismic testing for Safe Shutdown Earthquake
• Verification testing of component reliability
Support Structures, Movement Guidance Structures, Pressure Seals, Insulation, and Shielding
• Radiation shielding testing
• Extreme Temperature testing for devices
TRL5 Validation Rqmt's
Experimental Design and Testing
• Designs Completed
• System integraion results
Neutron Detector Assemblies
• ES Testing results
• Component movability and temperature data complete
Drive Mechanism and Controls
• Operating results for SRD, IFMU, and NCA operating mechanisms
• Results of electric noise
• Electrical components meet requirements
• Seismic test requirements
• Reliability results
Support Structures, Movement Guidance Structures, Pressure Seals, Insulation, and Shielding
• Radiation shielding required
• Results of tests at operating temperatures
Validated TRL 5
Reactor Control Systems Tasks to TRL6
Pilot Scale Design and Testing
• Complete Preliminary Design of SRD, Power range ex-vessel neutron detectors, IFMU, NCA, and NCA with Reserve Shutdown equipment
• Systems Integration testing
Detector Assembly Subsystems
• SRD and IFMU detectors movement testing
• Failure Mode testing
Drive Mechanism Subsystems
• Testing of motor loading extremes associated with guide-tube misalignment, core misalignment, etc.
• Testing should include attached motor power and controller cabling and electronics, switches, and instrumentation
Support Structures, Movement Guidance Structures, Pressure Seals, Insulation, and Shielding
• Testing of the movement guidance structures
TRL6 Validation Rqmt's
Pilot Scale Design and Testing
• Complete Preliminary Design
• Systems integration results
Detector Assembly Subsystems
• Movement results of the SRD and IFMU
• Failure mode results
Drive Mechanism Subsystems
• Extreme loading results
• Confirmation of operating temperature range
Support Structures, Movement Guidance Structures, Pressure Seals, Insulation, and Shielding
• Guidance structures device movement test results
Validated TRL 6
Reactor Control Systems Tasks to TRL7
Engineering Scale Design and Testing
• Complete NGNP Final Design of SRD, Power range ex-vessel neutron detectors, IFMU, NCA, and NCA with Reserve Shutdown equipment
System Interconnection and Alignment
• Test configuration to verify alignment
• Verification testing of proper cable separation procedures for the reliability design of the equipment
Seismic Testing
• Operability testing at the Safe Shutdown Earthquake seismic levels for safety equipment
• As-installed effects testing
Integrated System Operability
• Integrated systems testing to assure operability features
• Verification testing of properly manufactured SRD, IFMU, Power Range neutron detectors, and NCA systems
• Alignment and insertion/ withdrawal testing for the SRDs
• Rod run out limitation features, speed and positioning accuracy, control rod trip features, and RSCE backup features will undergo testing
TRL7 Validation Rqmt's
Engineering Scale Design
• Completed Final Designs
System Interconnection and Alignment
• Verified alignment of systems
• Verified cable separation procedures
Seismic Tests
• Verified operability at Safe Shutdown Earthquake seismic levels
• Magnitude adjustment results
Integrated System Operability
• Results of the integrated system level tests
• Verified power-on and continuity tests
• Alignment and Insertion / withdrawal results
• Verified rod run out limitation features, speed and positioning accuracy, control rod trip features, and RSCE backup features
Validated TRL 7
Reactor Control Systems Tasks to TRL8
Integrated Prototype Testing
• Perform Hot-startup readiness testing of NCA and NCA/ RSCE at sub-power conditions
• Pre-installation of Source Range Detector testing including power up and instrumentation checkout
• Power-up testing of the Power Range detectors and cabling
• Operational and pre-installation testing of IFMU equipment
• Sub-power nuclear instrumentation testing for hot startup readiness
TRL8 Validation Rqmt's
Integrated Prototype Tests
• Instrumentation power-up results for NCA, Power Range Detectors, and NCA/RSCE
• IFMU
• Instrumentation results
• Source Range Detector results
Validated TRL 8

4.2 SSC-2 Control Rods

TRL Rating Sheets, TRL 2 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-2.1.1	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Reactor Internals-Control Rods (CR)

Description:
Control Rods (CR) are located in 2 areas of the reactor core: near the inner boundary of the fuel and central replaceable reflector elements (12 rods); and near the outer boundary between the fuel and outer replaceable reflector reflectors (18 rods). The outer CR are used to control the power in the core and are inserted during normal operation. The inner CR are withdrawn during normal operation and are only used to shut down the nuclear reaction. The rod is a linear assembly of rigid links filled with boronated graphite compacts within a cylindrical sleeve. The joints between rigid links allow flexibility in the rod assembly. The sleeves and joints are the structural elements that contain the nonstructural absorber compacts and transfer the operational loads to the control rod drive. All control rods are identical to accommodate interchangeability.

Area:	☒ NHSS	☐ HTS	☐ HPS	☐ BCS	☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Basic principles observed	Application formulated	Proof of concept
TRL	**1**	**2**	**3**

Basis for Rating (Check box if continued on additional sheets) ☐

The CR design will be essentially the same as in Ft St Vrain but the higher temperatures to which the CR will be subjected during conduction cooldown events in the NGNP require use of a ceramic composite as the material of construction for the structural components. Ceramic composite materials are widely used in aerospace but little data is available on irradiation effects and corrosion in an impure He environment, so a substantial materials development program is needed (see HTR2008 conference paper HTR2008-58050 and GA Report 911125/0). Composite architectures specific to the geometries of the various CR structural components must be developed.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Perform engineering analyses to establish control rod operating conditions (e.g., temperatures, flow conditions, helium impurities, etc.) and develop control rod requirements	General Atomics (GA)	6 months starting near beginning of CD	350

DDN(s) Supported: N.11.03.53, N.11.03.54, N.11.03.55, N.11.03.56, C.11.03.24	**Technology Case File:**
Subject Matter Expert Making Determination: John Saurwein, Russ Vollman	
Date: 11-27-08	**Originating Organization:** General Atomics

Additional Actions Sheets(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Develop control rod conceptual design. Perform FEA and seismic analyses to calculate the expected mechanical and seismic loads. Initiate development of CR composite material performance models	GA	3 months starting after completion of action 1	200
3. Review ceramic composite materials knowledge base and composite materials supply network to select potential composite materials and parts manufacturers	GA/Rolls-Royce	6 months starting after completion of action 1	350
4. Develop composite architectures and manufacturing processes for the CR parts. Fabricate shapes having the selected composite architectures and cut out specimens for the tests in actions 5, 6, and 7	GA/Rolls-Royce and part manufacturers	1 year, complete by ~middle of PD	3000
5. Conduct baseline physical and mechanical properties tests on test specimens from action 4.	ORNL, INL, and/or commercial laboratories	1 year, complete by end of PD	1000
6. Conduct screening irradiation tests on test specimens from action 4 to determine irradiation induced dimensional changes and creep and to determine the effect of irradiation on the baseline physical and mechanical properties (from action 5)	INL and/or ORNL	3 years, complete 2 years into NGNP FD	TBD (A very rough estimate is ~$20M)
7. Conduct screening corrosion tests on test specimens from action 4 in a reactor helium environment at reactor operating temperatures (up to ~1500°C) to determine the effects on the baseline physical and mechanical properties (from action 5)	INL and/or ORNL	2 years starting in parallel with action 6	1000
8. Select composite materials and architectures	GA/Rolls-Royce and parts manufacturers	3 months starting as soon as data are available from actions 6 & 7	200

TRL Rating Sheet

Vendor: GA	**Document Number:** SCC-2.1.2	**Revision:** 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: **Reactor Internals-Control Rods (CR)**

Description:
Control Rods (CR) are located in 2 areas of the reactor core: near the inner boundary of the fuel and central replaceable reflector elements (12 rods); and near the outer boundary between the fuel and outer replaceable reflector reflectors (18 rods.) The outer CR are used to control the power in the core and are inserted during normal operation. The inner CR are withdrawn during normal operation and are only used to shut down the nuclear reaction. The rod is a linear assembly of rigid links filled with boronated graphite compacts within a cylindrical sleeve. The joints between rigid links allow flexibility in the rod assembly. The sleeves and joints are the structural elements that contain the nonstructural absorber compacts and transfer the operational loads to the control rod drive. All control rods are identical to accommodate interchangeability.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Application formulation	Proof of concept	Verified at bench scale
TRL	**2**	**3**	**4**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 3 is achieved when the following conditions are met: (1) All of the engineering analyses and design activities and all of the testing activities required to advance the TRL from 2 to 3 have been completed; (2) Composite architectures have been selected based on the screening baseline physical and mechanical properties tests, irradiation tests, and corrosion tests on candidate composite architectures; and (3) The results of the tests on the selected composite architectures show that CRs fabricated from these materials have a high probability of satisfying CR design requirements.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Finalize the composite architectures for the CR structural parts. Fabricate prototype parts and cut samples from the parts for actions 2, 3, and 4 below.	GA/Rolls-Royce, and parts manufacturers	6 months starting as soon composite architectures are selected	2000

DDN(s) Supported: N.11.03.53, N.11.03.54, N.11.03.55, N.11.03.56, C.11.03.24	**Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein, Russ Vollman

Date: 11-27-08	**Originating Organization:** General Atomics

Additional Action Sheet(s):			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Conduct baseline physical and mechanical properties tests on test specimens from action 1. The number of tests performed shall be adequate to obtain the data required for the statistically significant engineering data base required for the composite material performance models	INL, ORNL, and/or commercial laboratories	1 year starting as soon as test specimens from action 1 are available	1000
3. Conduct irradiation tests on test specimens from action 1 to determine irradiation induced dimensional changes and creep and to determine the effect of irradiation on the baseline physical and mechanical properties (from action 2). The testing shall be sufficient to establish a statistically significant engineering data base for the composite material performance models	INL and/or ORNL	2 years, must be completed by ~mid 2017 to support NGNP startup in 2021	TBD (a very rough estimate is ~$20m
4. Conduct corrosion tests on test specimens from action 1 in a reactor helium environment at reactor operating temperatures (up to ~1500°C) to determine the effects on the baseline physical and mechanical properties (from action 2)	INL and/or ORNL	2 years, must be completed by ~mid 2017 to support NGNP startup in 2021	2000
5. Complete composite material behavior and failure models based on the data from actions 2, 3, and 4	GA	6 months starting as soon as data from actions 2, 3, and 4 are available	400

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-2.1.3	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Reactor Internals-Control Rods (CR)

Description:
Control Rods (CR) are located in 2 areas of the reactor core: near the inner boundary of the fuel and central replaceable reflector elements (12 rods); and near the outer boundary between the fuel and outer replaceable reflector reflectors (18 rods). The outer CR are used to control the power in the core and are inserted during normal operation. The inner CR are withdrawn during normal operation and are only used to shut down the nuclear reaction. The rod is a linear assembly of rigid links filled with boronated graphite compacts within a cylindrical sleeve. The joints between rigid links allow flexibility in the rod assembly. The sleeves and joints are the structural elements that contain the nonstructural absorber compacts and transfer the operational loads to the control rod drive. All control rods are identical to accommodate interchangeability.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Proof of concept	Verified at bench scale	Verified at engineering scale
TRL	**3**	**4**	**5**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 4 is achieved when the following conditions are met: (1) The composite architectures for the CR structural parts have been finalized and prototype parts have been fabricated and cut up for testing; (2) Adequate physical and materials properties testing of unirradiated, irradiated, and corrosion specimens from the prototype parts have been performed to establish a statistically significant material properties engineering data base; and (3) Composite material behavior and failure models have been completed based on the composite properties engineering data base.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Conduct engineering analysis to verify that components meet design and safety requirements, including thermal-hydraulic, corrosion and stress, dynamic and seismic, life, reliability, and maintainability.	General Atomics	12 months starting about half-way through FD	1000

DDN(s) Supported: None **Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein, Russ Vollman

Date: 11-27-08 **Originating Organization:** General Atomics

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-2.1.4	**Revision:** 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: **Reactor Internals-Control Rods (CR)**

Description:
Control Rods (CR) are located in 2 areas of the reactor core: near the inner boundary of the fuel and central replaceable reflector elements (12 rods); and near the outer boundary between the fuel and outer replaceable reflector reflectors (18 rods). The outer CR are used to control the power in the core and are inserted during normal operation. The inner CR are withdrawn during normal operation and are only used to shut down the nuclear reaction. The rod is a linear assembly of rigid links filled with boronated graphite compacts within a cylindrical sleeve. The joints between rigid links allow flexibility in the rod assembly. The sleeves and joints are the structural elements that contain the nonstructural absorber compacts and transfer the operational loads to the control rod drive. All control rods are identical to accommodate interchangeability.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Verified at bench scale	Verified at experimental scale	Verified at pilot scale
TRL	**4**	**5**	**6**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 5 is achieved once engineering analyses have been completed and show that the control rod design meets design and safety requirements including thermal-hydraulic, corrosion and stress, dynamic and seismic, life, reliability, and maintainability requirements.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Conduct engineering analysis to show that the control rods can be inserted into the guide tubes and core graphite elements without interference for all normal and off-normal events and that the design helium coolant flow through the guide tubes, core graphite elements, and around the control rods will be adequate for cooling.	GA	6 months. Must be complete about 1.5 years before end of final design	200

DDN(s) Supported: None	**Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein, Russ Vollman

Date: 11-27-08	**Originating Organization:** General Atomics

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-2.1.5	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: **Reactor Internals-Control Rods (CR)**

Description:
Control Rods (CR) are located in 2 areas of the reactor core: near the inner boundary of the fuel and central replaceable reflector elements (12 rods); and near the outer boundary between the fuel and outer replaceable reflector reflectors (18 rods). The outer CR are used to control the power in the core and are inserted during normal operation. The inner CR are withdrawn during normal operation and are only used to shut down the nuclear reaction. The rod is a linear assembly of rigid links filled with boronated graphite compacts within a cylindrical sleeve. The joints between rigid links allow flexibility in the rod assembly. The sleeves and joints are the structural elements that contain the nonstructural absorber compacts and transfer the operational loads to the control rod drive. All control rods are identical to accommodate interchangeability.

Area:	☒ NHSS	☐ HTS	☐ HPS	☐ BCS	☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Verified at experimental scale	Verified at pilot scale	Verified at engineering scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 6 is achieved upon completion of engineering analyses that show that the control rods can be inserted into the guide tubes and core graphite elements without interference for all normal and off-normal events and that the design helium coolant flow through the guide tubes, core graphite elements, and around the control rods will be adequate for cooling.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Conduct vibration testing of a single full-scale control rod inside a guide tube and inside a column of graphite control-rod fuel elements.	GA, Wyle Laboratories, Hazen Research, or INL CTF	18 months, must be completed by first quarter of 2020 prior to fab. of CRs for NGNP	1700

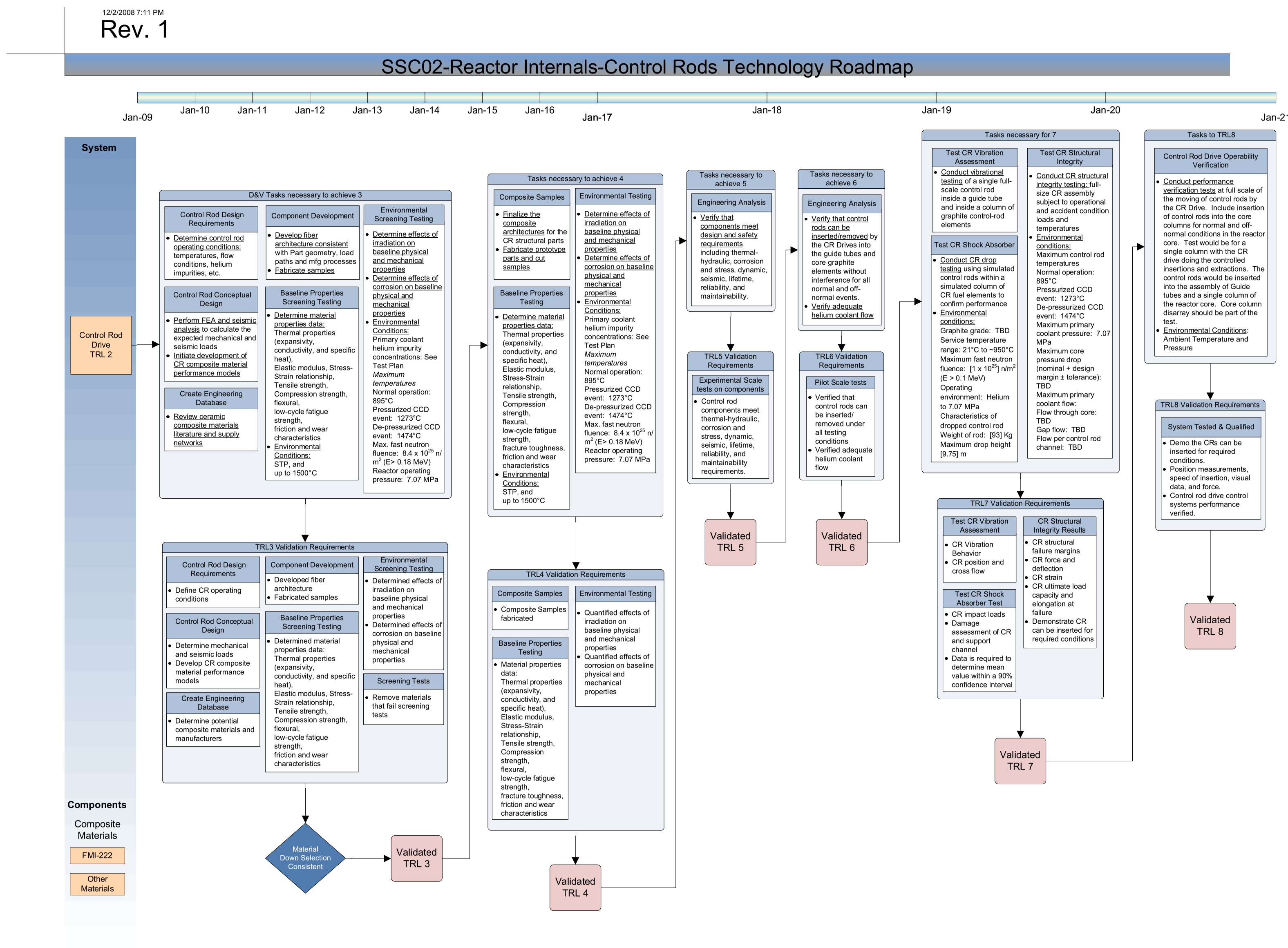
DDN(s) Supported: C.11.03.02, C.11.03.05, C11.03.06	**Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein, Russ Vollman

Date: 11-27-08 | **Originating Organization:** General Atomics

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Conduct CR shock absorber testing. A simulated CR will be dropped within a simulated column of CR fuel elements test various candidate shock absorber designs.	GA, Wyle Laboratories, Hazen Research, or other commercial laboratory	1 year, must be completed by first quarter of 2020 prior to fab. of CRs for NGNP	600
3. Conduct CR structural integrity testing. A full-size CR assembly will be subjected to operational and accident-condition loads and temperatures to quantify margins against structural failure. The tests will also determine ultimate load capacity and elongation at failure for these conditions.	GA, Wyle Laboratories, Hazen Research, or INL CTF	18 months, must be completed by first quarter of 2020 prior to fab. of CRs for NGNP	1900

<table>
<tr><th colspan="4">TRL Rating Sheet</th></tr>
<tr><td>Vendor: GA</td><td colspan="2">Document Number: SSC-2.1.6</td><td>Revision: 1</td></tr>
<tr><td colspan="4">☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology</td></tr>
<tr><td colspan="4">Title: Reactor Internals-Control Rods (CR)</td></tr>
<tr><td colspan="4">Description:
Control Rods (CR) are located in 2 areas of the reactor core: near the inner boundary of the fuel and central replaceable reflector elements (12 rods); and near the outer boundary between the fuel and outer replaceable reflector reflectors (18 rods). The outer CR are used to control the power in the core and are inserted during normal operation. The inner CR are withdrawn during normal operation and are only used to shut down the nuclear reaction. The rod is a linear assembly of rigid links filled with boronated graphite compacts within a cylindrical sleeve. The joints between rigid links allow flexibility in the rod assembly. The sleeves and joints are the structural elements that contain the nonstructural absorber compacts and transfer the operational loads to the control rod drive. All control rods are identical to accommodate interchangeability.</td></tr>
<tr><td>Area:</td><td colspan="3">☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP</td></tr>
<tr><td colspan="2">ASSCT:</td><td>Parent:</td><td>WBS:</td></tr>
<tr><th colspan="4">Technology Readiness Level</th></tr>
<tr><td></td><td>Next Lower Rating Level</td><td>Current Rating Level</td><td>Next Higher Rating Level</td></tr>
<tr><td>Generic Definitions (abbreviated)</td><td>Verified at pilot scale</td><td>Verified at engineering scale</td><td>Tested and Qualified</td></tr>
<tr><td>TRL</td><td>6</td><td>7</td><td>8</td></tr>
<tr><td colspan="4">Basis for Rating (Check box if continued on additional sheets) ☐</td></tr>
<tr><td colspan="4">TRL 7 is achieved when the following conditions are met: (1) CR flow-induced vibration testing has been completed and the results confirm that any potential damage to the CRs or graphite CR channels due to flow-induced vibrations will not inhibit impact the capability to insert or withdraw the CRs in the reactor; (2) CR shock absorber testing has been completed and the results have resulted in selection of a satisfactory shock absorber design; and (3) CR structural integrity testing has been completed and the results confirm that the CR design has adequate margin against operational failure.</td></tr>
<tr><td colspan="4">Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐</td></tr>
<tr><th>Actions (list all)</th><th>Actionee</th><th>Schedule</th><th>Cost ($K)</th></tr>
<tr><td>Perform testing at the NGNP to verify that the CRs can be inserted and withdrawn from the CR channels with the neutron control assembly (NCA) providing the motive force. This test will be performed as part of the integrated test of an NCA, CR guide tube, and CR as described in the GA Test Plan 911133.</td><td>GA and NGNP operator</td><td>Must be completed ~3 months prior to installation of NCAs and CRs in NGNP</td><td>200 (incremental cost for CR testing)</td></tr>
<tr><td colspan="2">DDN(s) Supported: None</td><td colspan="2">Technology Case File:</td></tr>
<tr><td colspan="4">Subject Matter Expert Making Determination: John Saurwein, Russ Vollman</td></tr>
<tr><td>Date: 11-27-08</td><td colspan="3">Originating Organization: General Atomics</td></tr>
</table>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Rev. 1
SSC02-Reactor Internals-Control Rods Technology Roadmap
Jan-09
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System
Control Rod Drive TRL 2
Components
Composite Materials
FMI-222
Other Materials
D&V Tasks necessary to achieve 3
Control Rod Design Requirements
• Determine control rod operating conditions: temperatures, flow conditions, helium impurities, etc.
Control Rod Conceptual Design
• Perform FEA and seismic analysis to calculate the expected mechanical and seismic loads
• Initiate development of CR composite material performance models
Create Engineering Database
• Review ceramic composite materials literature and supply networks
Component Development
• Develop fiber architecture consistent with Part geometry, load paths and mfg processes
• Fabricate samples
Baseline Properties Screening Testing
• Determine material properties data: Thermal properties (expansivity, conductivity, and specific heat), Elastic modulus, Stress-Strain relationship, Tensile strength, Compression strength, flexural, low-cycle fatigue strength, friction and wear characteristics
• Environmental Conditions: STP, and up to 1500°C
Environmental Screening Testing
• Determine effects of irradiation on baseline physical and mechanical properties
• Determine effects of corrosion on baseline physical and mechanical properties
• Environmental Conditions: Primary coolant helium impurity concentrations: See Test Plan Maximum temperatures Normal operation: 895°C Pressurized CCD event: 1273°C De-pressurized CCD event: 1474°C Max. fast neutron fluence: 8.4 x 10^25 n/m^2 (E> 0.18 MeV) Reactor operating pressure: 7.07 MPa
TRL3 Validation Requirements
Control Rod Design Requirements
• Define CR operating conditions
Control Rod Conceptual Design
• Determine mechanical and seismic loads
• Develop CR composite material performance models
Create Engineering Database
• Determine potential composite materials and manufacturers
Component Development
• Developed fiber architecture
• Fabricated samples
Baseline Properties Screening Testing
• Determined material properties data: Thermal properties (expansivity, conductivity, and specific heat), Elastic modulus, Stress-Strain relationship, Tensile strength, Compression strength, flexural, low-cycle fatigue strength, friction and wear characteristics
Environmental Screening Testing
• Determined effects of irradiation on baseline physical and mechanical properties
• Determined effects of corrosion on baseline physical and mechanical properties
Screening Tests
• Remove materials that fail screening tests
Material Down Selection Consistent
Validated TRL 3
Tasks necessary to achieve 4
Composite Samples
• Finalize the composite architectures for the CR structural parts
• Fabricate prototype parts and cut samples
Baseline Properties Testing
• Determine material properties data: Thermal properties (expansivity, conductivity, and specific heat), Elastic modulus, Stress-Strain relationship, Tensile strength, Compression strength, flexural, low-cycle fatigue strength, fracture toughness, friction and wear characteristics
• Environmental Conditions: STP, and up to 1500°C
Environmental Testing
• Determine effects of irradiation on baseline physical and mechanical properties
• Determine effects of corrosion on baseline physical and mechanical properties
• Environmental Conditions: Primary coolant helium impurity concentrations: See Test Plan Maximum temperatures Normal operation: 895°C Pressurized CCD event: 1273°C De-pressurized CCD event: 1474°C Max. fast neutron fluence: 8.4 x 10^25 n/m^2 (E> 0.18 MeV) Reactor operating pressure: 7.07 MPa
TRL4 Validation Requirements
Composite Samples
• Composite Samples fabricated
Baseline Properties Testing
• Material properties data: Thermal properties (expansivity, conductivity, and specific heat), Elastic modulus, Stress-Strain relationship, Tensile strength, Compression strength, flexural, low-cycle fatigue strength, fracture toughness, friction and wear characteristics
Environmental Testing
• Quantified effects of irradiation on baseline physical and mechanical properties
• Quantified effects of corrosion on baseline physical and mechanical properties
Validated TRL 4
Tasks necessary to achieve 5
Engineering Analysis
• Verify that components meet design and safety requirements including thermal-hydraulic, corrosion and stress, dynamic, seismic, lifetime, reliability, and maintainability.
TRL5 Validation Requirements
Experimental Scale tests on components
• Control rod components meet thermal-hydraulic, corrosion and stress, dynamic, seismic, lifetime, reliability, and maintainability requirements.
Validated TRL 5
Tasks necessary to achieve 6
Engineering Analysis
• Verify that control rods can be inserted/removed by the CR Drives into the guide tubes and core graphite elements without interference for all normal and off-normal events.
• Verify adequate helium coolant flow
TRL6 Validation Requirements
Pilot Scale tests
• Verified that control rods can be inserted/ removed under all testing conditions
• Verified adequate helium coolant flow
Validated TRL 6
Tasks necessary for 7
Test CR Vibration Assessment
• Conduct vibrational testing of a single full-scale control rod inside a guide tube and inside a column of graphite control-rod elements
Test CR Shock Absorber
• Conduct CR drop testing using simulated control rods within a simulated column of CR fuel elements to confirm performance
• Environmental conditions: Graphite grade: TBD Service temperature range: 21°C to ~950°C Maximum fast neutron fluence: [1 x 10^25] n/m^2 (E > 0.1 MeV) Operating environment: Helium to 7.07 MPa Characteristics of dropped control rod Weight of rod: [93] Kg Maximum drop height [9.75] m
Test CR Structural Integrity
• Conduct CR structural integrity testing: full-size CR assembly subject to operational and accident condition loads and temperatures
• Environmental conditions: Maximum control rod temperatures Normal operation: 895°C Pressurized CCD event: 1273°C De-pressurized CCD event: 1474°C Maximum primary coolant pressure: 7.07 MPa Maximum core pressure drop (nominal + design margin ± tolerance): TBD Maximum primary coolant flow: Flow through core: TBD Gap flow: TBD Flow per control rod channel: TBD
TRL7 Validation Requirements
Test CR Vibration Assessment
• CR Vibration Behavior
• CR position and cross flow
Test CR Shock Absorber Test
• CR impact loads
• Damage assessment of CR and support channel
• Data is required to determine mean value within a 90% confidence interval
CR Structural Integrity Results
• CR structural failure margins
• CR force and deflection
• CR strain
• CR ultimate load capacity and elongation at failure
• Demonstrate CR can be inserted for required conditions
Validated TRL 7
Tasks to TRL8
Control Rod Drive Operability Verification
• Conduct performance verification tests at full scale of the moving of control rods by the CR Drive. Include insertion of control rods into the core columns for normal and off-normal conditions in the reactor core. Test would be for a single column with the CR drive doing the controlled insertions and extractions. The control rods would be inserted into the assembly of Guide tubes and a single column of the reactor core. Core column disarray should be part of the test.
• Environmental Conditions: Ambient Temperature and Pressure
TRL8 Validation Requirements
System Tested & Qualified
• Demo the CRs can be inserted for required conditions.
• Position measurements, speed of insertion, visual data, and force.
• Control rod drive control systems performance verified.
Validated TRL 8

4.3 SSC-3 Hot Duct TRL

TRL Rating Sheets, TRL 2 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-3.1	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Hot Duct and Insulation Between Reactor and Steam Generator

Description:
The hot duct is an assembly of insulation and ducting through which the helium coolant exiting the reactor core at 950°C is transported to the steam generator. The ducting is located within the cross vessel and has a co-axial configuration with the cross vessel. The "cold" helium at 490°C exiting the steam generator is returned to the reactor vessel through the annular flow path between the hot duct and cross vessel. The nominal peak operating temperature of the hot duct is 950°C, but the hot duct could be exposed to somewhat higher temperatures due to hot streaking of the helium exiting the core. Considered insulation will be both internal and external to the duct, consistent with assumptions made in GA Report 911105/0.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:		**Parent:**		**WBS:**	3310

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Basic principles observed	Application formulated	Proof of principal
TRL	**1**	**2**	**3**

Basis for Rating (Check box if continued on additional sheets) ☒

An initial TRL of 2 has been assigned to the hot duct on the grounds that a proposed configuration for the insulated duct has been formulated and the technical challenges associated with containment of the flow of high temperature helium gas are understood. Additionally, published data indicate that there are commercially available insulating materials and duct alloys that are viable candidates for the application. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1) Establish safety class	GA	6 months	200
2) Establish code applicability (ASME pressure vessel, nuclear, piping, QA)	GA/URS-WD		

DDN(s) Supported: C.11.02.02, C.11.02.12, C.11.02.13, C.11.02.14, N.11.02.13, N.11.02.14	**Technology Case File:**

Subject Matter Expert Making Determination: Greg Walz

Date: 12-8-08	**Originating Organization:** Washington Division of URS

Additional Basis Sheet(s)
Basis (continued): However, critical functions and/or characteristics for a duct/insulation system have not been proven for the service conditions for NGNP. Analytical or experimental data testing the proposed configuration is not known to exist. Additionally, the critical functions of the duct/insulating system have not been finalized with regard to the following: • Safety Class and Code applicability – final determination for the safety classification for the hot leg helium duct is needed to address plant licensing. Is credit for leak before break needed to support plant licensing, e.g., to preclude a total cross vessel failure (hot and cold duct) from consideration? • Leak detection criteria and capability including that required to support credit for before break if needed, are critical characteristics that need to be defined for the specific configuration. • Inspectability of welds if necessary, and required weld examinations are critical characteristics that pose a challenge for the co-axial hot and cold leg configuration with internal and external insulation. • Inspectability of insulation: determine critical thickness of insulation, which is subject to erosion, and effect on overall system performance • Stress Analysis – detailed stress analysis of the specific configurations under consideration for NGNP have not been performed, including differential temperature expansion. • Accident Conditions – Design basis excursion pressures and temperatures to which the ductwork may be subjected have not been defined.

Additional Action Sheet(s)

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
3) Determine thermal and mechanical properties of duct material	GA/URS-WD		
4) Determine thermal and mechanical properties of insulation material	GA/URS-WD		
5) Duct material stress testing under design basis event conditions	GA/URS-WD		
6) Establish conditions of service	GA		
7) Material selection	GA/URS-WD		

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-3.2	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Hot Duct and Insulation Between Reactor and Steam Generator

Description:
The hot duct is an assembly of insulation and ducting through which the helium coolant exiting the reactor core at 950°C is transported to the steam generator. The ducting is located within the cross vessel and has a co-axial configuration with the cross vessel. The "cold" helium at 490°C exiting the steam generator is returned to the reactor vessel through the annular flow path between the hot duct and cross vessel. The nominal peak operating temperature of the hot duct is 950°C, but the hot duct could be exposed to somewhat higher temperatures due to hot streaking of the helium exiting the core. Considered insulation will be both internal and external to the duct, consistent with assumptions made in GA Report 911105/0.

Area:	☐ NHSS	☒ HTS	☐ HPS	☐ BCS	☐ BOP

ASSCT:	**Parent:**	**WBS:**	3310

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Application formulated	Proof of principal	Demonstrated at bench scale
TRL	**2**	**3**	**4**

Basis for Rating (Check box if continued on additional sheets) ☐

A TRL of 3 will be achieved on the basis of completion of the action items identified in the TRL rating sheet for TRL 2.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1) Erosion/corrosion accelerated wear testing	GA/URS-WD	1 year	$180
2) Environmental qualification of duct and insulation	GA/URS-WD		

DDN(s) Supported: C.11.02.02, C.11.02.12, C.11.02.13, C.11.02.14, N.11.02.13, N.11.02.14	**Technology Case File:**
Subject Matter Expert Making Determination: Greg Walz	
Date: 12-3-08	**Originating Organization:** Washington Division of URS

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
3) Upfront CFD Analysis	GA/URS-WD		
4) Upfront FEA Analysis	GA/URS-WD		
5) Hot to cold leg leak detection	GA/URS-WD		
6) Insulation connection method	GA/URS-WD		

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-3.3	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Hot Duct and Insulation Between Reactor and Steam Generator

Description:
The hot duct is an assembly of insulation and ducting through which the helium coolant exiting the reactor core at 950°C is transported to the steam generator. The ducting is located within the cross vessel and has a co-axial configuration with the cross vessel. The "cold" helium at 490°C exiting the steam generator is returned to the reactor vessel through the annular flow path between the hot duct and cross vessel. The nominal peak operating temperature of the hot duct is 950°C, but the hot duct could be exposed to somewhat higher temperatures due to hot streaking of the helium exiting the core. Considered insulation will be both internal and external to the duct, consistent with assumptions made in GA Report 911105/0.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:		**Parent:**		**WBS:**	3310

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Proof of principal	Demonstrated at bench scale	Demonstrated at experimental scale
TRL	**3**	**4**	**5**

Basis for Rating (Check box if continued on additional sheets) ☐

A TRL of 4 will be achieved on the basis of completion of the action items identified in the TRL rating sheet for TRL 3.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1) Hot duct and insulation materials property tests including: Room temperature and high temperature material properties verification, Environmental qualifications, Irradiation (specification and interpretation), Weldability and weld strength, Stress corrosion cracking	GA/URS-WD	1 year	610 – 810 Excluding INL and HFEF Costs

DDN(s) Supported: C.11.02.02, C.11.02.12, C.11.02.13, C.11.02.14, N.11.02.13, N.11.02.14	**Technology Case File:**

Subject Matter Expert Making Determination: Greg Walz

Date: 12-3-08	**Originating Organization:** Washington Division of URS

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2) Component level test	GA/URS-WD		
3) Life cycle cost analysis	GA/URS-WD		
4) RAMI analysis	GA/URS-WD		
5) Acoustic and Flow vibrations test	WD & SME		
6) Creep analysis	WD & SME		
7) Endurance limit analysis	Becht NS		
8) ALARA analysis	GA/URS-WD		
9) LIMIT analysis	Becht NS		

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-3.4	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Hot Duct and Insulation Between Reactor and Steam Generator

Description:
The hot duct is an assembly of insulation and ducting through which the helium coolant exiting the reactor core at 950°C is transported to the steam generator. The ducting is located within the cross vessel and has a co-axial configuration with the cross vessel. The “cold” helium at 490°C exiting the steam generator is returned to the reactor vessel through the annular flow path between the hot duct and cross vessel. The nominal peak operating temperature of the hot duct is 950°C, but the hot duct could be exposed to somewhat higher temperatures due to hot streaking of the helium exiting the core. Considered insulation will be both internal and external to the duct, consistent with assumptions made in GA Report 911105/0.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:** 3310

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Demonstrated at bench scale	Demonstrated at experimental scale	Demonstrated at pilot scale
TRL	**4**	**5**	**6**

Basis for Rating (Check box if continued on additional sheets) ☐

A TRL of 5 will be achieved on the basis of completion of the action items identified in the TRL rating sheet for TRL 4.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1) FEA analysis - Stress analysis to optimize physical configuration	GA/URS-WD	1 year	367 - 417
2) CFD analysis to optimize physical configuration - including insulation performance and flow conditions in the hot and cold duct sections (Cont.)	GA/URS-WD		

DDN(s) Supported: C.11.02.02, C.11.02.12, C.11.02.13, C.11.02.14, N.11.02.13, N.11.02.14	**Technology Case File:**

Subject Matter Expert Making Determination: Greg Walz

Date: 12-3-08	**Originating Organization:** Washington Division of URS

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
3) Sub-system level test 4) Final thermal expansion analysis	URS-WD URS-WD		

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-3.5	**Revision:** 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Hot Duct and Insulation Between Reactor and Steam Generator

Description:
The hot duct is an assembly of insulation and ducting through which the helium coolant exiting the reactor core at 950°C is transported to the steam generator. The ducting is located within the cross vessel and has a co-axial configuration with the cross vessel. The "cold" helium at 490°C exiting the steam generator is returned to the reactor vessel through the annular flow path between the hot duct and cross vessel. The nominal peak operating temperature of the hot duct is 950°C, but the hot duct could be exposed to somewhat higher temperatures due to hot streaking of the helium exiting the core. Considered insulation will be both internal and external to the duct, consistent with assumptions made in GA Report 911105/0.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:** 3310

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Demonstrated at experimental scale	Demonstrated at pilot scale	Demonstrated at engineering scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☐

A TRL of 6 will be achieved on the basis of completion of the action items identified in the TRL rating sheet for TRL 5.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1) Testing of integrated system using 1/10 scale model	GA/URS-WD	1 year	545 - 795
a) Measure parameters needed to validate models and observe scale model performance	GA/URS-WD		
b) Terminal end (nozzle) attachment method/allowable nozzle loads	GA/URS-WD		

DDN(s) Supported: C.11.02.02, C.11.02.12, C.11.02.13, C.11.02.14, N.11.02.13, N.11.02.14	**Technology Case File:**

Subject Matter Expert Making Determination: Greg Walz

Date: 12-3-08	**Originating Organization:** Washington Division of URS

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost**
c) Evaluate methods to conduct in-service inspections	URS-WD		
d) Establish and validate installation techniques	URS-WD		
2) Conduct risk assessment to determine extent of CTF testing requirements	URS-WD		

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-3.6	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Hot Duct and Insulation Between Reactor and Steam Generator

Description:
The hot duct is an assembly of insulation and ducting through which the helium coolant exiting the reactor core at 950°C is transported to the steam generator. The ducting is located within the cross vessel and has a co-axial configuration with the cross vessel. The “cold” helium at 490°C exiting the steam generator is returned to the reactor vessel through the annular flow path between the hot duct and cross vessel. The nominal peak operating temperature of the hot duct is 950°C, but the hot duct could be exposed to somewhat higher temperatures due to hot streaking of the helium exiting the core. Considered insulation will be both internal and external to the duct, consistent with assumptions made in GA Report 911105/0.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**	3310

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Demonstrated at pilot scale	Demonstrated at engineering scale	Tested and qualified
TRL	**6**	**7**	**8**

Basis for Rating (Check box if continued on additional sheets) ☐

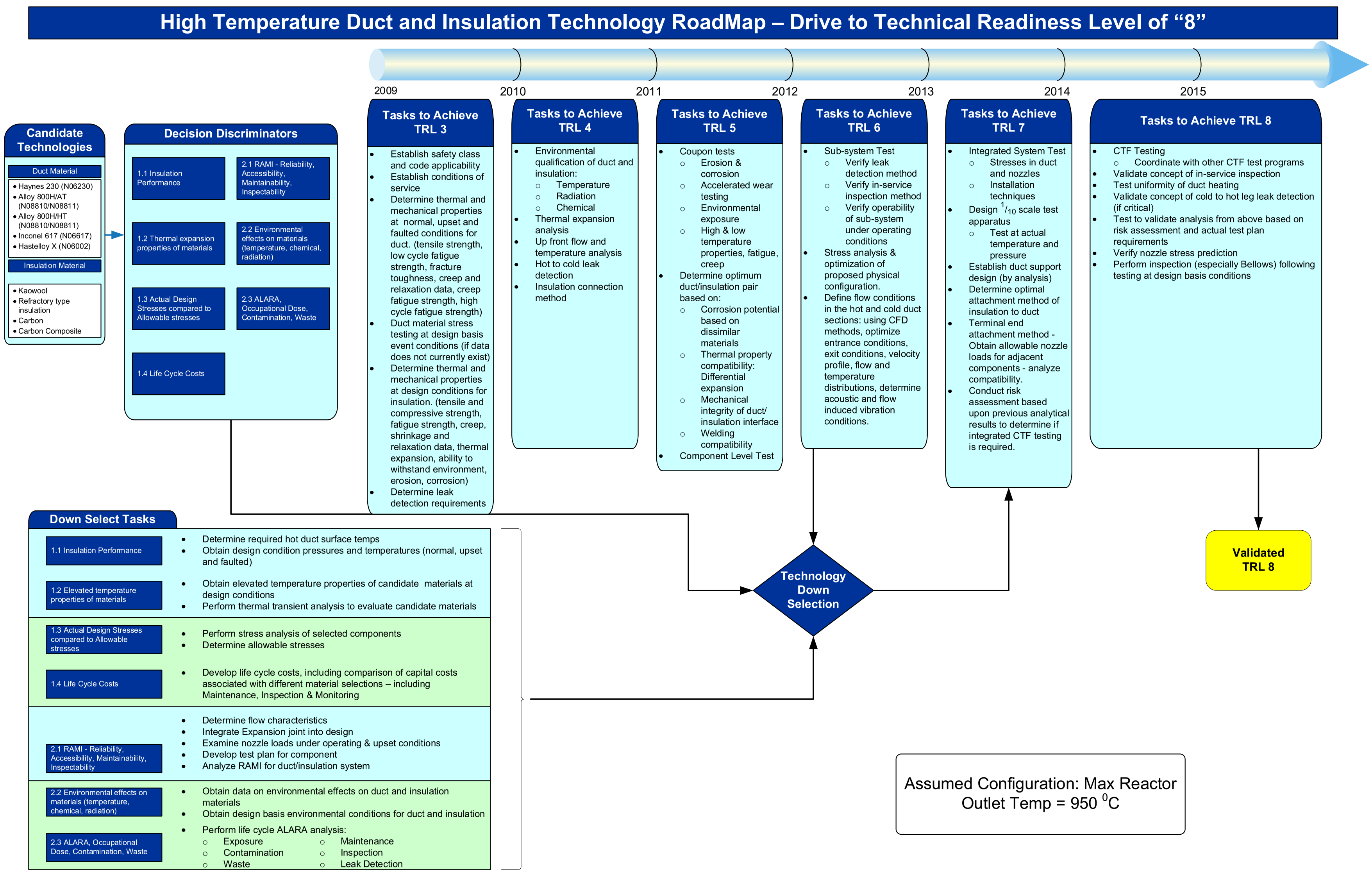
A TRL of 7 will be achieved on the basis of completion of the action items identified in the TRL rating sheet for TRL 6.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1) Integrated CTF testing (as a part of a larger test effort 2) Validate concept of in-service inspection 3) Test uniformity of duct heating	GA/URS-WD GA/URS-WD GA/URS-WD	2 Years (coordinate with others)	500 (INL/BEA scope not included)

DDN(s) Supported: C.11.02.02, C.11.02.12, C.11.02.13, C.11.02.14, N.11.02.13, N.11.02.14	**Technology Case File:**
Subject Matter Expert Making Determination: Greg Walz	
Date: 12-3-08	**Originating Organization:** Washington Division of URS

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost**
4) Validate concept of cold to hot leg leak detection 5) Validate analytical testing based on risk assessment - FEA simulations validation - CFD simulations validation	GA/URS-WD GA/URS-WD		


High Temperature Duct and Insulation Technology RoadMap – Drive to Technical Readiness Level of "8"
2009
2010
2011
2012
2013
2014
2015
Candidate Technologies
Duct Material
• Haynes 230 (N06230)
• Alloy 800H/AT (N08810/N08811)
• Alloy 800H/HT (N08810/N08811)
• Inconel 617 (N06617)
• Hastelloy X (N06002)
Insulation Material
• Kaowool
• Refractory type insulation
• Carbon
• Carbon Composite
Decision Discriminators
1.1 Insulation Performance
1.2 Thermal expansion properties of materials
1.3 Actual Design Stresses compared to Allowable stresses
1.4 Life Cycle Costs
2.1 RAMI - Reliability, Accessibility, Maintainability, Inspectability
2.2 Environmental effects on materials (temperature, chemical, radiation)
2.3 ALARA, Occupational Dose, Contamination, Waste
Tasks to Achieve TRL 3
• Establish safety class and code applicability
• Establish conditions of service
• Determine thermal and mechanical properties at normal, upset and faulted conditions for duct. (tensile strength, low cycle fatigue strength, fracture toughness, creep and relaxation data, creep fatigue strength, high cycle fatigue strength)
• Duct material stress testing at design basis event conditions (if data does not currently exist)
• Determine thermal and mechanical properties at design conditions for insulation. (tensile and compressive strength, fatigue strength, creep, shrinkage and relaxation data, thermal expansion, ability to withstand environment, erosion, corrosion)
• Determine leak detection requirements
Tasks to Achieve TRL 4
• Environmental qualification of duct and insulation:
○ Temperature
○ Radiation
○ Chemical
• Thermal expansion analysis
• Up front flow and temperature analysis
• Hot to cold leak detection
• Insulation connection method
Tasks to Achieve TRL 5
• Coupon tests
○ Erosion & corrosion
○ Accelerated wear testing
○ Environmental exposure
○ High & low temperature properties, fatigue, creep
• Determine optimum duct/insulation pair based on:
○ Corrosion potential based on dissimilar materials
○ Thermal property compatibility: Differential expansion
○ Mechanical integrity of duct/ insulation interface
○ Welding compatibility
• Component Level Test
Tasks to Achieve TRL 6
• Sub-system Test
○ Verify leak detection method
○ Verify in-service inspection method
○ Verify operability of sub-system under operating conditions
• Stress analysis & optimization of proposed physical configuration.
• Define flow conditions in the hot and cold duct sections: using CFD methods, optimize entrance conditions, exit conditions, velocity profile, flow and temperature distributions, determine acoustic and flow induced vibration conditions.
Tasks to Achieve TRL 7
• Integrated System Test
○ Stresses in duct and nozzles
○ Installation techniques
• Design $^{1}/_{10}$ scale test apparatus
○ Test at actual temperature and pressure
• Establish duct support design (by analysis)
• Determine optimal attachment method of insulation to duct
• Terminal end attachment method - Obtain allowable nozzle loads for adjacent components - analyze compatibility.
• Conduct risk assessment based upon previous analytical results to determine if integrated CTF testing is required.
Tasks to Achieve TRL 8
• CTF Testing
○ Coordinate with other CTF test programs
• Validate concept of in-service inspection
• Test uniformity of duct heating
• Validate concept of cold to hot leg leak detection (if critical)
• Test to validate analysis from above based on risk assessment and actual test plan requirements
• Verify nozzle stress prediction
• Perform inspection (especially Bellows) following testing at design basis conditions
Validated TRL 8
Technology Down Selection
Down Select Tasks
1.1 Insulation Performance
• Determine required hot duct surface temps
• Obtain design condition pressures and temperatures (normal, upset and faulted)
1.2 Elevated temperature properties of materials
• Obtain elevated temperature properties of candidate materials at design conditions
• Perform thermal transient analysis to evaluate candidate materials
1.3 Actual Design Stresses compared to Allowable stresses
• Perform stress analysis of selected components
• Determine allowable stresses
1.4 Life Cycle Costs
• Develop life cycle costs, including comparison of capital costs associated with different material selections – including Maintenance, Inspection & Monitoring
2.1 RAMI - Reliability, Accessibility, Maintainability, Inspectability
• Determine flow characteristics
• Integrate Expansion joint into design
• Examine nozzle loads under operating & upset conditions
• Develop test plan for component
• Analyze RAMI for duct/insulation system
2.2 Environmental effects on materials (temperature, chemical, radiation)
• Obtain data on environmental effects on duct and insulation materials
• Obtain design basis environmental conditions for duct and insulation
2.3 ALARA, Occupational Dose, Contamination, Waste
• Perform life cycle ALARA analysis:
○ Exposure
○ Contamination
○ Waste
○ Maintenance
○ Inspection
○ Leak Detection
Assumed Configuration: Max Reactor Outlet Temp = 950 ^{0}C

4.4 SSC-4a Reactor Core Assembly and SSC-4b Reactor Graphite Elements

Reactor Core Assembly, TRL Rating Sheets, TRL 5 through 7

Reactor Graphite Elements, TRL Rating Sheets, TRL 6 and 7

Technology Development Road Maps

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-4a.1	**Revision:** 1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Reactor Core

Description:
The primary functions of the Reactor Core are to generate high temperature heat using nuclear fission, transfer the heat to the helium coolant, and control radiation from the core. The Reactor Core consists of fuel elements, inner and outer reflector elements, upper reflector elements, and lower reflector elements (including flow distribution elements). All of these elements are hexagonal-shaped blocks manufactured from nuclear grade graphite. In terms of SSC categorization, the permanent side reflector is assumed to be part of Reactor Internals. (Cont. on additional description sheet)

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Components verified at bench scale	Components verified at experimental scale	Subsystem verified at pilot scale
TRL	**4**	**5**	**6**

Basis for Rating (Check box if continued on additional sheets) ☒

The testing programs to support design of the Ft. St. Vrain (FSV) reactor and the operational data base from FSV justify a high TRL level for this system. However, FSV used grades H-327 and H-451 graphite that are no longer available and the NGNP prismatic core design will likely adopt one of the new grades of graphite that are under development (e.g., PCEA, NBG-17 or NBG-18), as described in the NGNP Graphite Technology Development Plan prepared by INL. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Perform thermal/flow testing of individual fuel and reflector elements.	DOE Labs	1 year after start of final design	3,000
2. Perform detailed CFD modeling of individual fuel and reflector elements.	Vendor	2 years after start of final design	400

DDN(s) Supported: C.11.03.03, C.11.03.04, C.11.03.41, C.11.03.42, C.11.03.43, C.11.03.44	**Technology Case File:**

Subject Matter Expert Making Determination: Matt Richards

Date: 12-8-08	**Originating Organization:** General Atomics

Additional Description Sheet(s)
Description: The principal fuel elements are in the form of a right hexagonal prism, 793 mm high and 360 mm across the flats. The two other types of fuel elements are those with control-rod channels and those with reserve-shutdown channels. The active core (fueled region) consists of 102 fuel columns with 10 blocks per column, comprising a 3-row annular region. The active core is surrounded by prismatic blocks that form the upper, lower, inner, and side reflectors. Some of the columns in the outer reflector and active core (and possibly the inner reflector, depending on the final core design) contain channels for controls rods. Some of the columns in the active core also contain channels for reserve shutdown material.

Additional Basis Sheet(s)
Basis: In addition, the NGNP core configuration is significantly different from FSV (annular core for NGNP vs. cylindrical core for FSV and 10-block high core for NGNP vs. 6-block high core for FSV). For these reasons, the starting TRL level is judged to be 5 for this system. A TRL level of 6 is achieved after test programs to satisfy the following DDNS are successfully completed: C.11.03.03 (Core Element Dynamic Strength Data), C.11.03.04 (Core Element Failure Mode Data), C.11.03.41 (Fuel Element Channel Flow Data), C.11.03.42 (Control Rod Flow Channel Data).

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
3. Perform mechanical testing of individual fuel and reflector elements.	DOE Labs	1 year after start of final design	7,000
4. Perform detailed finite-element stress analyses of individual fuel and reflector elements.	Vendor	2 years after start of final design	400

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-4a.2		**Revision:** 1	
☐ Area	☒ System	☐ Subsystem/Structure	☐ Component	☐ Technology

Title: Reactor Core

Description:
The primary functions of the Reactor Core are to generate high temperature heat using nuclear fission, transfer the heat to the helium coolant, and control radiation from the core. The Reactor Core consists of fuel elements, inner and outer reflector elements, upper reflector elements, and lower reflector elements (including flow distribution elements). All of these elements are hexagonal-shaped blocks manufactured from nuclear grade graphite. In terms of SSC categorization, the permanent side reflector is assumed to be part of Reactor Internals. (Cont. on additional description sheet)

Area:	☒ NHSS	☐ HTS	☐ HPS	☐ BCS	☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Components verified at experimental scale	Subsystem verified at pilot scale	System verified at engineering scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☒

TRL 6 is achieved for this system after all test data have been obtained and detailed modeling has been performed for the individual fuel and reflector elements to satisfy DDNs C.11.03.03, C.11.03.04, C.11.03.41, C.11.03.42, C.11.03.43, C.11.03.44. To advance to TRL 7, testing programs must be completed to satisfy the following DDNs: C.11.03.01 (Core Column Vibration Data), C.11.03.45 (Core Crossflow Test Data), C.11.03.46 (Core Fluctuation Test Data), C.11.03.43 (Bottom Reflector/Core Support Pressure Drop and Flow Mixing Data), C.11.03.44 (Metallic Plenum Element and Top Reflector Pressure Drop and Flow Distribution).

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Perform multiple-block testing and to obtain core column vibration data.	DOE Labs	2 years after start of final design	4,000
2. Perform detailed modeling of core vibrations.	Vendor	3 years after start of final design	400

DDN(s) Supported: C.11.03.01, C.11.03.45, C.11.03.46	**Technology Case File:**
Subject Matter Expert Making Determination: Matt Richards	
Date: 12-8-08	**Originating Organization:** General Atomics

Additional Description Sheet(s)

Description:

The principal fuel elements are in the form of a right hexagonal prism, 793 mm high and 360 mm across the flats. The two other types of fuel elements are those with control-rod channels and those with reserve-shutdown channels. The active core (fueled region) consists of 102 fuel columns with 10 blocks per column, comprising a 3-row annular region. The active core is surrounded by prismatic blocks that form the upper, lower, inner, and side reflectors. Some of the columns in the outer reflector and active core (and possibly the inner reflector, depending on the final core design) contain channels for controls rods. Some of the columns in the active core also contain channels for reserve shutdown material.

Additional Action Sheet(s)

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
3. Perform mock-up testing to obtain data for horizontal cross flow.	DOE Labs	2 years after start of final design	3,000
4. Perform detailed CFD analyses of cross flow.	Vendor	3 years after start of final design	200
5. Perform mock-up testing to obtain data for core flow fluctuations and hot streaks.	DOE Labs	2 years after start of final design	5,000
6. Perform CFD analyses of core flow fluctuations and hot streaks.	Vendor	3 years after start of final design	400

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-4a.3	**Revision:** 1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Reactor Core

Description:
The Reactor Core consists of fuel elements, inner and outer reflector elements, upper reflector elements, and lower reflector elements (including flow distribution elements). All of these elements are hexagonal-shaped blocks manufactured from nuclear grade graphite. In terms of SSC categorization, the permanent side reflector is assumed to be part of Reactor Internals. The principal fuel elements are in the form of a right hexagonal prism, 793 mm high and 360 mm across the flats. The two other types of fuel elements are those with control-rod channels and those with reserve-shutdown channels. The active core (fueled region) consists of 102 fuel columns with 10 blocks per column, comprising a 3-row annular region. The active core is surrounded by prismatic blocks that form the upper, lower, inner, and side reflectors. Some of the columns in the outer reflector and active core (and possibly the inner reflector, depending on the final core design) contain channels for controls rods. Some of the columns in the active core also contain channels for reserve shutdown material.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Verified at pilot scale	Verified at engineering scale	System tested and qualified
TRL	**6**	**7**	**8**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 7 is achieved for this system after all integral test data have been obtained and detailed modeling has been performed to satisfy DDNs C.11.03.01, C.11.03.45, and C.11.03.46.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Perform instrumented tests as part of NGNP startup testing to confirm flow distributions, temperature distributions, and mechanical loadings are within design specifications.	GA/NGNP operator	TBD	TBD

DDN(s) Supported: C11.03.01, C11.03.45, C.11.03.46	**Technology Case File:**

Subject Matter Expert Making Determination: Matt Richards

Date: 12-8-08	**Originating Organization:** General Atomics

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-4b.1	**Revision:** 0	

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Graphite

Description:
The graphite components of the reactor system are the core (fuel elements and replaceable reflector elements), the permanent side reflector, and the core support structure.

Area:	☐ NHSS	☐ HTS	☐ HPS	☐ BCS	☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Component verified at experimental scale	Component verified at pilot scale	Component verified at engineering scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☒

Nuclear-grade graphite is a mature technology and has been used previously in several HTGRS, including Ft. St. Vrain, which used block graphite elements manufactured from H-451 graphite. However, H-451 graphite is no longer commercially available and a graphite to replace H-451 is needed for a block-type NGNP core. The 30 MWt HTTR reactor in Oarai, Japan uses block graphite fuel and reflector elements manufactured from IG-110 graphite. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Perform test programs to obtain the requisite design data to advance to TRL 7	DOE Labs	3 years before completion of final design.	84,000
2. Obtain necessary ASME/ASTM code approvals.	Vendor / DOE Labs	2 years before completion of final design	2,000
3. Perform detailed modeling of in-core and accident-condition performance of graphite elements to establish design margins.	Vendor	1 year before completion of final design	1,500

DDN(s) Supported: C.11.03.11 through C.11.03.21 and C.11.03.23.	**Technology Case File:**
Subject Matter Expert Making Determination: Matt Richards	
Date: 10/30/08	**Originating Organization:** General Atomics

Additional Basis Sheet(s)

Basis:

For NGNP, new nuclear-grade graphites are being developed and qualified, including grades PCEA, NBG-17 and NBG-18 that can be used for a block-type core. Because of the extensive experience base with the manufacture and irradiation of nuclear-grade graphite, a TRL level of 6 is judged to be appropriate for this component.

A TRL level of 7 is achieved after the requisite design data have been obtained for the new graphite. These data are specified in the following Design Data Needs (DDNs): C.11.03.11 (Graphite Multiaxial Strength Data), C.11.03.12 (Graphite Fatigue Data), C.11.03.13 (Graphite Mechanical Properties Data), C.11.03.14 (Graphite Irradiation Induced Dimensional Change Data), C.11.03.15 (Graphite Irradiation Induced Creep Data), C.11.03.16 (Graphite Thermal Properties Data), C.11.03.17 (Graphite Fracture Mechanics Data), C.11.03.18 (Graphite Corrosion Data), C.11.03.19 (Graphite Corrosion Data for Methods Validation), C.11.03.20 (Graphite Destructive and Nondestructive Examination Data), C.11.03.21 (Graphite Coke Source Qualification), and C.11.03.23 (Graphite Oxidation Data for Postulated Accidents). These DDNs will be satisfied by completion of the graphite technology development plan described in INL document PLN-2497, Rev. 0.

Note: Another possible strategy is to startup the NGNP without having obtained the complete data base as defined by the above DDNs and use data obtained during the startup phase (either from NGNP operation or ongoing testing at DOE laboratories) to satisfy some elements of these DDNs.

<table>
<tr><td colspan="8" align="center">TRL Rating Sheet</td></tr>
<tr><td colspan="2">Vendor: GA</td><td colspan="3">Document Number: SSC-4b.2</td><td colspan="3">Revision: 0</td></tr>
<tr><td colspan="8">☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology</td></tr>
<tr><td colspan="8">Title: Graphite</td></tr>
<tr><td colspan="8">Description:
The graphite components of the reactor system are the core (fuel elements and replaceable reflector elements), the permanent side reflector, and the core support structure.</td></tr>
<tr><td>Area:</td><td colspan="7">☐ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP</td></tr>
<tr><td colspan="3">ASSCT:</td><td colspan="3">Parent:</td><td colspan="2">WBS:</td></tr>
<tr><td colspan="8" align="center">Technology Readiness Level</td></tr>
<tr><td colspan="2"></td><td colspan="2">Next Lower Rating Level</td><td colspan="2">Current Rating Level</td><td colspan="2">Next Higher Rating Level</td></tr>
<tr><td colspan="2">Generic Definitions (abbreviated)</td><td colspan="2">Component verified at pilot scale</td><td colspan="2">Component verified at engineering scale</td><td colspan="2">System tested and qualified</td></tr>
<tr><td colspan="2">TRL</td><td colspan="2">6</td><td colspan="2">7</td><td colspan="2">8</td></tr>
<tr><td colspan="8">Basis for Rating (Check box if continued on additional sheets) ☐</td></tr>
<tr><td colspan="8">TRL 7 is achieved for this system after design data have been obtained for a replacement to H-451 graphite. The required design data are described in the following Design Data Needs (DDNs): C.11.03.11 through C.11.03.21 and C.11.02.23. These DDNs will be satisfied by completion of the graphite technology development plan described in INL document PLN-2497, Rev. 0.</td></tr>
<tr><td colspan="8">Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐</td></tr>
<tr><td colspan="4">Actions (list all)</td><td>Actionee</td><td colspan="2">Schedule</td><td>Cost ($K)</td></tr>
<tr><td colspan="4">1. Perform instrumented tests as part of NGNP startup testing to confirm physical, mechanical, thermal, and chemical properties. Perform inspections of selected fuel and reflector elements at the end of startup testing.</td><td>Vendor/ Operator</td><td colspan="2">NGNP Startup Phase</td><td>15,000</td></tr>
<tr><td colspan="4">DDN(s) Supported: None</td><td colspan="4">Technology Case File:</td></tr>
<tr><td colspan="8">Subject Matter Expert Making Determination: Matt Richards</td></tr>
<tr><td colspan="3">Date: 10/30/08</td><td colspan="5">Originating Organization: General Atomics</td></tr>
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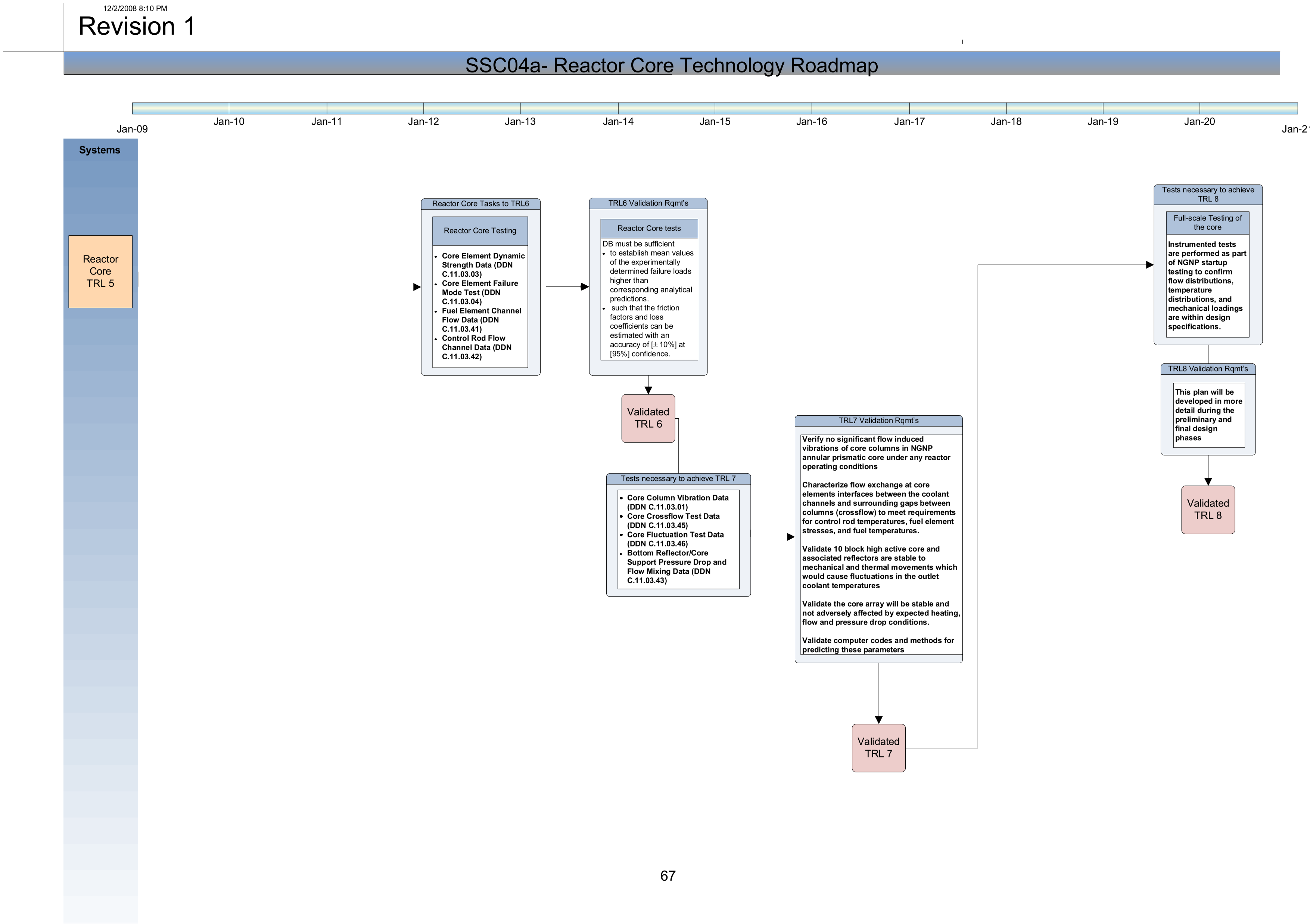
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Revision 1
SSC04a- Reactor Core Technology Roadmap
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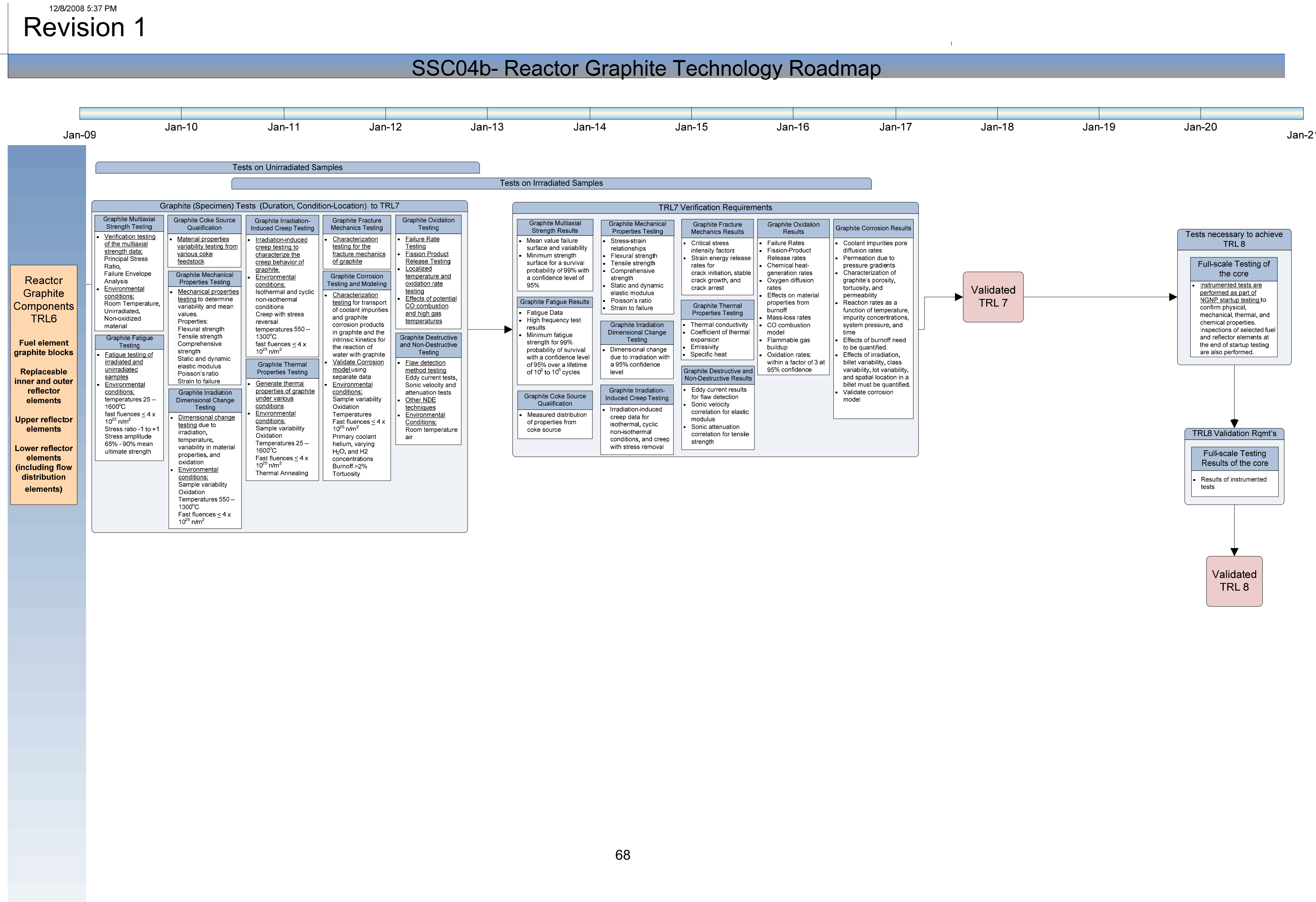
Systems
Reactor Core TRL 5
Reactor Core Tasks to TRL6
Reactor Core Testing
• Core Element Dynamic Strength Data (DDN C.11.03.03)
• Core Element Failure Mode Test (DDN C.11.03.04)
• Fuel Element Channel Flow Data (DDN C.11.03.41)
• Control Rod Flow Channel Data (DDN C.11.03.42)
TRL6 Validation Rqmt's
Reactor Core tests
DB must be sufficient
• to establish mean values of the experimentally determined failure loads higher than corresponding analytical predictions.
• such that the friction factors and loss coefficients can be estimated with an accuracy of [± 10%] at [95%] confidence.
Validated TRL 6
Tests necessary to achieve TRL 7
• Core Column Vibration Data (DDN C.11.03.01)
• Core Crossflow Test Data (DDN C.11.03.45)
• Core Fluctuation Test Data (DDN C.11.03.46)
• Bottom Reflector/Core Support Pressure Drop and Flow Mixing Data (DDN C.11.03.43)
TRL7 Validation Rqmt's
Verify no significant flow induced vibrations of core columns in NGNP annular prismatic core under any reactor operating conditions
Characterize flow exchange at core elements interfaces between the coolant channels and surrounding gaps between columns (crossflow) to meet requirements for control rod temperatures, fuel element stresses, and fuel temperatures.
Validate 10 block high active core and associated reflectors are stable to mechanical and thermal movements which would cause fluctuations in the outlet coolant temperatures
Validate the core array will be stable and not adversely affected by expected heating, flow and pressure drop conditions.
Validate computer codes and methods for predicting these parameters
Validated TRL 7
Tests necessary to achieve TRL 8
Full-scale Testing of the core
Instrumented tests are performed as part of NGNP startup testing to confirm flow distributions, temperature distributions, and mechanical loadings are within design specifications.
TRL8 Validation Rqmt's
This plan will be developed in more detail during the preliminary and final design phases
Validated TRL 8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SSC04b- Reactor Graphite Technology Roadmap
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Reactor Graphite Components TRL6
Fuel element graphite blocks
Replaceable inner and outer reflector elements
Upper reflector elements
Lower reflector elements (including flow distribution elements)
Tests on Unirradiated Samples
Tests on Irrradiated Samples
Graphite (Specimen) Tests (Duration, Condition-Location) to TRL7
Graphite Multiaxial Strength Testing
Verification testing of the multiaxial strength data: Principal Stress Ratio, Failure Envelope Analysis
Environmental conditions: Room Temperature, Unirradiated, Non-oxidized material
Graphite Fatigue Testing
Fatigue testing of irradiated and unirradiated samples
Environmental conditions: temperatures 25 – 1600°C fast fluences ≤ 4 x 10^25 n/m^2 Stress ratio -1 to +1 Stress amplitude 65% - 90% mean ultimate strength
Graphite Coke Source Qualification
Material properties variability testing from various coke feedstock
Graphite Mechanical Properties Testing
Mechanical properties testing to determine variability and mean values. Properties: Flexural strength Tensile strength Comprehensive strength Static and dynamic elastic modulus Poisson's ratio Strain to failure
Graphite Irradiation Dimensional Change Testing
Dimensional change testing due to irradiation, temperature, variability in material properties, and oxidation
Environmental conditions: Sample variability Oxidation Temperatures 550 – 1300°C Fast fluences ≤ 4 x 10^25 n/m^2
Graphite Irradiation-Induced Creep Testing
Irradiation-induced creep testing to characterize the creep behavior of graphite.
Environmental conditions: Isothermal and cyclic non-isothermal conditions Creep with stress reversal temperatures 550 – 1300°C fast fluences ≤ 4 x 10^25 n/m^2
Graphite Thermal Properties Testing
Generate thermal properties of graphite under various conditions
Environmental conditions: Sample variability Oxidation Temperatures 25 – 1600°C Fast fluences ≤ 4 x 10^25 n/m^2 Thermal Annealing
Graphite Fracture Mechanics Testing
Characterization testing for the fracture mechanics of graphite
Graphite Corrosion Testing and Modeling
Characterization testing for transport of coolant impurities and graphite corrosion products in graphite and the intrinsic kinetics for the reaction of water with graphite
Validate Corrosion model using separate data
Environmental conditions: Sample variability Oxidation Temperatures Fast fluences ≤ 4 x 10^25 n/m^2 Primary coolant helium, varying H2O, and H2 concentrations Burnoff >2% Tortuosity
Graphite Oxidation Testing
Failure Rate Testing
Fission Product Release Testing
Localized temperature and oxidation rate testing
Effects of potential CO combustion and high gas temperatures
Graphite Destructive and Non-Destructive Testing
Flaw detection method testing Eddy current tests, Sonic velocity and attenuation tests
Other NDE techniques
Environmental Conditions: Room temperature air
TRL7 Verification Requirements
Graphite Multiaxial Strength Results
Mean value failure surface and variability
Minimum strength surface for a survival probability of 99% with a confidence level of 95%
Graphite Fatigue Results
Fatigue Data
High frequency test results
Minimum fatigue strength for 99% probability of survival with a confidence level of 95% over a lifetime of 10^2 to 10^5 cycles
Graphite Coke Source Qualification
Measured distribution of properties from coke source
Graphite Mechanical Properties Testing
Stress-strain relationships
Flexural strength
Tensile strength
Comprehensive strength
Static and dynamic elastic modulus
Poisson's ratio
Strain to failure
Graphite Irradiation Dimensional Change Testing
Dimensional change due to irradiation with a 95% confidence level
Graphite Irradiation-Induced Creep Testing
Irradiation-induced creep data for isothermal, cyclic non-isothermal conditions, and creep with stress removal
Graphite Fracture Mechanics Results
Critical stress intensity factors
Strain energy release rates for crack initiation, stable crack growth, and crack arrest
Graphite Thermal Properties Testing
Thermal conductivity
Coefficient of thermal expansion
Emissivity
Specific heat
Graphite Destructive and Non-Destructive Results
Eddy current results for flaw detection
Sonic velocity correlation for elastic modulus
Sonic attenuation correlation for tensile strength
Graphite Oxidation Results
Failure Rates
Fission-Product Release rates
Chemical heat-generation rates
Oxygen diffusion rates
Effects on material properties from burnoff
Mass-loss rates
CO combustion model
Flammable gas buildup
Oxidation rates: within a factor of 3 at 95% confidence
Graphite Corrosion Results
Coolant impurities pore diffusion rates
Permeation due to pressure gradients
Characterization of graphite's porosity, tortuosity, and permeability
Reaction rates as a function of temperature, impurity concentrations, system pressure, and time
Effects of burnoff need to be quantified.
Effects of irradiation, billet variability, class variability, lot variability, and spatial location in a billet must be quantified.
Validate corrosion model
Validated TRL 7
Tests necessary to achieve TRL 8
Full-scale Testing of the core
Instrumented tests are performed as part of NGNP startup testing to confirm physical, mechanical, thermal, and chemical properties. Inspections of selected fuel and reflector elements at the end of startup testing are also performed.
TRL8 Validation Rqmt's
Full-scale Testing Results of the core
Results of instrumented tests
Validated TRL 8

4.5 SSC-5 Reactor Pressure Vessel/Vessel Cooling System

TRL Rating Sheets, TRL 5 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-5.1	**Revision:** 1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Reactor Pressure Vessel (RPV)

Description:
The RPV houses the reactor, the reactor internals, and the reactor support structure. The RPV consists of a main cylindrical section with hemispherical upper and lower heads. The upper head, which is bolted to the cylindrical section, has penetrations for the neutron control assemblies and in-vessel flux monitoring unit. The lower section, which is welded to the cylindrical section, has penetrations for the Shutdown Cooling System, the In-Service Inspection access, and source range neutron detectors. For a 600 MWt prismatic NGNP, the RPV would be larger in diameter (about 7.2 m I.D) than most LWR vessels, but the wall thickness would be comparable. A direct vessel cooling system is used in the NGNP design to keep maximum vessel temperatures within ASME code limits.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Component verified at bench scale	Component verified at engineering scale	Subsystem verified at pilot scale
TRL	**4**	**5**	**6**

Basis for Rating (Check box if continued on additional sheets) ☒

As discussed in GA report PC-000566, GA has concluded that SA-508/533 steel should be used as the material of construction for the NGNP RPV. SA-508/533 steel has an extensive experience base as the material used for current generation LWR RPVs, and it has been codified in Section III of the ASME code. The RPV for a 600-MWt prismatic NGNP would be larger in diameter than most LWR vessels, but the wall thickness would be comparable, and it has been determined that forgings of the (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Develop RPV and VCS design requirements and the VCS conceptual design. This activity will include thermal-hydraulic analyses to calculate RPV temperatures and to assess the sensitivity of RPV temperatures to key parameters such as emissivity. Analyses will also be performed to define the design and expected helium impurity levels specific to (Cont.)	GA	9 months starting early in CD	1000

DDN(s) Supported: None	**Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein

Date: 11-27-08	**Originating Organization:** General Atomics

Additional Basis Sheet(s)

Basis:

required size are within the capabilities of a major forging supplier (Japan Steel Works). Use of a high-alloy steel with higher temperature capability would place less burden on optimizing the reactor design, but such materials pose a significant level of programmatic risk because of their very limited experience base for nuclear applications, lack of approval in Section III of the ASME code, and the lack of a credible source of manufacture.

GA has also concluded that it will be necessary to include an active vessel cooling system (VCS) in the NGNP design to ensure with high confidence that peak vessel operating temperatures are below the ASME code limit of 371°C for SA508/533 steel. Calculations performed by KAERI and presented in GA Report 911118 suggest that active vessel cooling may not be required if the reactor core inlet temperature is limited to 490°C, but the confidence level associated with the calculations was 50% and the RPV operating temperature margin was relatively small. The small operating temperature margin is a concern given that creep effects may need to be considered for an NGNP RPV fabricated from SA-508/533 if the operating metal temperatures are pushing against the 371°C boundary and the design lifetime of the RPV is very long (e.g., 60 years). Consequently, the VCS should be designed to keep maximum vessel operating temperatures well below 371°C.

Although, previous MHR designs have not included a VCS, the system is not envisioned to be particularly complex or to require development of any new technology. However, because of the importance of the system, it is expected that design verification testing of the RPV/VCS system will be necessary to advance the TRL level of the RPV/VCS system to 7.

Although there is a sufficient mechanical properties database for SA508/533, there is limited data available on the thermal aging effects on the mechanical properties, so additional information is needed on long-term aging effects. In particular, no data is available on the effects of impure helium on the long-term corrosion and mechanical properties of this material. Consequently, additional data on thermal aging and environmental effects are considered to be needed to support licensing. Also, as discussed in INL document PLN-2803, INL and ORNL have identified creep deformation as a potential concern for the NGNP SA-508/533 RPV and have recommended an extensive program of stress-rupture testing to address this concern. This concern derives from the 60-year design lifetime for the RPV and the assumption that the temperature of the RPV will be about 350°C during normal reactor operation.

In summary, GA has assigned a technology readiness level (TRL) of 5 to the RPV based on the extensive experience base for SA508/SA533 as the material of construction for current generation LWR RPVs and codification of this material in Section III of the ASME code. GA does not consider long-term creep effects to be a potential problem for the NGNP RPV based on the assumption that the VCS can be designed to keep RPV temperatures well below 350°C during normal reactor operations. Further, although some testing will be needed for confirmation and licensing purposes, GA does not believe that there are likely to be any significant deleterious effects of impure helium on the mechanical properties of the SA-508/533 vessel based on the experience with 2.25Cr-1Mo steel in the HTTR.

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
the NGNP design and operating conditions.			
2. Define required SA-508/533 testing program. This activity will involve preparation of an NGNP RPV materials research and development plan that is specific to the NGNP RPV/VCS conceptual design developed in Action 1. This plan will be based on INL document PLN-2803; however, it is believed that many of the tests recommended in PLN-2803 will be determined to be unnecessary because the VCS design will keep RPV temperatures well below 350°C during normal reactor operations.	GA, INL, and ORNL	4 months starting about one year into CD	350
3. Conduct SA-508/533 testing in accordance with the NGNP RPV materials research and development plan from action 2	INL, ORNL, and/or commercial materials testing laboratories	2.5 years starting as soon as the testing program has been defined	9000

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-5.2	**Revision:** 1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Reactor Pressure Vessel (RPV)

Description:
The RPV houses the reactor, the reactor internals, and the reactor support structure. The RPV consists of a main cylindrical section with hemispherical upper and lower heads. The upper head, which is bolted to the cylindrical section, has penetrations for the neutron control assemblies and in-vessel flux monitoring unit. The lower section, which is welded to the cylindrical section, has penetrations for the Shutdown Cooling System, the In-Service Inspection access, and source range neutron detectors. For a 600 MWt prismatic NGNP, the RPV would be larger in diameter (about 7.2 m I.D) than most LWR vessels, but the wall thickness would be comparable. A direct vessel cooling system is used in the NGNP design to keep maximum vessel temperatures within ASME code limits.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Component verified at experimental scale	Component verified at pilot scale	Component verified at engineering scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 6 is achieved when the following conditions are met: (1) The design requirements for the RPV/VCS system have been defined and a conceptual design as been developed that limits RPV temperatures during normal reactor operation to less than 350°C with adequate margin, and (2) the necessary testing program for SA-508/533 has been defined and performed, and the data needed to support final design of the RPV/VCS and to support NGNP licensing has been obtained.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Develop the final design of the RPV, and prepare and issue the procurement specifications for the RPV	GA	6 months starting early in FD	350

DDN(s) Supported: New DDN needed	**Technology Case File:**
Subject Matter Expert Making Determination: John Saurwein	
Date: 11-27-08	**Originating Organization:** General Atomics

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Develop the detailed design of the RPV/VCS and confirm that the design satisfies all ASME code rules for the RPV. Perform analyses to verify with high confidence that the maximum RPV temperatures will be within ASME code limits for SA-508/533 with adequate margin to eliminate concerns about creep effects over a 60-year lifetime.	GA	1 year starting about 1.5 years into NGNP FD	700
3. Conduct design verification testing of an engineering-scale model of the vessel and vessel cooling system. It is anticipated that design verification testing of the RPV/VCS system will be performed concurrently with design verification testing of the reactor core as discussed in the Test Plan for the reactor core assembly (GA Test Plan 911135). The first part of this activity will be to design the test and to prepare the Test Specification. However, it is anticipated that the engineering-scale model will include a heat source and will simulate the vessel, core barrel, permanent side reflector, and the upper and lower plenums to the extent necessary to precisely represent the flow path(s) for direct vessel cooling and the potential paths for in-leakage of primary coolant into the direct vessel cooling flow path(s). The engineering-scale model will include the necessary instrumentation to record vessel temperatures and helium flow rates during the test.	INL CTF Other alternatives include Wyle Laboratories and perhaps GA	2 years, must be completed about 2 years prior to start of NGNP startup testing	6,000 (incremental cost of RPV/VCS testing is estimated at about $1M)

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-5.3	**Revision:**	1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Reactor Pressure Vessel (RPV)

Description:
The RPV houses the reactor, the reactor internals, and the reactor support structure. The RPV consists of a main cylindrical section with hemispherical upper and lower heads. The upper head, which is bolted to the cylindrical section, has penetrations for the neutron control assemblies and in-vessel flux monitoring unit. The lower section, which is welded to the cylindrical section, has penetrations for the Shutdown Cooling System, the In-Service Inspection access, and source range neutron detectors. For a 600 MWt prismatic NGNP, the RPV would be larger in diameter (about 7.2 m I.D) than most LWR vessels, but the wall thickness would be comparable. A direct vessel cooling system is used in the NGNP design to keep maximum vessel temperatures within ASME code limits.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Component verified at pilot scale	Component verified at engineering scale	Component tested and qualified
TRL	**6**	**7**	**8**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 7 is achieved when the following conditions are met: (1) The engineering-scale test of the vessel and VCS has been completed, (2) the results of the test confirm the VCS thermal/hydraulic characteristics and the RPV temperature distributions predicted using the detailed RPV/VCS computer model developed during final design, and (3) any significant discrepancies between the results obtained with the computer and physical models have been resolved and factored into the VCS final design.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Conduct testing of the RPV/VCS in the actual operating environment (i.e., in the NGNP during start-up testing) to verify that the system meets reliability requirements and maintains vessel temperatures within the limits defined in the NGNP technical specifications. The first part of this activity will be to prepare the Test Specification (or alternately to define the test in the NGNP start-up plan).	GA, NGNP operator	During NGNP start-up testing	Cost to be covered under NGNP start-up testing

DDN(s) Supported: None **Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein

Date: 11-27-08 **Originating Organization:** General Atomics

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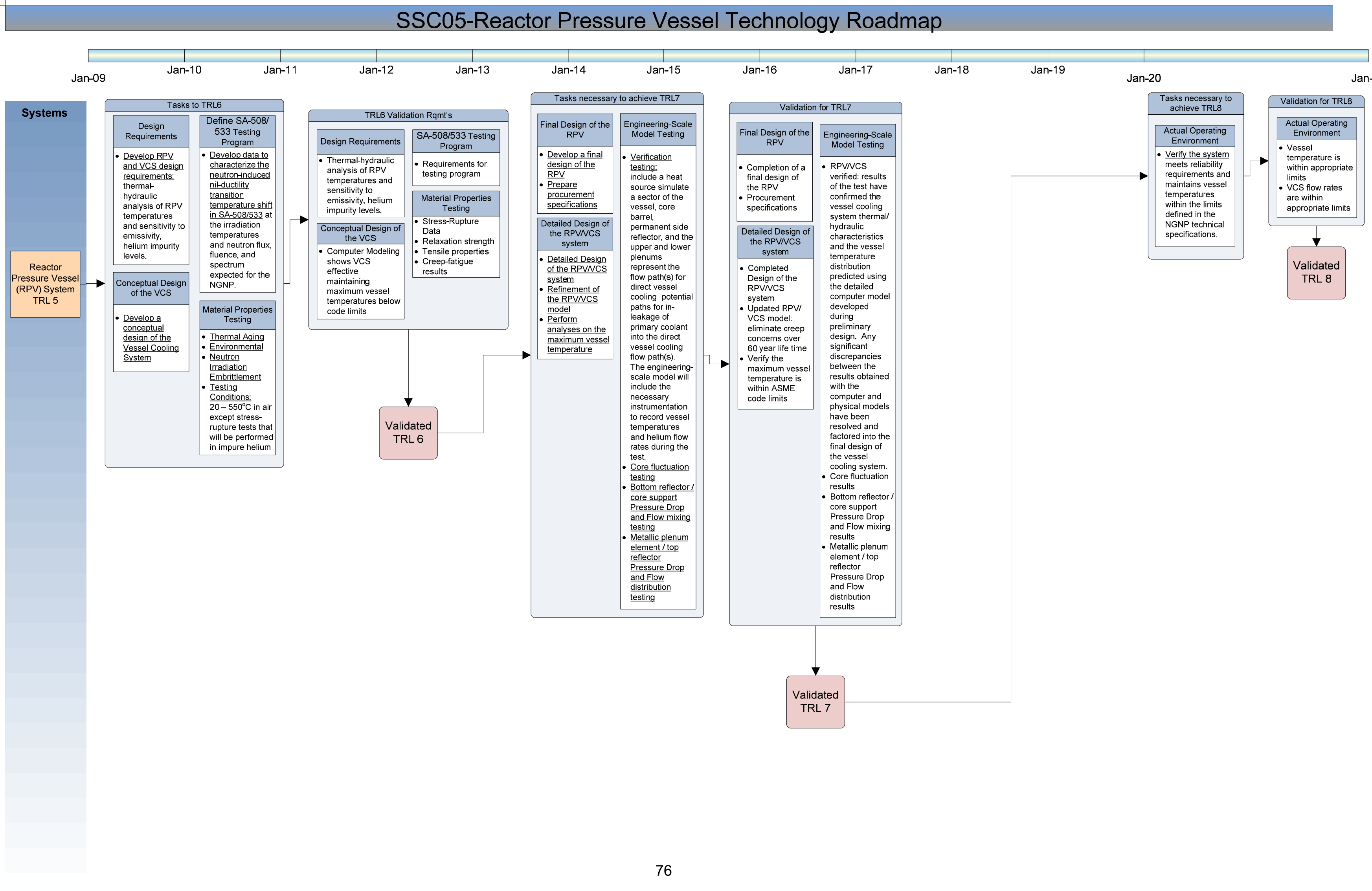
Rev. 1



4.6 SSC-6 Helium Circulator

TRL Rating Sheets, TRL 6 and 7

Technology Development Road Map

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-6.1	**Revision:**	1

☐ Area ☐ System ☒ Subsystem/Structure ☐ Component ☐ Technology

Title: Helium Circulators (PHTS, SCS, SHTS)

Description:
Main Circulator: The NGNP circulator is a variable speed, electric motor-driven axial flow helium compressor that facilitates thermal energy transfer from the reactor core to the steam generator or Intermediate Heat Exchanger (IHX) and, hence, to the external turbo-generator set.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Item verified at experimental scale	Item verified at pilot scale	Item verified at engineering scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☒

The NGNP helium circulator development builds on earlier studies carried out by circulator vendors for GA. The design proposed by Howden for the MHTGR program in 1989 was a two-stage axial flow machine running at 4500 rpm, with a maximum power rating of 4 MWe. It featured an induction motor and an AMB system. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. **Bearing Design Verification:** a. Determine static and dynamic axial thrust load capacities, stiffness, and damping coefficients over the operating speed range. b. Determine sensitivity of the associated electronic control system to external disturbances c. Rotor dynamic response to externally induced unbalance loads occurring in the impeller plane of rotation d. Magnitude of drag losses	Vendor, INL CTF or PBMR HTF	2012-2013	2,900

DDN(s) Supported: C.14.01.01, M.21.01.01, M.21.01.03	**Technology Case File:**
Subject Matter Expert Making Determination:	Puja Gupta
Date: 12-8-08	**Originating Organization:** General Atomics

Additional Basis Sheet(s)

Basis:

Further to this, in 1993, Howden also designed the helium circulator for the New Production-Modular High Temperature Gas-cooled Reactor (NP-MHTGR) program. The selected design had radial flow impeller, oil-bath lubricated bearings, submerged motor drive, rotational speed of about 3000 rpm and a maximum power level of approximately 6 MWe. The James Howden Company has designed and built 112 machines for the commercial Advanced Gas Reactor (AGR) plants. Howden has designed a 4 MWe helium circulator to the concept stage for GA. Data on helium circulators are primarily available from component testing performed for Fort St. Vrain and the proposed Delmarva plant. The database has applicability limited to the design of axial compressors and shutoff valves. Considerable operating experience with magnetic bearings in various industrial applications has been accumulated, and covers the size and load range of a circulator of 4 to 5 MWe. Societe de Mecanique Magnetique (S2M), the world's leading manufacturer of magnetic bearings, has some proprietary data under various non-representative conditions. There is also experience with magnetic bearings for use in centrifuge enrichment equipment as part of some classified government programs. Part of this work has recently been declassified.

Data on characteristics and performance of AMBs operating in conditions representative of the NGNP MC environment have not been established. There is a lack of data on the reliability of backup "catcher" bearings for vertical rotors to repeatedly support the turning rotor for a limited time when the active magnetic field supporting the rotor is lost.

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
e. Development testing of alternate bearings, operating procedures, lubricants, and/or materials, if the reference design is unsatisfactory f. Evaluation of aerodynamic load simulation, including decay, in the test rig g. Demonstrate capability of catcher bearings to support the full scale vertical circulator rotor with failed AMBs during the coast down at all steady state, transient pressurized and depressurized operating conditions in helium			
2. Scale Model Circulator Aerodynamic Flow Testing: a. Determine pressure rise across the compressor as a function of speed and helium flow through the compressor b. Determine overall efficiency including impeller and diffuser efficiency	Vendor, INL CTF or PBMR HTF	2012-2013	1,100
3. Motor Cooling Design and Insulation Dielectric Strength Verification: a. Measure necessary buffer gas flow to prevent the leakage of radioactive helium into the motor cavity b. Perform flashover tests in air at atmospheric pressure and in helium at three pressures: atmospheric, operating pressure, and an intermediate pressure to obtain flashover data as a function of helium pressure for the various insulation and rectifier components c. Obtain corona start data for the stator insulation versus helium pressure. d. Confirm the satisfactory performance of insulation and diode by test.	Vendor, INL CTF or PBMR HTF	2014	550

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-6.2	**Revision:** 1

☐ Area ☐ System ☒ Subsystem/Structure ☐ Component ☐ Technology

Title: Helium Circulators (PHTS, SCS, SHTS)

Description:
Main Circulator: The NGNP circulator is a variable speed, electric motor-driven axial flow helium compressor that facilitates thermal energy transfer from the reactor core to the steam generator or Intermediate Heat Exchanger (IHX) and, hence, to the external turbo-generator set.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Item verified at pilot scale	Item verified at engineering scale	Item tested and qualified
TRL	**6**	**7**	**8**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 7 is achieved upon successful completion of the testing outlined in the TRL rating sheet for TRL 6 (and Section 3 of Test Plan 911138). Successful completion of these tests would demonstrate reliability/availability goals for the components such as insulation, diodes, motor cooling heat exchanger, shaft mounted motor cooling fans, journal and thrust AMB, catcher bearings, as well as full size labyrinth shaft seal have demonstrated reliability/availability goals under relevant environment.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Prototype Circulator Design Verification a. Buffer Helium Transient Tests b. Shaft Brake Test c. Low Speed Test d. Hot Restart Test e. Rapid Depressurization Test f. Endurance Test g. Acoustic and Vibration Test h. Spin Test	INL CTF or NGNP Prototype Location	2014-2016	25,000

DDN(s) Supported: C.14.01.03, M.21.01.02, M.57.01.02	**Technology Case File:**

Subject Matter Expert Making Determination: Puja Gupta

Date: 12-8-08	**Originating Organization:** General Atomics

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Extended Duration Testing	INL CTF or NGNP Prototype Location	2017 (qt 1,2)	
3. Modified Main Circulator Testing (if necessary)	INL CTF or NGNP Prototype Location	2017 (qt 3,4)	
4. Addition Circulators Proof Testing in Support of First Plant Operation	INL CTF or NGNP Prototype Location	2018-2020	

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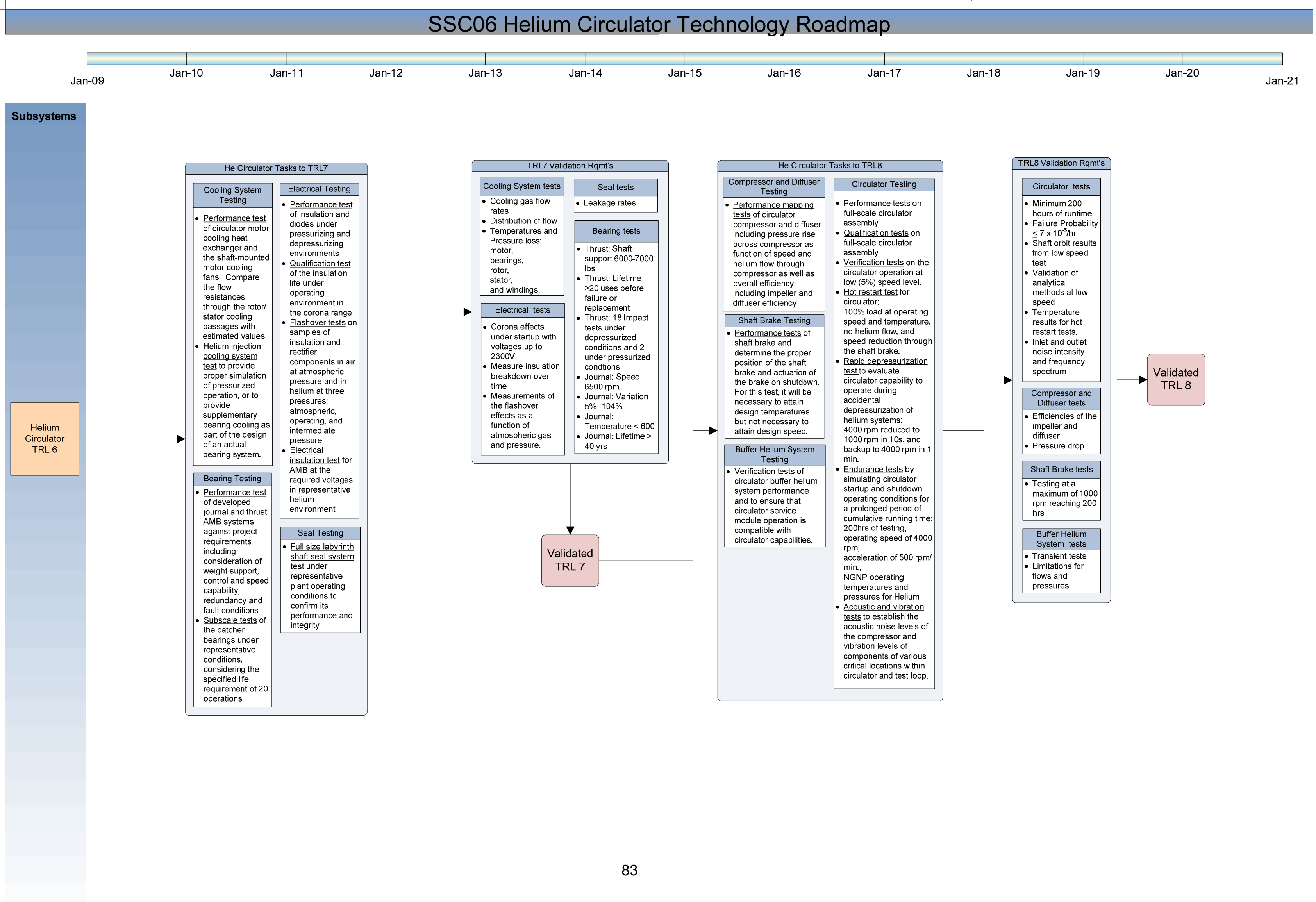


4.7 SSC-7 Intermediate Heat Exchanger

TRL Rating Sheets, TRL 2 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-7.1	**Revision:**	1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Compact Intermediate Heat Exchanger (IHX)

Description:
The IHX is a high temperature gas-to-gas heat exchanger that transfers thermal energy from the NGNP primary coolant to a secondary loop. It is subject to temperatures of up to 950°C on the primary side and temperatures 25°C to 50°C lower on the secondary side. The pressure is 7 MPa on the primary side and 7.1 MPa on the secondary side. GA has selected helium to be the working fluid in the secondary loop. The NGNP IHX will have a heat transfer duty of 65 MWt in GA's preferred NGNP configuration, but could have a heat transfer duty of up to about 600 MWt in other possible NGNP configurations. It is assumed that the IHX will be a compact printed-circuit type heat exchanger (PCHE) comprised of a number of identical PCHE modules. The PCHE modules are fabricated by etching channels into metal plates and diffusion bonding the plates together. These modules along with the connecting ducting, headers, supports, etc. comprise the heat transfer subsystem, which is enclosed within a pressure vessel. The vessel and heat transfer internals comprise the IHX system.

Area:	☐ NHSS	☒ HTS	☐ HPS	☐ BCS	☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Basic principles observed	Application formulated	Proof of concept
TRL	**1**	**2**	**3**

Basis for Rating (Check box if continued on additional sheets) ☒

PCHE technology has been developed and commercially deployed by Heatric Corp, but for non-nuclear applications, and there has been no demonstration of a PCHE of the size required for the NGNP or in the expected operating conditions of the NGNP. For NGNP, the PCHE module and other internal components must be fabricated from a material that has adequate creep and fatigue strength at 950°C and is resistant to deleterious aging in a high-temperature impure helium environment. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Perform modeling to confirm the feasibility of an integrated IHX design of the required size. (Cont.)	GA/IHX Vendor	6 months (Beginning of CD)	150

DDN(s) Supported: N13.02.01, N13.02.02	**Technology Case File:**

Subject Matter Expert Making Determination: D. Carosella, A. Bozek, J. Saurwein

Date: 12-10-08	**Originating Organization:** General Atomics

Additional Basis Sheet(s)

Basis for rating:

The NGNP Technology Development Program has identified Alloy 617 as the preferred material for the PCHE and Haynes 230 as a potential backup material. However, the high cobalt content (i.e., 10 to 15%) of Alloy 617 has been a concern to GA. Specifically, erosion of cobalt-containing surface scales that may form as a result of reactions between Alloy 617 and impurities in the helium may result in entrainment of Co particulates in the primary coolant. Activation of these particulates in the reactor core would result in a high level of radioactivity in the primary coolant loop. However, recent measurements by INL of the cobalt content of the surface scale that formed on Alloy-617 samples aged in an oxidizing impure helium environment revealed that the cobalt content of the surface scale was only about 0.2%. So as long as the helium coolant chemistry supports the formation of a stable oxide film on the Alloy 617, cobalt entrainment in the primary coolant should not be a significant problem.

The Heatric Corp is currently conducting a PCHE development program for Alloy 617. Heatric has recently reported success in fabricating diffusion bonded specimens that meet Alloy 617 strength requirements and in fabricating a demonstration PCHE module that meets Heatric's leakage requirements. Unfortunately, Heatric is very secretive about its PCHE design and fabrication processes, so development of design rules within the ASME code for a compact heat exchanger based on Heatric PCHE technology may not be possible.

It is planned to use LWR reactor steel (SA508/SA533) for the NGNP IHX vessel, and sufficient insulation will be needed to keep vessel temperatures below ASME code limits. The design of the NGNP IHX vessel thermal barrier is expected to be similar to the IHX vessel thermal barrier design for the HTTR IHX in Japan. Operation of the HTTR IHX with a helium inlet temperature of 950°C has been demonstrated. However, the NGNP IHX vessel thermal barrier will still have to be tested to verify its performance and durability within the expected NGNP operating environment. Based on the demonstration of IHX vessel thermal barrier technology in the HTTR, the technology maturity of the vessel is considered to be at least TRL = 4, so the TRL of the vessel is not limiting with respect to the overall technology readiness of the IHX.

In spite of the above-noted progress in demonstrating the availability of a suitable material (Alloy 617) for the IHX, an initial TRL of 2 is assigned to the IHX because the technology required to build an integrated IHX of the size required for NGNP is not judged to be sufficiently mature to warrant a TRL rating of 3 (proof of concept). Also, additional environmental aging and thermal cycling testing of Alloy 617 may be needed to more-conclusively prove that there is minimal potential to introduce cobalt into the primary helium coolant under all realistic operating conditions for the NGNP.

Additional Action Sheets(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Experimentally determine environmental aging and thermal cycling effects on Alloy 617 at elevated temperature in plausible impure helium environments to confirm that Alloy 617 is a suitable material (or test other materials to identify a viable alternative).	INL and/or ORNL	1.5 years (Complete during CD)	1000
3. Verify diffusion bonding and chemical etching processes for selected material (assumed to be Alloy 617)	IHX vendor	1 year (Complete during CD)	200

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-7.2	**Revision:**	1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Compact Intermediate Heat Exchanger (IHX)

Description:
The IHX is a high temperature gas-to-gas heat exchanger that transfers thermal energy from the NGNP primary coolant to a secondary loop. It is subject to temperatures of up to 950°C on the primary side and temperatures 25°C to 50°C lower on the secondary side. The pressure is 7 MPa on the primary side and 7.1 MPa on the secondary side. GA has selected helium to be the working fluid in the secondary loop. The NGNP IHX will have a heat transfer duty of 65 MWt in GA's preferred NGNP configuration, but could have a heat transfer duty of up to about 600 MWt in other possible NGNP configurations. It is assumed that the IHX will be a compact printed-circuit type heat exchanger (PCHE) comprised of a number of identical PCHE modules. The PCHE modules are fabricated by etching channels into metal plates and diffusion bonding the plates together. These modules along with the connecting ducting, headers, supports, etc. comprise the heat transfer subsystem, which is enclosed within a pressure vessel. The vessel and heat transfer internals comprise the IHX system.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Application formulated	Proof of concept	Component verified at bench scale
TRL	**2**	**3**	**4**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 3 is achieved after CFD analysis and structural analysis using models based on the initial design concept for the IHX internals (i.e., PCHE modules with connecting ducts, headers, supports, etc.) have verified the feasibility of the concept and that the design should have acceptable thermal-hydraulic and structural performance. Proof of concept has been further established by (1) completing the environmental aging and thermal cycling testing required to confirm Alloy 617 (or an alternate material) as a suitable material of construction for the PCHE, and (2) demonstration of the ability to achieve the required strength in diffusion bonded samples of Alloy 617 (or alternate material).

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Construct a bench-scale PCHE module from the selected material and perform tests to confirm (1) acceptable leak tightness, and (2) the temperature distribution within the module is consistent with temperature predictions from analytical modeling.	IHX vendor	1 year (Complete early in preliminary design)	1000

DDN(s) Supported: N13.02.04 | **Technology Case File:**

Subject Matter Expert Making Determination: D. Carosella, A. Bozek, J. Saurwein

Date: 12-10-08 | **Originating Organization:** General Atomics

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-7.3	**Revision:**	1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Compact Intermediate Heat Exchanger (IHX)

Description:
The IHX is a high temperature gas-to-gas heat exchanger that transfers thermal energy from the NGNP primary coolant to a secondary loop. It is subject to temperatures of up to 950°C on the primary side and temperatures 25°C to 50°C lower on the secondary side. The pressure is 7 MPa on the primary side and 7.1 MPa on the secondary side. GA has selected helium to be the working fluid in the secondary loop. The NGNP IHX will have a heat transfer duty of 65 MWt in GA's preferred NGNP configuration, but could have a heat transfer duty of up to about 600 MWt in other possible NGNP configurations. It is assumed that the IHX will be a compact printed-circuit type heat exchanger (PCHE) comprised of a number of identical PCHE modules. The PCHE modules are fabricated by etching channels into metal plates and diffusion bonding the plates together. These modules along with the connecting ducting, headers, supports, etc. comprise the heat transfer subsystem, which is enclosed within a pressure vessel. The vessel and heat transfer internals comprise the IHX system.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Proof of concept	Components verified at bench scale	Components verified at experimental scale
TRL	**3**	**4**	**5**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 4 is achieved when testing of a bench-scale PCHE module has been completed and the test results indicate (1) acceptable leak tightness, and (2) the temperature distribution within the module is consistent with temperature predictions from analytical modeling.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Complete design of the IHX vessel thermal barrier, construct an experimental-scale model of the vessel, and perform tests to verify acceptable operating characteristics of the thermal barrier as a function of thermal cycling, mechanical vibration, and flow and thermal gradients.	GA/IHX vendor	2 years (Complete by end of PD)	2000

DDN(s) Supported: N13.02.07	**Technology Case File:**

Subject Matter Expert Making Determination: D. Carosella, A. Bozek, J. Saurwein

Date: 12-10-08	**Originating Organization:** General Atomics

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Cut specimens representative of diffusion-bonded plates and heat-affected areas (from welding of module-connecting piping, supports, etc.) from the bench-scale model and subject the specimens to mechanical properties and environmental aging tests.	IHX vendor and INL and/or ORNL	18 months (Complete by end of PD)	4000

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-7.4	**Revision:** 1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Compact Intermediate Heat Exchanger (IHX)

Description:
The IHX is a high temperature gas-to-gas heat exchanger that transfers thermal energy from the NGNP primary coolant to a secondary loop. It is subject to temperatures of up to 950°C on the primary side and temperatures 25°C to 50°C lower on the secondary side. The pressure is 7 MPa on the primary side and 7.1 MPa on the secondary side. GA has selected helium to be the working fluid in the secondary loop. The NGNP IHX will have a heat transfer duty of 65 MWt in GA's preferred NGNP configuration, but could have a heat transfer duty of up to about 600 MWt in other possible NGNP configurations. It is assumed that the IHX will be a compact printed-circuit type heat exchanger (PCHE) comprised of a number of identical PCHE modules. The PCHE modules are fabricated by etching channels into metal plates and diffusion bonding the plates together. These modules along with the connecting ducting, headers, supports, etc. comprise the heat transfer subsystem, which is enclosed within a pressure vessel. The vessel and heat transfer internals comprise the IHX system.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Components verified at bench scale	Components verified at experimental scale	System verified at pilot scale
TRL	**4**	**5**	**6**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 5 is achieved when the PCHE module design and fabrication process have been verified by testing, as has the IHX vessel thermal barrier design.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Use computer modeling to size and configure the IHX heat transfer subsystem to meet operational requirements including heat transport duty, pressure drop, operating lifetime, etc.	IHX vendor	6 months (Complete by end of PD)	150

DDN(s) Supported: None	**Technology Case File:**
Subject Matter Expert Making Determination: D. Carosella, A. Bozek, J. Saurwein	
Date: 12-10-08	**Originating Organization:** General Atomics

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-7.5	**Revision:**	1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Compact Intermediate Heat Exchanger (IHX)

Description:
The IHX is a high temperature gas-to-gas heat exchanger that transfers thermal energy from the NGNP primary coolant to a secondary loop. It is subject to temperatures of up to 950°C on the primary side and temperatures 25°C to 50°C lower on the secondary side. The pressure is 7 MPa on the primary side and 7.1 MPa on the secondary side. GA has selected helium to be the working fluid in the secondary loop. The NGNP IHX will have a heat transfer duty of 65 MWt in GA's preferred NGNP configuration, but could have a heat transfer duty of up to about 600 MWt in other possible NGNP configurations. It is assumed that the IHX will be a compact printed-circuit type heat exchanger (PCHE) comprised of a number of identical PCHE modules. The PCHE modules are fabricated by etching channels into metal plates and diffusion bonding the plates together. These modules along with the connecting ducting, headers, supports, etc. comprise the heat transfer subsystem, which is enclosed within a pressure vessel. The vessel and heat transfer internals comprise the IHX system.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Components verified at experimental scale	System verified at pilot scale	System verified at engineering scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 6 is achieved when the computer modeling has been completed to size and configure the IHX heat transfer subsystem to meet anticipated operational requirements including heat transport duty, pressure drop, operating lifetime, etc.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Perform design support testing on engineering-scale mock-ups to verify critical design features of the IHX. The required testing will be design specific and will be determined during IHX preliminary design. However, it is anticipated that the following testing will be needed. (Cont.)	GA/IHX vendor or CTF	3 years beginning at start of final design	10,000

DDN(s) Supported: N.13.02.03, N.13.02.05, N.13.02.06; N.13.02.07, N.13.02.08, N.13.02.09

Technology Case File:

Subject Matter Expert Making Determination: D. Carosella, A. Bozek, J. Saurwein

Date: 10/16/08 **Originating Organization:** General Atomics

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
a. Perform testing to confirm the predicted thermal and hydraulic characteristics, including heat transfer, vessel temperatures, and overall IHX system pressure losses b. Confirm by experiment the flow distribution throughout the IHX (both primary and secondary inlets and outlets) accompanied by analytical evaluation. c. Perform testing to obtain data on the frequency spectra and sound pressure levels that may be generated by the IHX as a function of flow velocities d. Perform testing to determine the physical and operational characteristics of insulation relative to thermal cycling, mechanical and acoustic vibrations, and the effects of flow and thermal gradients. e. Various sliding seals, expansion joints, and other seals are expected in the IHX design for installation and replacement purposes. Perform testing to obtain the data needed to confirm the design feasibility, measure leak rates under operating conditions, and measure the influence of various factors on seal performance. f. Perform testing to obtain the data needed to accurately determine the flow-induced vibration characteristics around the IHX and its associated piping. The flow induced excitation mechanisms of concern are turbulent buffeting, vortex shedding and fluid elastic instability.			
2. Finalize IHX design and issue procurement specifications for prototype IHX	GA/IHX vendor	15 months	N/A (design cost)

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-7.6	**Revision:**	1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Compact Intermediate Heat Exchanger (IHX)

Description:
The IHX is a high temperature gas-to-gas heat exchanger that transfers thermal energy from the NGNP primary coolant to a secondary loop. It is subject to 950°C, 7MPa helium on the primary side and a variety of possible conditions on the secondary side. The working fluid in the secondary loop may be helium or a mixture of helium and nitrogen. The NGNP IHX will have a minimum heat transfer duty of 65 MWt, and could have a heat transfer duty of up to about 600 MWt depending on the NGNP design. It is assumed that the IHX will be a compact printed-circuit type heat exchanger (PCHE) comprised of a number of identical PCHE modules. The PCHE modules are fabricated by etching channels into metal plates and diffusion bonding the plates together. These modules along with the connecting ducting, headers, supports, etc. comprise the heat transfer subsystem, which is enclosed within a pressure vessel. The vessel and heat transfer internals comprise the IHX system.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Subsystem verified at pilot scale	System verified at engineering scale	System tested and qualified
TRL	**6**	**7**	**8**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 7 is achieved when the engineering-scale testing of IHX mock-ups in a relevant environment (as outlined in the TRL rating sheet for TRL 6) has confirmed the thermal and hydraulic characteristics predicted by modeling (or the models have been modified to reflect test results), verified acceptable leakage rates for the IHX internals, and confirmed the acoustic and vibrational stability of the design.

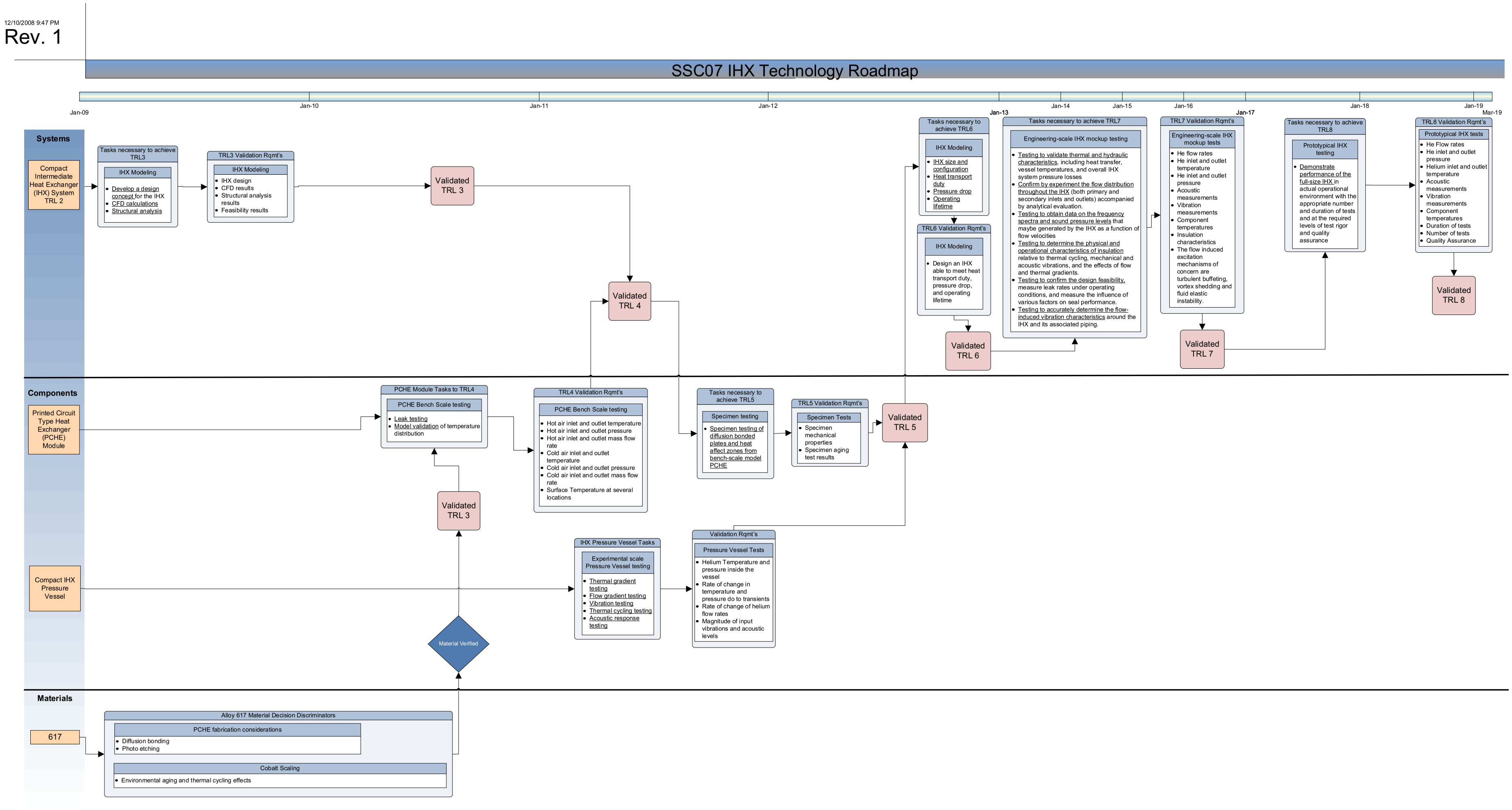
Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Demonstrate a full-size IHX prototype in the NGNP operational environment with the appropriate number and duration of tests and at the required levels of test rigor and quality assurance. Subject the IHX to an appropriate number of transient and off-design condition cycles to demonstrate the performance of the IHX under these conditions.	CTF	2 years with completion six months before NGNP startup testing	3000

DDN(s) Supported: None **Technology Case File:**

Subject Matter Expert Making Determination: D. Carosella, A. Bozek, J. Saurwein

Date: 12-10-08 **Originating Organization:** General Atomics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Rev. 1
SSC07 IHX Technology Roadmap
Jan-09
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Mar-19
Systems
Compact Intermediate Heat Exchanger (IHX) System TRL 2
Tasks necessary to achieve TRL3
IHX Modeling
Develop a design concept for the IHX
CFD calculations
Structural analysis
TRL3 Validation Rqmt's
IHX Modeling
IHX design
CFD results
Structural analysis results
Feasibility results
Validated TRL 3
Validated TRL 4
Tasks necessary to achieve TRL6
IHX Modeling
IHX size and configuration
Heat transport duty
Pressure drop
Operating lifetime
TRL6 Validation Rqmt's
IHX Modeling
Design an IHX able to meet heat transport duty, pressure drop, and operating lifetime
Validated TRL 6
Tasks necessary to achieve TRL7
Engineering-scale IHX mockup testing
Testing to validate thermal and hydraulic characteristics, including heat transfer, vessel temperatures, and overall IHX system pressure losses
Confirm by experiment the flow distribution throughout the IHX (both primary and secondary inlets and outlets) accompanied by analytical evaluation.
Testing to obtain data on the frequency spectra and sound pressure levels that maybe generated by the IHX as a function of flow velocities
Testing to determine the physical and operational characteristics of insulation relative to thermal cycling, mechanical and acoustic vibrations, and the effects of flow and thermal gradients.
Testing to confirm the design feasibility, measure leak rates under operating conditions, and measure the influence of various factors on seal performance.
Testing to accurately determine the flow-induced vibration characteristics around the IHX and its associated piping.
TRL7 Validation Rqmt's
Engineering-scale IHX mockup tests
He flow rates
He inlet and outlet temperature
He inlet and outlet pressure
Acoustic measurements
Vibration measurements
Component temperatures
Insulation characteristics
The flow induced excitation mechanisms of concern are turbulent buffeting, vortex shedding and fluid elastic instability.
Validated TRL 7
Tasks necessary to achieve TRL8
Prototypical IHX testing
Demonstrate performance of the full-size IHX in actual operational environment with the appropriate number and duration of tests and at the required levels of test rigor and quality assurance
TRL8 Validation Rqmt's
Prototypical IHX tests
He Flow rates
He inlet and outlet pressure
Helium inlet and outlet temperature
Acoustic measurements
Vibration measurements
Component temperatures
Duration of tests
Number of tests
Quality Assurance
Validated TRL 8
Components
Printed Circuit Type Heat Exchanger (PCHE) Module
PCHE Module Tasks to TRL4
PCHE Bench Scale testing
Leak testing
Model validation of temperature distribution
TRL4 Validation Rqmt's
PCHE Bench Scale testing
Hot air inlet and outlet temperature
Hot air inlet and outlet pressure
Hot air inlet and outlet mass flow rate
Cold air inlet and outlet temperature
Cold air inlet and outlet pressure
Cold air inlet and outlet mass flow rate
Surface Temperature at several locations
Tasks necessary to achieve TRL5
Specimen testing
Specimen testing of diffusion bonded plates and heat affect zones from bench-scale model PCHE
TRL5 Validation Rqmt's
Specimen Tests
Specimen mechanical properties
Specimen aging test results
Validated TRL 5
Validated TRL 3
Compact IHX Pressure Vessel
IHX Pressure Vessel Tasks
Experimental scale Pressure Vessel testing
Thermal gradient testing
Flow gradient testing
Vibration testing
Thermal cycling testing
Acoustic response testing
Validation Rqmt's
Pressure Vessel Tests
Helium Temperature and pressure inside the vessel
Rate of change in temperature and pressure do to transients
Rate of change of helium flow rates
Magnitude of input vibrations and acoustic levels
Material Verified
Materials
617
Alloy 617 Material Decision Discriminators
PCHE fabrication considerations
Diffusion bonding
Photo etching
Cobalt Scaling
Environmental aging and thermal cycling effects

4.8 SSC-8 Shutdown Cooling Heat Exchanger

TRL Rating Sheets, TRL 4 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-8.1	**Revision:** 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Shutdown Cooling Heat Exchanger (SCHE)

Description:
The Shutdown Cooling Heat Exchanger (SCHE) is a multi-tube helical coil heat exchanger. It is similar in design to the evaporator/economizer portion of the FSV steam generator. Its function is to cool the reactor whenever the primary cooling system is not available. It is a vertical cross-counter flow heat exchanger. The tubes are made of 2-1/4 Croloy. The heat is removed by 60°C-pressurized (4.8MPa) water. The SCHE does not have a safety function.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Proof of concept	Verified at bench scale	Verified at experimental scale
TRL	**3**	**4**	**5**

Basis for Rating (Check box if continued on additional sheets) ☐

The Shutdown Cooling Heat Exchanger (SCHE) will be a helical coil tube heat exchanger similar in design to successfully operating heat exchangers in other gas cooled reactor plants including FSV and THTR. The previous experience with helical coiled heat exchangers has shown that the heat transfer correlations for flow across tube bundles match the predicted values (ASME Paper 79-WA/NE-1) thus providing a starting point for heat exchanger sizing. The selected tube material is 2-1/4 Croloy.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Computer models will be used to evaluate the following: 1) The heat exchanger thermal sizing which is based on pressurized cooldown from 100% power operation. 2) The heat exchanger gas side pressure drop evaluation, which is based on depressurized cooldown from 100% power operation. 3) The structural evaluation of the tubes, which is based on the maximum tube temperature in a hot streak location. (Cont.)	GA	2 years starting at beginning of conceptual design	2000

DDN(s) Supported: None	**Technology Case File:**
Subject Matter Expert Making Determination: Dave Carosella, Bob Schleicher	
Date: 12-9-08	**Originating Organization:** General Atomics

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Actions: The shroud material, the shroud insulation and the shroud insulation cover sheet must be selected based on evaluation of the effect of the environment on the shroud its insulation and the insulation cover sheet. Analysis can be used to determine the shroud, the insulation and the insulation cover sheet temperature levels. Temperature levels must be determined for the following operating conditions: steady state operation at full power, conduction cooldown and shutdown on the SCHE. The possible shroud and cover sheet material choices include Alloy 800H, Inconel 617, Haynes 230 or Hastelloy XR. The possible shroud insulation choices include: Kaowool, Alltemp Insulation and porous carbon Insulation.			

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-8.2	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Shutdown Cooling Heat Exchanger (SCHE)

Description:
The Shutdown Cooling Heat Exchanger (SCHE) is a multi-tube helical coil heat exchanger. It is similar in design to the evaporator/economizer portion of the FSV steam generator. Its function is to cool the reactor whenever the primary cooling system is not available. It is a vertical cross-counter flow heat exchanger. The tubes are made of 2-1/4 Croloy. The heat is removed by 60^{o}C-pressurized (4.8MPa) water. The SCHE does not have a safety function.

Area:	☒ NHSS	☐ HTS	☐ HPS	☐ BCS	☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Verified at bench scale	Verified at experimental scale	Verified at pilot scale
TRL	**4**	**5**	**6**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 5 will be achieved upon completion of the computer modeling activity described in the TRL rating sheet for TRL 4. In the computer modeling task, the heat exchanger was sized; the heat exchanger pressure drop was evaluated; the structural analysis was performed; and the temperatures and stress levels of the shroud, the insulation, and the insulation cover plates were calculated and used to select the materials for these components.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
The following tests will be performed on a mockup of an actual heat exchanger bundle with shrouds. 1. Perform testing of the SCHE shroud seal to determine measured leakage rates for combinations of different surface finishes, flatness tolerances, (cont.)	GA/SCHE vendor	2 years starting at beginning of preliminary design	2150

DDN(s) Supported: C.14.04.01, C.14.04.05, C.14.04.06, C.14.04.07	**Technology Case File:**
Subject Matter Expert Making Determination: Dave Carosella, Bob Schleicher	
Date: 12-9-08	**Originating Organization:** General Atomics

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
loads, and differential pressures under typical operating conditions. The shroud seal design consists of a metal bellows spring that compresses a circumferential seal. The testing will also include mechanical testing of the bellow assembly 2. Perform tests to characterize inlet flow and temperatures distributions under various operating modes and to determine the need for inlet flow distribution devices. If such devices are determined to be needed, perform tests to evaluate and compare candidate designs. 3. Perform tests to investigate the effects of acoustic noise on the large surface area structures of the SCHE. These tests will include the effects of coolant flow through the tube bundle so that the combined effects of acoustic noise and flow induced vibration can be assessed. 4. Evaluate the frequency response and dynamic loads of the insulation cover plates and attachments during the flow distribution and acoustics testing. 5. Determine the pressure drop flow characteristics of the water-side inlet orifice. Note: SCHE DDNs C.14.04.02, C.14.04.03, and C.14.04.09 will be satisfied by the steam generator technology development program			

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-8.3	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Shutdown Cooling Heat Exchanger (SCHE)

Description:
The Shutdown Cooling Heat Exchanger (SCHE) is a multi-tube helical coil heat exchanger. It is similar in design to the evaporator/economizer portion of the FSV steam generator. Its function is to cool the reactor whenever the primary cooling system is not available. It is a vertical cross-counter flow heat exchanger. The tubes are made of 2-1/4 Croloy. The heat is removed by 60°C-pressurized (4.8MPa) water. The SCHE does not have a safety function.

Area:	☒ NHSS	☐ HTS	☐ HPS	☐ BCS	☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Verified at experimental scale	Verified at pilot scale	Verified at engineering scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 6 is achieved upon successful completion of the SCHE design support testing identified in the TRL rating sheet for TRL 5. Achievement of TRL 6 for the SCHE is also dependent on successful completion of the elements of the steam generator design support testing program that is required to satisfy SCHE DDNs C.14.04.02 (vibrational fretting wear and sliding wear of wear protection devices for bare tubes), C14.04.03 (instrument attachment tests), and C14.04.09 (helical coil tube fabrication development).

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Complete final design	GA	2 years	3,000
2. Build a full-size prototype SCHE	SCHE vendor	1 year	6,000
3. Perform flow testing on the full size SCHE to verify thermal/hydraulic and flow induced vibration performance. This test will verify the heat transfer and pressure drop and flow induced vibration characteristics of the tube bundle.	GA/Test Facility	1 year ending 3 years into final design	5,000

DDN(s) Supported: C.14.04.08 **Technology Case File:**

Subject Matter Expert Making Determination: Dave Carosella, Bob Schleicher

Date: 12-09-08 **Originating Organization:** General Atomics

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-8.4	**Revision:** 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Shutdown Cooling Heat Exchanger (SCHE)

Description:
The Shutdown Cooling Heat Exchanger (SCHE) is a multi-tube helical coil heat exchanger. It is similar in design to the evaporator/economizer portion of the FSV steam generator. Its function is to cool the reactor whenever the primary cooling system is not available. It is a vertical cross-counter flow heat exchanger. The tubes are made of 2-1/4 Croloy. The heat is removed by 60^{o}C-pressurized (4.8MPa) water. The SCHE does not have a safety function.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Verified at pilot scale	Verified at engineering scale	Item tested and qualified
TRL	6	7	8

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 7 is achieved upon successful completion of final design, fabrication of a full-size prototype SCHE, and the heat transfer and flow resistance characteristics testing identified in the TRL rating sheet for TRL 6.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Test the SCHE at all possible operating conditions including standby mode and transients. These tests are to be performed at design conditions in a helium environment and will verify the final performance characteristics of the SCHE.	INL CTF	2 years with completion 2 years before NGNP startup testing	5,000

DDN(s) Supported: None	**Technology Case File:**

Subject Matter Expert Making Determination: Dave Carosella, Bob Schleicher

Date: 12-9-08	**Originating Organization:** General Atomics

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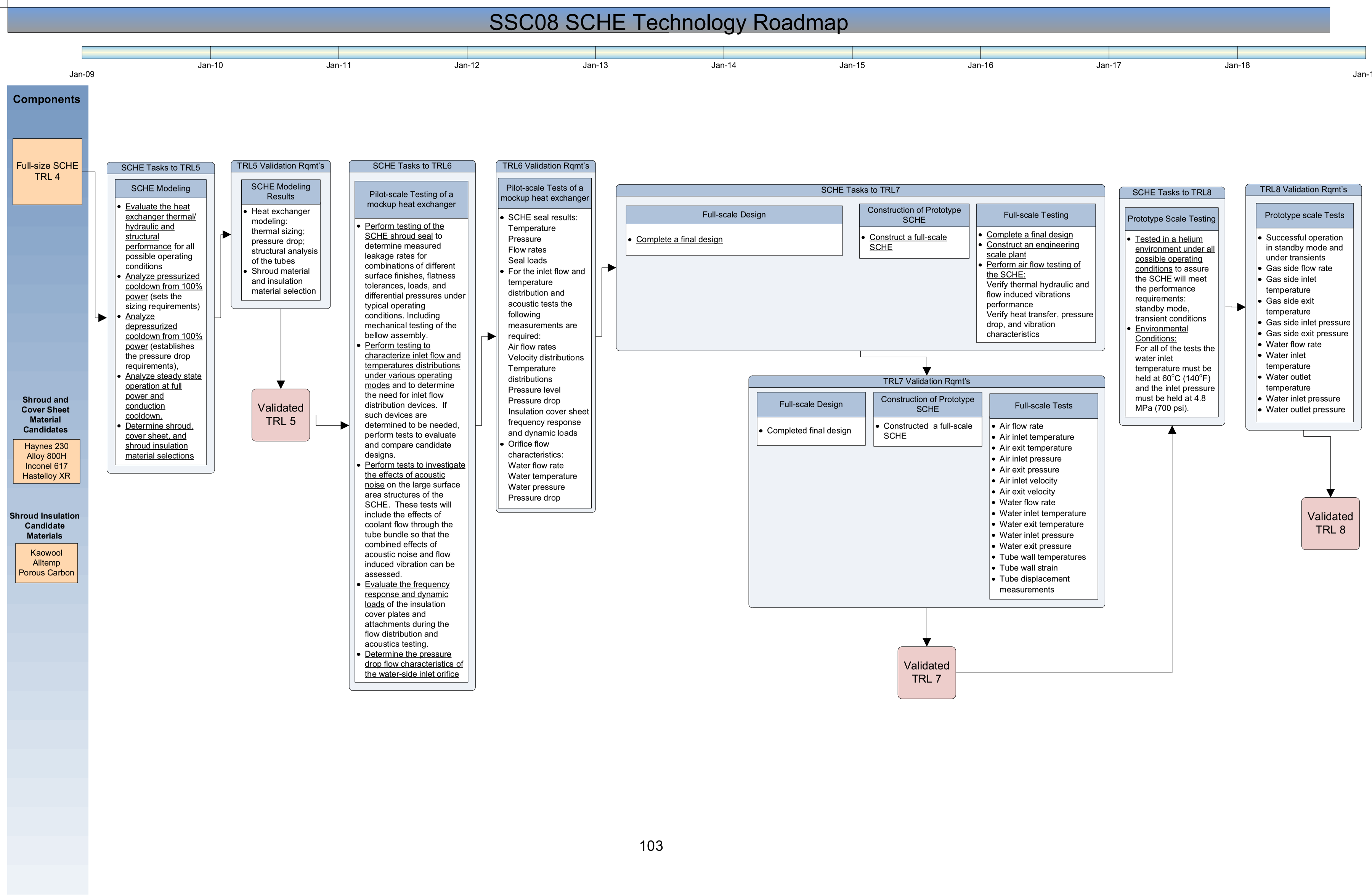
Revision 1



4.9 SSC-9 Reactor Cavity Cooling System

TRL Rating Sheets, TRL 4 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-9.1	**Revision:** 1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Reactor Cavity Cooling System (RCCS)

Description:
The RCCS protects the concrete structure surrounding the reactor vessel from overheating during all modes of operation and provides an alternative means from removing reactor core decay heat when neither the PCS not the SCS is available. The RCCS cooling panels transfer heat from the reactor core to a passive outside air system. The RCCS panels also form a part of the barrier that separates the ambient atmosphere from the reactor cavity atmosphere.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Proof of Concept	Component verified at bench scale	Component verified at experimental scale
TRL	**3**	**4**	**5**

Basis for Rating (Check box if continued on additional sheets) ☐

Sufficient conservative material properties data are available to demonstrate feasibility and to perform the required safety analyses. Natural convection heat transfer, buoyancy-driven flow, friction, and pressure loss are sufficiently understood based on experimental studies of basic phenomena. Expanded data is required to refine the design to include all operational environments anticipated.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Conduct testing to determine the mean and variation of emissivity from one panel to the next. 2. Determine emissivity variation over a large surface. 3. Determine the sensitvity of emissivity to various factors including manufacturing processes, operating service conditions and aging.	Advanced Fuel Research, Inc.	Exp. data 1 yr before start of final design. Overall duration of 15 months.	194

DDN(s) Supported: C.16.00.01	**Technology Case File:**

Subject Matter Expert Making Determination: John Bolin

Date: 12-8-08	**Originating Organization:** General Atomics

TRL Rating Sheet					
Vendor:	GA	**Document Number:**	SSC-9.2	**Revision:**	1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Reactor Cavity Cooling System (RCCS)

Description:
The RCCS protects the concrete structure surrounding the reactor vessel from overheating during all modes of operation and provides an alternative means from removing reactor core decay heat when neither the PCS not the SCS is available. The RCCS I/O structure is an above-grade structure that provides atmospheric air flow to and from the RCCS cooling panels.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	1.4.3	**Parent:**	1.4	**WBS:**	

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Component verified at bench scale	Component verified at experimental scale	Subsystem verified at pilot scale
TRL	**4**	**5**	**6**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 5 is achieved upon successful completion of the RCCS panel emissivity testing called for in the TRL rating sheet for TRL 4.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
The technical feasibility of the I/O structure concept has been demonstrated by a variety of similar applications: however, the RCCS outlet design is unique to MHR. No experimental or wind effect data exists for the configuration expected to be used that for the NGNP. Consequently, it is necessary to perform scale-model testing to determine pressure profiles inside and in the vicinity of I/O structure for various locations of the I/O structure along the length of the nuclear Island and for various wind directions and velocities.	Oran W. Nicks Low Speed Wind Tunnel, Texas A&M	Exp. data 1 yr before start of final design. Overall duration 21 months.	400

DDN(s) Supported: C16.00.02	**Technology Case File:**

Subject Matter Expert Making Determination: John Bolin

Date: 12-8-08	**Originating Organization:** General Atomics

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-9.3	**Revision:** 1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Reactor Cavity Cooling System (RCCS)

Description:
The RCCS protects the concrete structure surrounding the reactor vessel from overheating during all modes of operation and provides an alternative means from removing reactor core decay heat when neither the PCS not the SCS is available. The RCCS cooling panels transfer heat from the reactor core to a passive outside air system. The RCCS panels also form a part of the barrier that separates the ambient atmosphere from the reactor cavity atmosphere.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Component verified at experimental scale	Subsystem verified at pilot scale	System verified at engineering scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 6 is achieved when the subsystem testing called for in the TRL rating sheet for TRL 5 has been successfully completed. The next step is to test the complete system at less-than-full scale.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Perform engineering-scale-model testing to determine overall performance of the RCCS under all expected operating conditions. Determine effect of: 1. Temperature, heat flux, Reynolds number on heat transfer, friction factor data for geometrically similar riser tubes.	NSTF in Bldg 310 at Argonne National Lab	Exp. data before end of first year of final design. Overall duration 24 months.	3,450

DDN(s) Supported: N.16.00.07, C.16.00.03, C.16.00.04	**Technology Case File:**

Subject Matter Expert Making Determination: John Bolin

Date: 12-8-08	**Originating Organization:** General Atomics

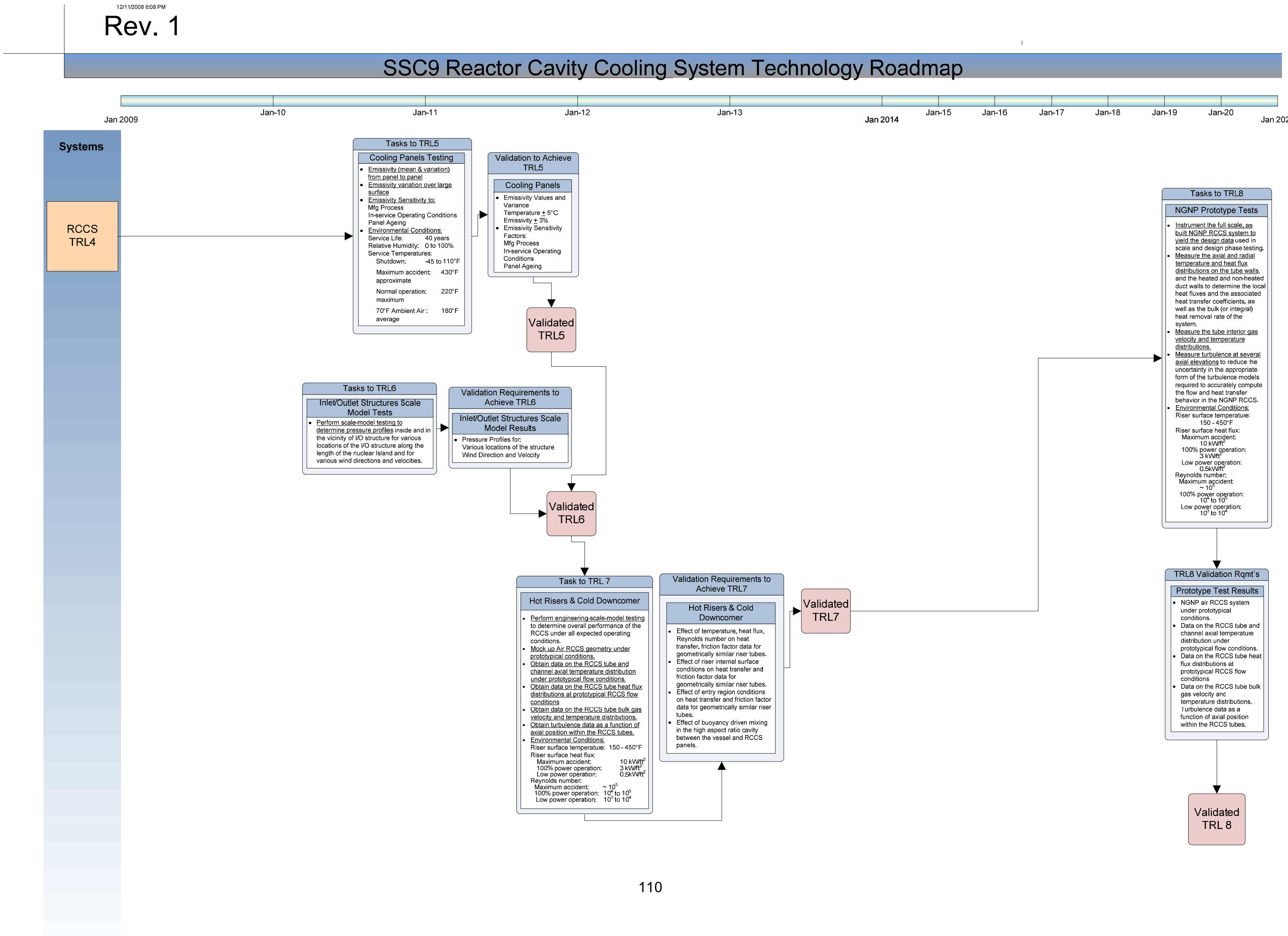
Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Actions: 2. Riser internal surface conditions on heat transfer and friction factor data for geometrically similar riser tubes. 3. Entry region conditions on heat transfer and friction factor data for geometrically similar riser tubes. 4. Buoyancy driven mixing in the high aspect ratio cavity between the vessel and RCCS panels.			

TRL Rating Sheet					
Vendor: GA		**Document Number:** SSC-9.4		**Revision:** 1	
☐ Area	☒ System	☐ Subsystem/Structure	☐ Component	☐ Technology	
Title: Reactor Cavity Cooling System (RCCS)					
Description: The RCCS protects the concrete structure surrounding the reactor vessel from overheating during all modes of operation and provides an alternative means from removing reactor core decay heat when neither the PCS not the SCS is available. The RCCS cooling panels transfer heat from the reactor core to a passive outside air system. The RCCS panels also form a part of the barrier that separates the ambient atmosphere from the reactor cavity atmosphere.					
Area:	☒ NHSS	☐ HTS	☐ HPS	☐ BCS	☐ BOP
ASSCT:		**Parent:**		**WBS:**	

Technology Readiness Level			
	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Subsystem verified at pilot scale	System verified at engineering scale	System tested and qualified
TRL	**6**	**7**	**8**
Basis for Rating (Check box if continued on additional sheets) ☐			
TRL 7 is achieved when RCCS system performance has been demonstrated at less-than-full scale in a relevant environment as called for in the TRL rating sheet for TRL 6 (see document SSC-9.3). The RCCS will achieve TRL of 8 by virtue of pre-commissioning testing at NGNP.			
Outline of plan to get from current level to next level. (Check box if continued on additional sheets) ☐			

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Perform testing of NGNP RCCS to verify design under all expected operating conditions.	As-built NGNP RCCS	Part of startup testing. Overall duration 24 months.	Part of startup testing.

DDN(s) Supported: None	**Technology Case File:**
Subject Matter Expert Making Determination: John Bolin	
Date: 12-8-08	**Originating Organization:** General Atomics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Rev. 1
SSC9 Reactor Cavity Cooling System Technology Roadmap
Jan 2009
Jan-10
Jan-11
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Jan-13
Jan 2014
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Jan-17
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Jan-19
Jan-20
Jan 2021
Systems
RCCS TRL4
Tasks to TRL5
Cooling Panels Testing
Emissivity (mean & variation) from panel to panel
Emissivity variation over large surface
Emissivity Sensitivity to: Mfg Process, In-service Operating Conditions, Panel Ageing
Environmental Conditions: Service Life: 40 years; Relative Humidity: 0 to 100%; Service Temperatures: Shutdown: -45 to 110°F; Maximum accident: 430°F approximate; Normal operation: 220°F maximum; 70°F Ambient Air: 180°F average
Validation to Achieve TRL5
Cooling Panels
Emissivity Values and Variance: Temperature ± 5°C, Emissivity ± 3%
Emissivity Sensitivity Factors: Mfg Process, In-service Operating Conditions, Panel Ageing
Validated TRL5
Tasks to TRL6
Inlet/Outlet Structures Scale Model Tests
Perform scale-model testing to determine pressure profiles inside and in the vicinity of I/O structure for various locations of the I/O structure along the length of the nuclear Island and for various wind directions and velocities.
Validation Requirements to Achieve TRL6
Inlet/Outlet Structures Scale Model Results
Pressure Profiles for: Various locations of the structure, Wind Direction and Velocity
Validated TRL6
Task to TRL 7
Hot Risers & Cold Downcomer
Perform engineering-scale-model testing to determine overall performance of the RCCS under all expected operating conditions.
Mock up Air RCCS geometry under prototypical conditions.
Obtain data on the RCCS tube and channel axial temperature distribution under prototypical flow conditions.
Obtain data on the RCCS tube heat flux distributions at prototypical RCCS flow conditions
Obtain data on the RCCS tube bulk gas velocity and temperature distributions.
Obtain turbulence data as a function of axial position within the RCCS tubes.
Environmental Conditions: Riser surface temperature: 150 - 450°F; Riser surface heat flux: Maximum accident: 10 kW/ft²; 100% power operation: 3 kW/ft²; Low power operation: 0.5kW/ft²; Reynolds number: Maximum accident: ~ 10⁵; 100% power operation: 10⁴ to 10⁵; Low power operation: 10³ to 10⁴
Validation Requirements to Achieve TRL7
Hot Risers & Cold Downcomer
Effect of temperature, heat flux, Reynolds number on heat transfer, friction factor data for geometrically similar riser tubes.
Effect of riser internal surface conditions on heat transfer and friction factor data for geometrically similar riser tubes.
Effect of entry region conditions on heat transfer and friction factor data for geometrically similar riser tubes.
Effect of buoyancy driven mixing in the high aspect ratio cavity between the vessel and RCCS panels.
Validated TRL7
Tasks to TRL8
NGNP Prototype Tests
Instrument the full scale, as built NGNP RCCS system to yield the design data used in scale and design phase testing.
Measure the axial and radial temperature and heat flux distributions on the tube walls, and the heated and non-heated duct walls to determine the local heat fluxes and the associated heat transfer coefficients, as well as the bulk (or integral) heat removal rate of the system.
Measure the tube interior gas velocity and temperature distributions.
Measure turbulence at several axial elevations to reduce the uncertainty in the appropriate form of the turbulence models required to accurately compute the flow and heat transfer behavior in the NGNP RCCS.
Environmental Conditions: Riser surface temperature: 150 - 450°F; Riser surface heat flux: Maximum accident: 10 kW/ft²; 100% power operation: 3 kW/ft²; Low power operation: 0.5kW/ft²; Reynolds number: Maximum accident: ~ 10⁵; 100% power operation: 10⁴ to 10⁵; Low power operation: 10³ to 10⁴
TRL8 Validation Rqmt's
Prototype Test Results
NGNP air RCCS system under prototypical conditions.
Data on the RCCS tube and channel axial temperature distribution under prototypical flow conditions.
Data on the RCCS tube heat flux distributions at prototypical RCCS flow conditions
Data on the RCCS tube bulk gas velocity and temperature distributions. Turbulence data as a function of axial position within the RCCS tubes.
Validated TRL 8

4.10 SSC-10 Steam Generator

Steam Generator, 750°C Inlet Temperature, TRL Rating Sheets TRL 4 through 7

Steam Generator, 950°C Inlet Temperature, TRL Rating Sheets, TRL 3 and 4

Technology Development Road Map

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-10.1.1	**Revision:** 1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Steam Generator – 750°C Gas Inlet Temperature

Description:
The steam generator (SG) is a multi-tube, helical coil, cross-counter flow helium-to-water/steam heat exchanger similar in design to the FSV SG. In this SG, high-temperature helium @7MPa pressure heats feedwater @ 200°C & 19.5 MPa to 540°C steam @ 17.3MPa. The SG comprises two sections: the lower-temperature Economizer/Evaporator/ Superheater (EES) and the higher-temperature Finishing Superheater. The two sections are connected with a vertical tube section that is about 1 meter long. This vertical section, which is exterior to the main helium flow path, contains a bimetallic weld. A detailed description of the SG can be found in GA document 911120 “NGNP Steam Generator Alternative Study” and in GA Test Plan 911142.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Proof of Concept	Demonstrated at bench scale	Demonstrated at experimental scale
TRL	3	**4**	**5**

Basis for Rating (Check box if continued on additional sheets) ☐

GA has assigned a TRL of 4 to the 750°C SG primarily because (1) the FSV reactor experience has demonstrated the basic helical-coil SG thermal and hydraulic design and the SG material selections, and (2) the considerable level of SG design definition already available from the MHTGR Program. The validation of the helium-side heat transfer coefficients is documented in ASME paper 79-WA/NE-1. The FSV SG, although smaller then the NGNP SG was of the same basic configuration. The THTR heat exchanger was of similar design.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Perform SG conceptual design and analysis. Use computer models to 1) Size the SG for design operating conditions @ 100% heat load, 2) Determine the steady state pressure drop, and 3) Perform structural analyses of the various SG components including the tubes and tube supports. Define the DDNs for the NGNP SG and prepare a design support program plan that outlines the testing required to satisfy the DDNs.	GA	About 1.5 years with completion by end of CD	3000

DDN(s) Supported: None	**Technology Case File:**

Subject Matter Expert Making Determination: Dave Carosella

Date: 12-10-08	**Originating Organization:** General Atomics

<table>
<tr><td colspan="4">TRL Rating Sheet</td></tr>
<tr><td>Vendor: GA</td><td colspan="2">Document Number: SSC-10.1.2</td><td>Revision: 1</td></tr>
<tr><td colspan="4">☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology</td></tr>
<tr><td colspan="4">Title: Steam Generator – 750°C and 950°C Gas Inlet Temperature</td></tr>
<tr><td colspan="4">Description:
The steam generator (SG) is a multi-tube, helical coil, cross-counter flow helium-to-water/steam heat exchanger similar in design to the FSV SG. In this SG, high-temperature helium @7MPa pressure heats feedwater @ 200°C & 19.5 MPa to 540°C steam @ 17.3MPa. The SG comprises two sections: the lower-temperature Economizer/Evaporator/ Superheater (EES) and the higher-temperature Finishing Superheater. The two sections are connected with a vertical tube section that is about 1 meter long. This vertical section, which is exterior to the main helium flow path, contains a bimetallic weld. A detailed description of the SG can be found in GA document 911120 “NGNP Steam Generator Alternative Study” and in GA Test Plan 911142.</td></tr>
<tr><td>Area:</td><td colspan="3">☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP</td></tr>
<tr><td colspan="2">ASSCT:</td><td>Parent:</td><td>WBS:</td></tr>
<tr><td colspan="4">Technology Readiness Level</td></tr>
<tr><td></td><td>Next Lower Rating Level</td><td>Current Rating Level</td><td>Next Higher Rating Level</td></tr>
<tr><td>Generic Definitions (abbreviated)</td><td>Demonstrated at bench scale</td><td>Demonstration at experimental scale</td><td>Demonstrated at pilot scale</td></tr>
<tr><td>TRL</td><td>4</td><td>5</td><td>6</td></tr>
<tr><td colspan="4">Basis for Rating (Check box if continued on additional sheets) ☐</td></tr>
<tr><td colspan="4">TRL 5 is achieved when the design and analysis activities defined in the TRL rating sheet for TRL 4 have been successfully completed. Specifically, the conceptual design of the NGNP has been developed, including completion of analyses to size the SG, calculate the pressure drop, and verify the structural design. Additionally, the DDNs for the NGNP SG have been defined and a design support program plan that outlines the testing required to satisfy the DDNs has been prepared.</td></tr>
<tr><td colspan="4">Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒</td></tr>
</table>

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1) Demonstrate the ability to fabricate the helical coiled tubes.	GA/Vendor	1 year starting last year of PD	1,250

DDN(s) Supported: M.13.02.01, M.13.02.02, M.13.02.03, M.13.02.04, M.13.02.07, M.13.02.08, M.13.02.10, M.13.02.11, M.13.02.12, M.13.02.14, M.13.02.15	**Technology Case File:**
Subject Matter Expert Making Determination: Dave Carosella	
Date: 12-14-08	**Originating Organization:** General Atomics

Additional Action Sheet(s)

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Actions:			
2) Perform mockup tests to establish lead-in lead-out and transition lead expansion and assembly room.	GA/Vendor	1 year Starting 2nd year of PD	600
3) Select and design the SG tube support system and ware protection devices. Perform testing to assure no unacceptable tube ware due to vibration and/or fretting.	GA/Vendor	1 year Starting 2nd year of PD	2,450
4) Perform testing to assure the ability to assemble the SG bundle using the support system.	GA/Vendor		
5) Perform testing to verify the mechanical and thermal characteristics of the vessel insulation and its cover sheet. Tests must be performed at NGNP design operating conditions.	GA/Vendor	1 Year starting last year of PD	700
6) Perform air-flow testing to determine if vortex shedding and flow separation caused by the tube bundle will damage insulation cover sheets and/or flow shrouds. If damage occurs it will be necessary to design and test protection methods for the cover sheets and/or flow shrouds.	GA/Vendor		780
7) Perform testing to verify the mechanical properties of alloy 800H under NGNP design conditions	GA/Test Lab	3 Years starting at start of CD	15,000
8) Perform testing to verify the mechanical properties of alloy 2¼Cr - 1Mo under NGNP design conditions			
9) Perform testing to verify the mechanical properties of the bimetallic weld under NGNP design conditions.			
10) Perform an airflow test of the steam generator inlet region to determine extent of flow maldistribution. b) Design and test flow control device to eliminate gas side flow maldistribution.	GA/Test Lab	1 year Starting 2nd Year of PD	$1,090
11) Perform flow testing to verify the flow/pressure drop characteristics of the orifice on the secondary side			

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-10.1.3	**Revision:** 1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Steam Generator – 750°C and 950°C Gas Inlet Temperature

Description:
The steam generator (SG) is a multi-tube, helical coil, cross-counter flow helium-to-water/steam heat exchanger similar in design to the FSV SG. In this SG, high-temperature helium @7MPa pressure heats feedwater @ 200°C & 19.5 MPa to 540°C steam @ 17.3MPa. The SG comprises two sections: the lower-temperature Economizer/Evaporator/ Superheater (EES) and the higher-temperature Finishing Superheater. The two sections are connected with a vertical tube section that is about 1 meter long. This vertical section, which is exterior to the main helium flow path, contains a bimetallic weld. A detailed description of the SG can be found in GA document 911120 "NGNP Steam Generator Alternative Study" and in GA Test Plan 911142.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Demonstrated at experimental scale	Demonstrated at pilot scale	Demonstrated at engineering scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☒

TRL 6 is achieved upon successful completion of the required design support testing defined in the TRL rating sheet for TRL 5.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Complete final design and fabricate full-size prototype Steam Generator	GA/SG vendor	4 years starting at beginning of FD	TBD
2. Perform flow testing of full-size prototype SG to verify the heat transfer, pressure drop and vibration characteristics of the SG.	GA/SG vendor	2 years with completion one year before NGNP startup	6,000

DDN(s) Supported: None	**Technology Case File:**

Subject Matter Expert Making Determination: Dave Carosella

Date: 12-14-08	**Originating Organization:** General Atomics

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-10.1.4	**Revision:** 1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Steam Generator – 750°C and 950°C Gas Inlet Temperature

Description:
The steam generator (SG) is a multi-tube, helical coil, cross-counter flow helium-to-water/steam heat exchanger similar in design to the FSV SG. In this SG, high-temperature helium @7MPa pressure heats feedwater @ 200°C & 19.5 MPa to 540°C steam @ 17.3MPa. The SG comprises two sections: the lower-temperature Economizer/Evaporator/ Superheater (EES) and the higher-temperature Finishing Superheater. The two sections are connected with a vertical tube section that is about 1 meter long. This vertical section, which is exterior to the main helium flow path, contains a bimetallic weld. A detailed description of the SG can be found in GA document 911120 "NGNP Steam Generator Alternative Study" and in GA Test Plan 911142.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Demonstrated at pilot scale	Demonstrated at engineering scale	System tested and qualified
TRL	**6**	**7**	**8**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 7 is achieved upon successful completion of the air-flow testing of a full size SG to verify the heat transfer, pressure drop and vibration characteristics of the steam SG (as defined in the TRL rating sheet for TRL 6.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Test the steam generator thermal/hydraulic characteristics in the NGNP helium environment under design conditions including steady state and transient operating conditions.	GA/NGNP operator	1.5 years	TBD

DDN(s) Supported: None	**Technology Case File:**

Subject Matter Expert Making Determination: Dave Carosella

Date: 12-14-08	**Originating Organization:** General Atomics

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-10.2.1	**Revision:** 1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Steam Generator – 950°C Gas Inlet Temperature

Description:
The steam generator (SG) is a multi-tube, helical coil, cross-counter flow helium-to-water/steam heat exchanger similar in design to the FSV SG. In this SG, high-temperature helium @7MPa pressure heats feedwater @ 200^{o}C & 19.5 MPa to 540^{o}C steam @ 17.3MPa. The SG comprises two sections: the lower-temperature Economizer/Evaporator/ Superheater (EES) and the higher-temperature Finishing Superheater. The two sections are connected with a vertical tube section that is about 1 meter long. This vertical section, which is exterior to the main helium flow path, contains a bimetallic weld. A detailed description of the SG can be found in GA document 911120 "NGNP Steam Generator Alternative Study" and in GA Test Plan 911142.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ PCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Application Formulated	Proof of Concept	Demonstrated at bench-scale
TRL	**2**	**3**	**4**

Basis for Rating (Check box if continued on additional sheets) ☐

The FSV reactor experience has shown the validity of the helical coil steam generator thermal and hydraulic design. The validation of the helium side heat transfer coefficients is documented in ASME paper 79-WA/NE-1. This heat exchanger, although smaller than the NGNP heat exchanger was of the same basic configuration. The THTR heat exchanger was of similar configuration. However, a TRL of 3 was assigned for an SG designed to operate with an inlet helium temperature of 950°C because of the need to qualify and use higher-temperature materials than used in past GA SG designs, both in the Finishing Superheater section and the Economizer/Evaporator/ Superheater section.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
For an SG operating with an inlet helium temperature of 950°C, high-temperature alloys must be considered for the SG finishing superheater section. These alloys include Inconel 617, Haynes 230, and Hastelloy XR, which are the same materials being considered for the NGNP IHX. (Cont.)	INL/ORNL	Material selection required early in preliminary design	Cost covered by IHX materials R&D Program (see below)

DDN(s) Supported: New DDN to be defined	**Technology Case File:**

Subject Matter Expert Making Determination: Dave Carosella

Date: 12-8-08	**Originating Organization:** General Atomics

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Actions: Further, higher-temperature materials such as 9Cr - 1Mo, Alloy 800H, or Alloy 800HT might be needed to replace 2¼Cr - 1Mo in the economizer, evaporator, and superheater (EES) section. The SG materials will be selected based on the following criteria: a) The material must have adequate strength and life expectancy operating under the design conditions, b) The material composition must not be a source of cobalt contamination in the primary circuit, c) The bimetallic weld characteristics where the Finishing Superheater section is connected to the EES bundle must not be a problem. An R&D program to develop high-temperature materials for the NGNP has been defined, is currently in progress, and is focused primarily on Alloy 617. Recently, an IHX Materials Research and Development Plan (INL PLN-2804) was issued. This plan outlines an extensive R&D program that is designed to acquire the data needed for an ASME code case for Alloy 617. Although this program is specific to the IHX, the data generated by this program, and the resultant ASME code case should also support the use of Alloy 617 in the finishing superheater section of the SG.		Schedule to complete the R&D program outlined in INL PLN-2804 is currently undefined	Total cost of testing in INL PLN-2804 is about $8.2M.

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-10.2.2	**Revision:**	1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Steam Generator – 950C Gas Inlet Temperature

Description:
The steam generator (SG) is a multi-tube, helical coil, cross-counter flow helium-to-water/steam heat exchanger similar in design to the FSV SG. In this SG, high-temperature helium @7MPa pressure heats feedwater @ 200°C & 19.5 MPa to 540°C steam @ 17.3MPa. The SG comprises two sections: the lower-temperature Economizer/Evaporator/ Superheater (EES) and the higher-temperature Finishing Superheater. The two sections are connected with a vertical tube section that is about 1 meter long. This vertical section, which is exterior to the main helium flow path, contains a bimetallic weld. A detailed description of the SG can be found in GA document 911120 "NGNP Steam Generator Alternative Study" and in GA Test Plan 911142.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Proof of Concept	Demonstrated at bench scale	Demonstrated at experimental scale
TRL	**3**	**4**	**5**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 4 is achieved upon completion of the actions identified in the TRL rating sheet for TRL 3. Specifically, sufficient testing of candidate materials for the 950C SG has been performed to provide the data necessary to allow for selection of materials that have a high probability of meeting SG design requirements.

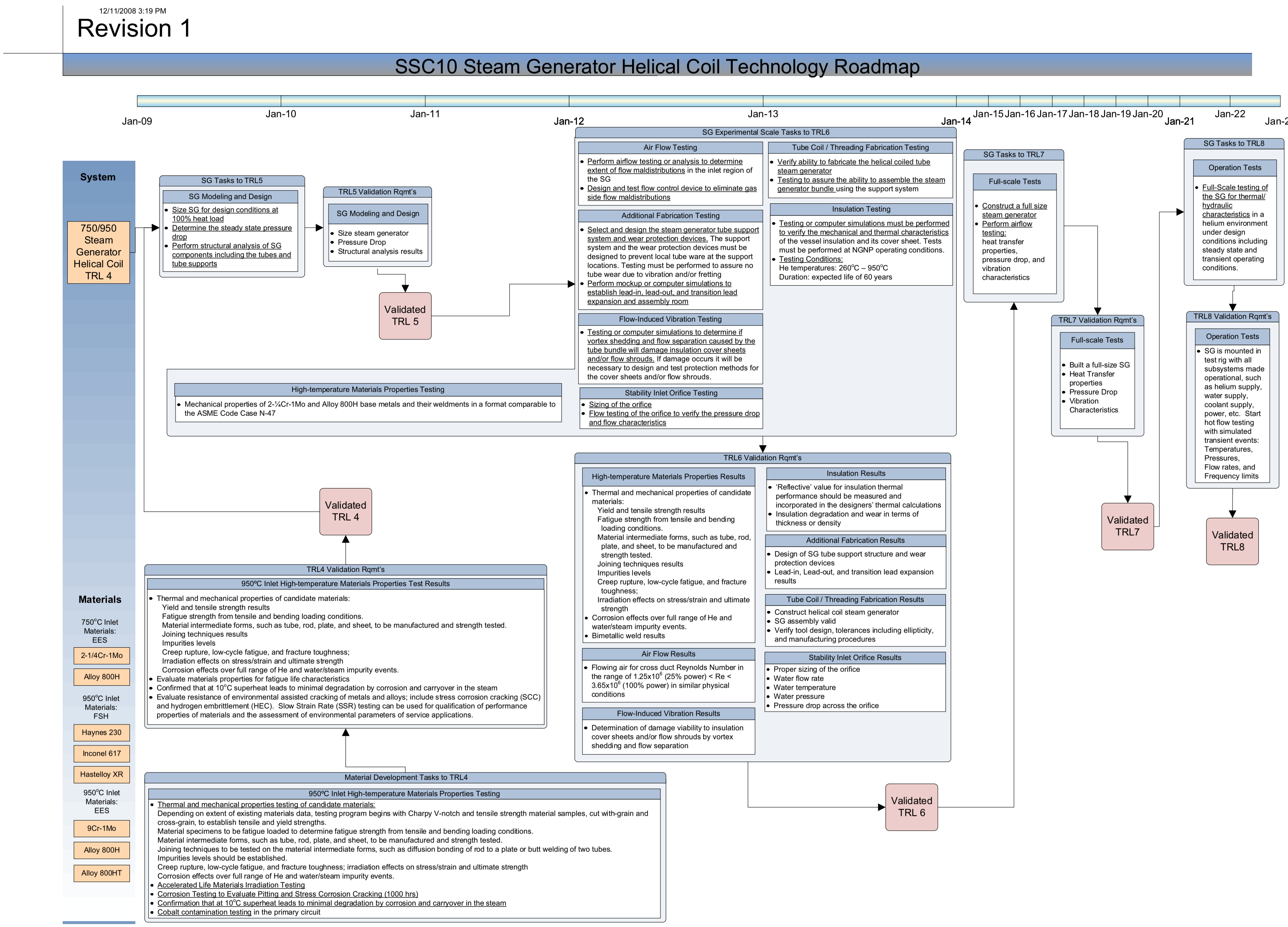
Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Perform SG conceptual design and analysis. Use computer models to 1) Size the SG for design operating conditions @ 100% heat load, 2) Determine the steady state pressure drop, and 3) Perform structural analyses of the various SG components including the tubes and tube supports. Define the DDNs for the NGNP SG and prepare a design support program plan that outlines the testing required to satisfy the DDNs.	GA	About 1.5 years with completion by end of CD	3,000

DDN(s) Supported: None | **Technology Case File:**

Subject Matter Expert Making Determination: Dave Carosella, John Saurwein

Date: 12-8-08 | **Originating Organization:** General Atomics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SSC10 Steam Generator Helical Coil Technology Roadmap
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System
750/950 Steam Generator Helical Coil TRL 4
Materials
750°C Inlet Materials: EES
2-1/4Cr-1Mo
Alloy 800H
950°C Inlet Materials: FSH
Haynes 230
Inconel 617
Hastelloy XR
950°C Inlet Materials: EES
9Cr-1Mo
Alloy 800H
Alloy 800HT
SG Tasks to TRL5
SG Modeling and Design
• Size SG for design conditions at 100% heat load
• Determine the steady state pressure drop
• Perform structural analysis of SG components including the tubes and tube supports
TRL5 Validation Rqmt's
SG Modeling and Design
• Size steam generator
• Pressure Drop
• Structural analysis results
Validated TRL 5
High-temperature Materials Properties Testing
• Mechanical properties of 2-¼Cr-1Mo and Alloy 800H base metals and their weldments in a format comparable to the ASME Code Case N-47
Validated TRL 4
TRL4 Validation Rqmt's
950°C Inlet High-temperature Materials Properties Test Results
• Thermal and mechanical properties of candidate materials:
Yield and tensile strength results
Fatigue strength from tensile and bending loading conditions.
Material intermediate forms, such as tube, rod, plate, and sheet, to be manufactured and strength tested.
Joining techniques results
Impurities levels
Creep rupture, low-cycle fatigue, and fracture toughness;
Irradiation effects on stress/strain and ultimate strength
Corrosion effects over full range of He and water/steam impurity events.
• Evaluate materials properties for fatigue life characteristics
• Confirmed that at 10°C superheat leads to minimal degradation by corrosion and carryover in the steam
• Evaluate resistance of environmental assisted cracking of metals and alloys; include stress corrosion cracking (SCC) and hydrogen embrittlement (HEC). Slow Strain Rate (SSR) testing can be used for qualification of performance properties of materials and the assessment of environmental parameters of service applications.
Material Development Tasks to TRL4
950°C Inlet High-temperature Materials Properties Testing
• Thermal and mechanical properties testing of candidate materials:
Depending on extent of existing materials data, testing program begins with Charpy V-notch and tensile strength material samples, cut with-grain and cross-grain, to establish tensile and yield strengths.
Material specimens to be fatigue loaded to determine fatigue strength from tensile and bending loading conditions.
Material intermediate forms, such as tube, rod, plate, and sheet, to be manufactured and strength tested.
Joining techniques to be tested on the material intermediate forms, such as diffusion bonding of rod to a plate or butt welding of two tubes.
Impurities levels should be established.
Creep rupture, low-cycle fatigue, and fracture toughness; irradiation effects on stress/strain and ultimate strength
Corrosion effects over full range of He and water/steam impurity events.
• Accelerated Life Materials Irradiation Testing
• Corrosion Testing to Evaluate Pitting and Stress Corrosion Cracking (1000 hrs)
• Confirmation that at 10°C superheat leads to minimal degradation by corrosion and carryover in the steam
• Cobalt contamination testing in the primary circuit
SG Experimental Scale Tasks to TRL6
Air Flow Testing
• Perform airflow testing or analysis to determine extent of flow maldistributions in the inlet region of the SG
• Design and test flow control device to eliminate gas side flow maldistributions
Tube Coil / Threading Fabrication Testing
• Verify ability to fabricate the helical coiled tube steam generator
• Testing to assure the ability to assemble the steam generator bundle using the support system
Additional Fabrication Testing
• Select and design the steam generator tube support system and wear protection devices. The support system and the wear protection devices must be designed to prevent local tube ware at the support locations. Testing must be performed to assure no tube wear due to vibration and/or fretting
• Perform mockup or computer simulations to establish lead-in, lead-out, and transition lead expansion and assembly room
Insulation Testing
• Testing or computer simulations must be performed to verify the mechanical and thermal characteristics of the vessel insulation and its cover sheet. Tests must be performed at NGNP operating conditions.
• Testing Conditions:
He temperatures: 260°C – 950°C
Duration: expected life of 60 years
Flow-Induced Vibration Testing
• Testing or computer simulations to determine if vortex shedding and flow separation caused by the tube bundle will damage insulation cover sheets and/or flow shrouds. If damage occurs it will be necessary to design and test protection methods for the cover sheets and/or flow shrouds.
Stability Inlet Orifice Testing
• Sizing of the orifice
• Flow testing of the orifice to verify the pressure drop and flow characteristics
TRL6 Validation Rqmt's
High-temperature Materials Properties Results
• Thermal and mechanical properties of candidate materials:
Yield and tensile strength results
Fatigue strength from tensile and bending loading conditions.
Material intermediate forms, such as tube, rod, plate, and sheet, to be manufactured and strength tested.
Joining techniques results
Impurities levels
Creep rupture, low-cycle fatigue, and fracture toughness;
Irradiation effects on stress/strain and ultimate strength
• Corrosion effects over full range of He and water/steam impurity events.
• Bimetallic weld results
Insulation Results
• 'Reflective' value for insulation thermal performance should be measured and incorporated in the designers' thermal calculations
• Insulation degradation and wear in terms of thickness or density
Additional Fabrication Results
• Design of SG tube support structure and wear protection devices
• Lead-in, Lead-out, and transition lead expansion results
Tube Coil / Threading Fabrication Results
• Construct helical coil steam generator
• SG assembly valid
• Verify tool design, tolerances including ellipticity, and manufacturing procedures
Air Flow Results
• Flowing air for cross duct Reynolds Number in the range of 1.25x10^6 (25% power) < Re < 3.65x10^6 (100% power) in similar physical conditions
Stability Inlet Orifice Results
• Proper sizing of the orifice
• Water flow rate
• Water temperature
• Water pressure
• Pressure drop across the orifice
Flow-Induced Vibration Results
• Determination of damage viability to insulation cover sheets and/or flow shrouds by vortex shedding and flow separation
Validated TRL 6
SG Tasks to TRL7
Full-scale Tests
• Construct a full size steam generator
• Perform airflow testing:
heat transfer properties, pressure drop, and vibration characteristics
TRL7 Validation Rqmt's
Full-scale Tests
• Built a full-size SG
• Heat Transfer properties
• Pressure Drop
• Vibration Characteristics
Validated TRL7
SG Tasks to TRL8
Operation Tests
• Full-Scale testing of the SG for thermal/ hydraulic characteristics in a helium environment under design conditions including steady state and transient operating conditions.
TRL8 Validation Rqmt's
Operation Tests
• SG is mounted in test rig with all subsystems made operational, such as helium supply, water supply, coolant supply, power, etc. Start hot flow testing with simulated transient events: Temperatures, Pressures, Flow rates, and Frequency limits
Validated TRL8

4.11 SSC-11 PCS Turbomachinery

TRL Rating Sheets, TRL 4 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-11.1	**Revision:**	1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Power Conversion System (PCS)

Description:
The combined gas and steam cycle consists of a 66MWt gas turbine generator with the remainder of the power driving the steam cycle. The key features of this concept relative to the GT-MHR PCS design are: (1) the recuperator is no longer required (a steam generator would be required, but this is considered much lower risk), (2) electromagnetic bearing risks are reduced by reducing generator weight from 35t to around 10t, and turbomachinery shaft weight from 32t to around 10t, (3) power electronics costs are reduced (since generator is reduced from 300MW to 66MW in gas turbine section), (4) plant efficiency is increased compared to the GT-MHR Brayton cycle, (5) steam turbine and steam cycle electrical generator are commercial off-the-shelf items - low cost and low risk.

Area: ☐ NHSS ☐ HTS ☐ HPS ☒ PCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Proof of Concept	Components Verified at Bench Scale	Subsystem Verified at Experimental Scale
TRL	**3**	**4**	**5**

Basis for Rating (Check box if continued on additional sheets) ☐

The requirements to achieve TRL 4 have been met. The materials and designs chosen for the combined-cycle steam generator, turbo compressor, generator, and various seals/couplers have been modeled using engineering analysis software to demonstrate technical feasibility and functionality. The PCS component designs have been demonstrated with similar designs already in-service. Materials data has been referenced during the design process for experience high temperatures (<850ºC) and high pressure helium (7,020 kPa; 1,020 psi).

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Testing of various subcomponents that comprise the PCS all need to advance to achieve the next TRL. Additional sheet provides more detail.	See Additional Action Sheet	2009-2014	31,000

DDN(s) Supported: GT-MHR Russian Program PCU TDPP document and DDN C.41.00.01	**Technology Case File:**

Subject Matter Expert Making Determination: A. Bozek

Date: 12/11/2008	**Originating Organization:** General Atomics

Additional Action Sheet(s)

Achieve successful test results for each of the following:

GENERATOR:
- test winding insulation samples to validate dielectric characteristics at various temperatures;
- test electrical lead-outs to verify mechanical strength, leak-tightness, insulation resistance, and electric strength of insulation.

TURBINE:
The principal focus is upon the high-temperature turbine region of the turbo compressor, including disks, blades, stator vanes, volute, stator casing and fastening parts:
- test turbine stage aerodynamic performance including clearances required (minimum clearance determined by the clearance between the rotor and catcher bearings).
- test rotating seal performance, including electromagnetic bearings and catcher bearings, which maintain high efficiency and isolate the primary helium circuit from the generator enclosure.
- test electromagnetic bearing performance and verify against rotor dynamics analysis and system control software.
- test catcher bearing friction performance to verify friction material performance

COMPRESSOR:
- verify in testing that titanium and steel materials chosen for compressor components do not suffer extensive embrittlement in helium environment
- verify in testing that no self-welding of materials occurs in helium environment.

Possible Actionees:
<u>Generator Winding Insulation and Electrical Lead-Out Testing</u>: Northrop Grumman; REMEC; NTS
<u>Turbine Aerodynamics Performance Testing</u>: Siemens; General Electric; He Test Facility at PBMR
<u>Turbocompressor Rotating Seals Testing</u>: OKBM; Timken Bearing
<u>Turbocompressor Bearings Testing</u>: OKBM; S2M; SKF; Waukesha; Synchrony

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-11.2	**Revision:** 1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Power Conversion System (PCS)

Description:

The combined gas and steam cycle consists of a 66MWt gas turbine generator with the remainder of the power driving the steam cycle. The key features of this concept relative to the GT-MHR PCS design are: (1) the recuperator is no longer required (a steam generator would be required, but this is considered much lower risk), (2) electromagnetic bearing risks are reduced by reducing generator weight from 35t to around 10t, and turbomachinery shaft weight from 32t to around 10t, (3) power electronics costs are reduced (since generator is reduced from 300MW to 66MW in gas turbine section), (4) plant efficiency is increased compared to the GT-MHR Brayton cycle, (5) steam turbine and steam cycle electrical generator are commercial off-the-shelf items - low cost and low risk.

Area: ☐ NHSS ☐ HTS ☐ HPS ☒ PCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Components Verified at Bench Scale	Subsystem Verified at Experimental Scale	Subsystem Verified at Pilot Scale
TRL	**4**	**5**	**6**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 5 is achieved upon successful completion of the testing activities outlined in the TRL rating sheet for TRL 4. Specifically, (1) the generator winding insulation and electrical lead tests have been successfully performed, (2) the design of the turbine's components (blades, etc) has been tested for aerodynamic performance, (3) seal and bearing tests have been completed successfully, and (4) it has been verified that compressor materials will not suffer excessive embrittlement in the helium environment.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Components that comprise the PCS will be fabricated and integrated for testing in operating temperatures and helium pressures to achieve TRL 6. Additional sheet provides more detail.	See Additional Action Sheet	2009-2014	26,000

DDN(s) Supported: GT-MHR Russian Program PCU TDPP document and DDN C.41.00.01	**Technology Case File:**
Subject Matter Expert Making Determination: A. Bozek	
Date: 12/11/2008	**Originating Organization:** General Atomics

Additional Action Sheet(s)

Achieve successful test results for each of the following:

GENERATOR:
- test winding insulation on powered generator components to validate dielectric characteristics at operating temperatures and pressures in a helium environment;
- test electrical lead-outs to verify mechanical strength, leak-tightness, insulation resistance, and electric strength of insulation in a helium environment at operating temperatures and helium pressures.

TURBINE:
The principal focus is upon the high-temperature turbine region of the turbocompressor, including disks, blades, stator vanes, volute, stator casing and fastening parts:
- test turbine stage aerodynamic performance including clearances required (minimum clearance determined by the clearance between the rotor and catcher bearings) at operating temperatures and helium pressures.
- test rotating seal performance, including electromagnetic bearings and catcher bearings, which maintain high efficiency and isolate the primary helium circuit from the generator enclosure, at operating temperatures and helium pressures.
- test electromagnetic bearing performance and verify against rotor dynamics analysis and system control software at operating temperatures (assuming purged bearings).
- test catcher bearing friction performance to verify friction material performance at operating temperatures and helium pressures.

COMPRESSOR:
- test compressor fabricated sections in operating temperatures and pressures to verify operation at below optimum speeds
- test compressor components to verify non-excessive acoustic loads

Possible Actionees:
Generator Winding Insulation and Electrical Lead-Out Testing: Northrop Grumman; REMEC; NTS
Turbine Aerodynamics Performance Testing: Siemens; General Electric; He Test Facility at PBMR
Turbocompressor Rotating Seals Testing: OKBM; Timken Bearing
Turbocompressor Bearings Testing: OKBM; S2M; SKF; Waukesha; Synchrony

TRL Rating Sheet

Vendor: GA		**Document Number:** SSC-11.3		**Revision:** 1
☐ Area	☒ System	☐ Subsystem/Structure	☐ Component	☐ Technology

Title: Power Conversion System (PCS)

Description:
The combined gas and steam cycle consists of a 66MWt gas turbine generator with the remainder of the power driving the steam cycle. The key features of this concept relative to the GT-MHR design are: (1) the recuperator is no longer required (a steam generator would be required, but this is considered much lower risk), (2) electromagnetic bearing risks are reduced by reducing generator weight from 35t to around 10t, and turbomachinery shaft weight from 32t to around 10t, (3) power electronics costs are reduced (since generator is reduced from 300MW to 66MW in gas turbine section), (4) plant efficiency is increased compared to the GT-MHR Brayton cycle, (5) steam turbine and steam cycle electrical generator are commercial off-the-shelf items - low cost and low risk.

Area:	☐ NHSS	☐ HTS	☐ HPS	☒ PCS	☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Subsystem Verified at Experimental Scale	Subsystem Verified at Pilot Scale	System Verified at Engineering Scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 6 is achieved upon successful completion of the required testing outlined in the TRL rating sheet for TRL 5. Specifically, (1) the generator winding insulation and electrical lead tests have been successfully performed at operating temperatures (<850ºC) and helium operating pressures (7,020 kPa; 1,020 psi), (2) the design of the turbine's components (blades, etc) has been tested for aerodynamic performance at operating temperatures and helium pressures, (3) seal and bearing tests have been completed successfully in operating temperatures and helium pressures, and (4) it has been verified that compressor materials will not suffer excessive embrittlement in the helium environment.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Full-scale components that comprise the PCS will be tested in ambient air conditions to achieve TRL 7. Additional sheet provides more detail.	See Additional Action Sheet	2009-2014	10,000

DDN(s) Supported: GT-MHR Russian Program PCU TDPP doc [filename TDPP_Aug06.doc]	**Technology Case File:**
Subject Matter Expert Making Determination: A. Bozek	
Date: 12/11/2008	**Originating Organization:** General Atomics

Additional Action Sheet(s)

Achieve successful test results for each of the following:

GENERATOR:
- fabricate a full-scale generator and test in ambient temperature and pressure environment. Include control software and instrumentation.

TURBOCOMPRESSOR (TC):
- fabricate a full-scale turbocompressor and test is ambient temperature and pressure environment. The test rotor should consist of turbine rotors, compressor, and the diaphragm coupling. The test stator should consist of casings and stationary components of the turbine and compressor, electromagnetic bearings (including catcher bearings), and buffer/repair/stator seals. All turbocompressor components should be integrated into one test rig. Include control software and instrumentation.

Possible Actionees:
Generator and Turbocompressor Fabrication and Integration Testing: Northrop Grumman; Transcanada Turbines; MILCON P-104 Gas Turbine Test Facility

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-11.4	**Revision:** 0

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Power Conversion System (PCS)

Description:
The combined gas and steam cycle consists of a 66MWt gas turbine generator with the remainder of the power driving the steam cycle. The key features of this concept are: (1) the recuperator is no longer required (a steam generator would be required, but this is considered much lower risk), (2) electromagnetic bearing risks are reduced by reducing generator weight from 35t to around 10t, and turbomachinery shaft weight from 32t to around 10t, (3) power electronics costs are reduced (since generator is reduced from 300MW to 66MW in gas turbine section), (4) plant efficiency is increased compared to the GT-MHR Brayton cycle, (5) steam turbine and steam cycle electrical generator are commercial off-the-shelf items - low cost and low risk.

Area: ☐ NHSS ☐ HTS ☐ HPS ☒ PCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Subsystem Verified at Pilot Scale	System Verified at Engineering Scale	System Tested and Qualified
TRL	**6**	**7**	**8**

Basis for Rating (Check box if continued on additional sheets) ☐

TRL 7 is achieved upon successful completion of the testing activities outlined in the TRL rating sheet for TRL 6. Specifically, (1) the full-scale generator and the diaphragm coupling were successfully tested at ambient conditions, and (2) the full-scale turbocompressor and its bearings and seals were successfully tested at ambient conditions.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
An integrated prototype of the PCS turbomachinery (TC, generator, and bearings) will be tested in operating temperature (<850ºC) and helium pressure (7,020 kPa; 1,020 psi) to achieve TRL 8. Additional sheet provides more detail.	NGNP	1/2015 through 3/2018	10,000

DDN(s) Supported: GT-MHR Russian Program PCU TDPP doc [filename TDPP_Aug06.doc]	**Technology Case File:**

Subject Matter Expert Making Determination: A. Bozek

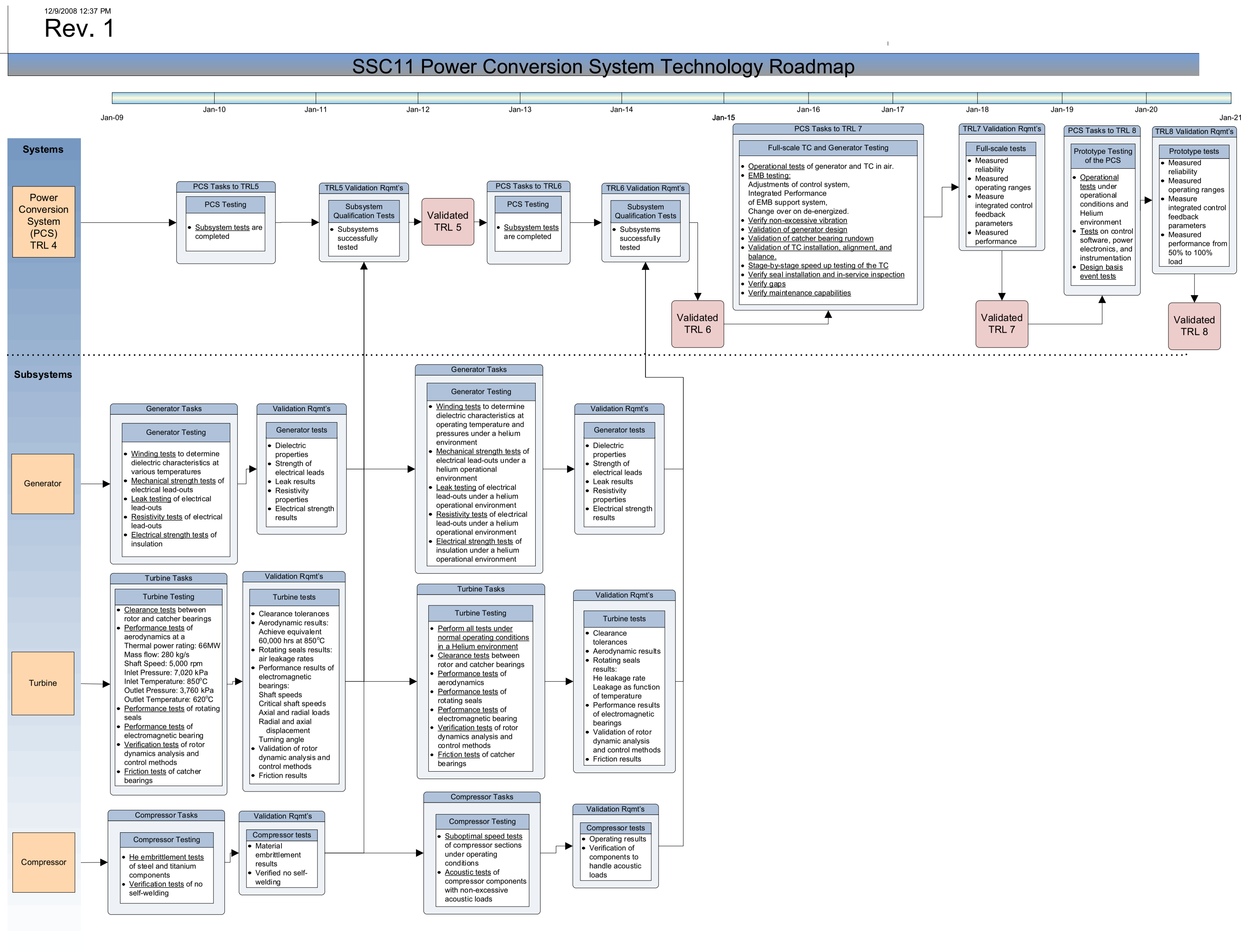
Date: 12/11/2008	**Originating Organization:** General Atomics

Additional Action Sheet(s)

Actions:

Achieve successful test results for each of the following:

TURBOMACHINERY (GENERATOR, TURBOCOMPRESSOR, AND BEARINGS):
- fabricate full-scale turbomachinery and test in simulated operating temperature and helium pressure environment. Include control software, instrumentation, and power control electronics. Simulate design basis events during testing with turbomachinery response. Assuming diaphragm coupling between TC and generator, test reduction in resonance modes and independent operation of bearing systems between TC and generator.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Rev. 1
SSC11 Power Conversion System Technology Roadmap
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Systems
Power Conversion System (PCS) TRL 4
PCS Tasks to TRL5
PCS Testing
Subsystem tests are completed
TRL5 Validation Rqmt's
Subsystem Qualification Tests
Subsystems successfully tested
Validated TRL 5
PCS Tasks to TRL6
PCS Testing
Subsystem tests are completed
TRL6 Validation Rqmt's
Subsystem Qualification Tests
Subsystems successfully tested
Validated TRL 6
PCS Tasks to TRL 7
Full-scale TC and Generator Testing
Operational tests of generator and TC in air.
EMB testing: Adjustments of control system, Integrated Performance of EMB support system, Change over on de-energized.
Verify non-excessive vibration
Validation of generator design
Validation of catcher bearing rundown
Validation of TC installation, alignment, and balance.
Stage-by-stage speed up testing of the TC
Verify seal installation and in-service inspection
Verify gaps
Verify maintenance capabilities
TRL7 Validation Rqmt's
Full-scale tests
Measured reliability
Measured operating ranges
Measure integrated control feedback parameters
Measured performance
Validated TRL 7
PCS Tasks to TRL 8
Prototype Testing of the PCS
Operational tests under operational conditions and Helium environment
Tests on control software, power electronics, and instrumentation
Design basis event tests
TRL8 Validation Rqmt's
Prototype tests
Measured reliability
Measured operating ranges
Measure integrated control feedback parameters
Measured performance from 50% to 100% load
Validated TRL 8
Subsystems
Generator
Generator Tasks
Generator Testing
Winding tests to determine dielectric characteristics at various temperatures
Mechanical strength tests of electrical lead-outs
Leak testing of electrical lead-outs
Resistivity tests of electrical lead-outs
Electrical strength tests of insulation
Validation Rqmt's
Generator tests
Dielectric properties
Strength of electrical leads
Leak results
Resistivity properties
Electrical strength results
Generator Tasks
Generator Testing
Winding tests to determine dielectric characteristics at operating temperature and pressures under a helium environment
Mechanical strength tests of electrical lead-outs under a helium operational environment
Leak testing of electrical lead-outs under a helium operational environment
Resistivity tests of electrical lead-outs under a helium operational environment
Electrical strength tests of insulation under a helium operational environment
Validation Rqmt's
Generator tests
Dielectric properties
Strength of electrical leads
Leak results
Resistivity properties
Electrical strength results
Turbine
Turbine Tasks
Turbine Testing
Clearance tests between rotor and catcher bearings
Performance tests of aerodynamics at a Thermal power rating: 66MW Mass flow: 280 kg/s Shaft Speed: 5,000 rpm Inlet Pressure: 7,020 kPa Inlet Temperature: 850°C Outlet Pressure: 3,760 kPa Outlet Temperature: 620°C
Performance tests of rotating seals
Performance tests of electromagnetic bearing
Verification tests of rotor dynamics analysis and control methods
Friction tests of catcher bearings
Validation Rqmt's
Turbine tests
Clearance tolerances
Aerodynamic results: Achieve equivalent 60,000 hrs at 850°C
Rotating seals results: air leakage rates
Performance results of electromagnetic bearings: Shaft speeds Critical shaft speeds Axial and radial loads Radial and axial displacement Turning angle
Validation of rotor dynamic analysis and control methods
Friction results
Turbine Tasks
Turbine Testing
Perform all tests under normal operating conditions in a Helium environment
Clearance tests between rotor and catcher bearings
Performance tests of aerodynamics
Performance tests of rotating seals
Performance tests of electromagnetic bearing
Verification tests of rotor dynamics analysis and control methods
Friction tests of catcher bearings
Validation Rqmt's
Turbine tests
Clearance tolerances
Aerodynamic results
Rotating seals results: He leakage rate Leakage as function of temperature
Performance results of electromagnetic bearings
Validation of rotor dynamic analysis and control methods
Friction results
Compressor
Compressor Tasks
Compressor Testing
He embrittlement tests of steel and titanium components
Verification tests of no self-welding
Validation Rqmt's
Compressor tests
Material embrittlement results
Verified no self-welding
Compressor Tasks
Compressor Testing
Suboptimal speed tests of compressor sections under operating conditions
Acoustic tests of compressor components with non-excessive acoustic loads
Validation Rqmt's
Compressor tests
Operating results
Verification of components to handle acoustic loads

4.12 SSC-12 High Temperature Valves

TRL Rating Sheets, TRL 3 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-12.1	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: High Temperature Isolation Valves and Pressure Relief Valve

Description:
High temperature valves are located in the secondary heat transport loop, on the inlet to the main circulator, and on the inlet to the secondary shutdown circulator piping. The secondary side helium temperatures are assumed to be 925°C and 565°C for the hot and cold legs respectively. This is consistent with the secondary heat transport loop temperatures assumptions in GA Report 911105/0. It is also assumed that there will be three (3) valves on each hot and cold legs for various reasons outlined in the GA report mentioned above. These valves will be an integral part of the plant protective system actions for secondary loop isolation events. (Reference GA Report 911120/0). Valves may be 2 way or 3 way, globe type or ball, gate, spring or pilot operated, manual, and automatic or actuated.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Application formulated	Proof of principal	Component verified at bench scale
TRL	**2**	**3**	**4**

Basis for Rating (Check box if continued on additional sheets) ☒

A TRL of 3 is assigned to the high-temperature valves on the grounds that while a significant amount of test data and research is available on high temperature valves, test data specific to the service conditions and configuration for NGNP is not known to be available. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1) Establish relevant standards and code applicability (ASME Boiler & pressure vessel Section III Class 1, nuclear, piping, NQA-1, ASME OME-1-2007, ASME OM-2008) (Cont.)	URS-WD	1 year	300 to 350

DDN(s) Supported: C.14.01.04, N.42.02.01, N.42.02.02

Technology Case File:

Subject Matter Expert Making Determination: David T. Carroccia

Date: 12-1-08 **Originating Organization:** Washington Division of URS

Additional Basis Sheet(s)

Basis:

Additionally, the critical characteristics for the valves in question have not been proven for the service conditions at NGNP. The critical characteristics for which these valves must be designed are not defined, particularly:

- Allowable Valve Leakage
- Valve response times required
- Acceptable valve open pressure drop
- Accident excursion temperatures
- Accident excursion pressures
- Valve Configuration and Actuator type

<table>
<tr><th colspan="4">Additional Action Sheet(s)</th></tr>
<tr><th>Actions (list all)</th><th>Actionee</th><th>Schedule</th><th>Cost ($K)</th></tr>
<tr><td>2) Determine thermal and mechanical properties of valve materials through coupon tests as needed (to fill gaps in the literature) including:
- Chemistry
- Erosion
- Room temperature properties
- Endurance limit analysis
- Weld
- Material corrosion
- Stress corrosion cracking
- Elevated temperature properties
- Irradiation and post irradiation examination
- Environmental exposure/embrittlement
- Fasteners, and seals
- Helium permeability
- Sliding surface friction
- Variation in properties following exposure and aging
- Actuator torque requirements
- Performance characteristics
- Lubrication
- Determine applicability of EPRI PPM

3) Establish conditions of service under normal and design basis event conditions

4) Valve material stress testing

5) Material durability tests

6) Determine performance of gaskets, packing material and seals</td><td></td><td></td><td></td></tr>
</table>

TRL Rating Sheet

Vendor: GA	Document Number: SSC-12.2	Revision: 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: High Temperature Isolation Valves and Pressure Relief Valve

Description:
High temperature valves are located in the secondary heat transport loop, on the inlet to the main circulator, and on the inlet to the secondary shutdown circulator piping. The secondary side helium temperatures are assumed to be 925°C and 565°C for the hot and cold legs respectively. This is consistent with the secondary heat transport loop temperatures assumptions in GA Report 911105/0. It is also assumed that there will be three (3) valves on each hot and cold legs for various reasons outlined in the GA report mentioned above. These valves will be an integral part of the plant protective system actions for secondary loop isolation events. (Reference GA Report 911120/0). Valves may be 2 way or 3 way, globe type or ball, gate, spring or pilot operated, manual, and automatic or actuated.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	Parent:	WBS:

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Proof of Principal	Component verified at bench scale	Component verified at experimental scale
TRL	**3**	**4**	**5**

Basis for Rating (Check box if continued on additional sheets) ☐

A TRL of 4 is achieved upon successful completion of the action items identified in the TRL rating sheet for TRL 3.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1) Material Selection and Valve Configuration (Body, Bonnet, Seat, Seal, Stem and Packing) 2) 3d Modeling and analytical simulation including FEA stress analysis, heat transfer analysis and CFD modeling (Cont.)	GA/URS-WD GA/URS-WD	1 year	450 (not including GA Scope)

DDN(s) Supported: C.14.01.04, N.42.02.01, N.42.02.02	**Technology Case File:**
Subject Matter Expert Making Determination: David T. Carroccia	
Date: 12-8-08	**Originating Organization:** Washington Division of URS

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
3) Endurance Limit and Creep Analysis	GA/URS-WD, SME		
4) Identify Maintenance Requirements, ALARA analysis and RAMI characteristics	GA/URS-WD		
5) Erosion and corrosion accelerated wear testing, environmental qualification of valve materials, He leak tightness & Weld Methods, dissimilar materials and differential thermal expansion	GA/URS-WD		
6) Interfaces with adjoining structures, piping. Insulation, installation, maintenance access, contamination control	GA/URS-WD		

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-12.3	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: High Temperature Isolation Valves and Pressure Relief Valve

Description:
High temperature valves are located in the secondary heat transport loop, on the inlet to the main circulator, and on the inlet to the secondary shutdown circulator piping. The secondary side helium temperatures are assumed to be 925°C and 565°C for the hot and cold legs respectively. This is consistent with the secondary heat transport loop temperatures assumptions in GA Report 911105/0. It is also assumed that there will be three (3) valves on each hot and cold legs for various reasons outlined in the GA report mentioned above. These valves will be an integral part of the plant protective system actions for secondary loop isolation events. (Reference GA Report 911120/0). Valves may be 2 way or 3 way, globe type or ball, gate, spring or pilot operated, manual, and automatic or actuated.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Component verified at bench scale	Component verified at experimental scale	Component verified at pilot scale
TRL	**4**	**5**	**6**

Basis for Rating (Check box if continued on additional sheets) ☐

A TRL of 5 is achieved upon successful completion of the action items identified in the TRL rating sheet for TRL 4.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1) Physical Test Preparation for Pilot Scale Test Articles which are representative of valve designs 2) Tests using test apparatus 3) Determination of applicable NDE methods 4) Verify 3d (scale) models based on test results 5) Determine Leak Rate Detection Method Validation	GA/URS-WD	1 year	400 - 450

DDN(s) Supported: C.14.01.04, N.42.02.01, N.42.02.02

Technology Case File:

Subject Matter Expert Making Determination: David T. Carroccia

Date: 12-3-08 **Originating Organization:** Washington Division of URS

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-12.4	**Revision:** 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: High Temperature Isolation Valves and Pressure Relief Valve

Description:
High temperature valves are located in the secondary heat transport loop, on the inlet to the main circulator, and on the inlet to the secondary shutdown circulator piping. The secondary side helium temperatures are assumed to be 925°C and 565°C for the hot and cold legs respectively. This is consistent with the secondary heat transport loop temperatures assumptions in GA Report 911105/0. It is also assumed that there will be three (3) valves on each hot and cold legs for various reasons outlined in the GA report mentioned above. These valves will be an integral part of the plant protective system actions for secondary loop isolation events. (Reference GA Report 911120/0). Valves may be 2 way or 3 way, globe type or ball, gate, spring or pilot operated, manual, and automatic or actuated.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Component verified at experimental scale	Component verified at pilot scale	Component verified at engineering scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☐

A TRL of 6 is achieved upon successful completion of the action items identified in the TRL rating sheet for TRL 5.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1) Valve design verification including valve body, bonnet, plug, seal, packing, insulation, ball and seat, stem, bellows, jacket and actuator (as equipped) 2) Integrated experimental scale model testing including relief valve and isolation valve (Cont.)	GA/URS-WD	1 year	750 - 800

DDN(s) Supported: C.14.01.04, N.42.02.01, N.42.02.02	**Technology Case File:**

Subject Matter Expert Making Determination: David T. Carroccia

Date: 12-3-08	**Originating Organization:** Washington Division of URS

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
3) CFD/FEA validation and optimization 4) Leak detection validation 5) Validate installation, inspection and maintenance techniques 6) Assess test plan results and perform risk based analysis on need for next level of testing			

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-12.5	**Revision:** 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: High Temperature Isolation Valves and Pressure Relief Valve

Description:
High temperature valves are located in the secondary heat transport loop, on the inlet to the main circulator, and on the inlet to the secondary shutdown circulator piping. The secondary side helium temperatures are assumed to be 925°C and 565°C for the hot and cold legs respectively. This is consistent with the secondary heat transport loop temperatures assumptions in GA Report 911105/0. It is also assumed that there will be three (3) valves on each hot and cold legs for various reasons outlined in the GA report mentioned above. These valves will be an integral part of the plant protective system actions for secondary loop isolation events. (Reference GA Report 911120/0). Valves may be 2 way or 3 way, globe type or ball, gate, spring or pilot operated, manual, and automatic or actuated.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Component verified at pilot scale	Component verified at engineering scale	Component tested and qualified
TRL	**6**	**7**	**8**

Basis for Rating (Check box if continued on additional sheets) ☐

A TRL of 7 is achieved upon successful completion of the action items identified in the TRL rating sheet for TRL 6. Risk based determination has been made at the previous level whether sufficient need exists to proceed to this level of testing.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1) Integrated CTF Testing (as part of a larger test effort) 2) Stress Analysis Validation 3) Temperature and Flow Analysis Validation 4) In-Service Inspection Techniques Validation	GA/URS-WD	2 years coordinated with other activities at CTF)	750 – 950 Not including INL/BEA scope

DDN(s) Supported: C.14.01.04, N.42.02.01, N.42.02.02	**Technology Case File:**
Subject Matter Expert Making Determination: David T. Carroccia	
Date: 12-8-08	**Originating Organization:** Washington Division of URS

High Temperature Isolation Valves and Relief Valves Technology RoadMap - Drive to Technical Readiness Level of "8"

TRL 3

Tasks to Achieve TRL 4

- Define design conditions (normal, upset and fault)
- Establish safety class and code applicability
- Obtain or determine material properties at design conditions for all materials under consideration.
- Construct Preliminary FEA & CFD Models

Tasks to Achieve TRL 5

- Based on material property information, select valve body and seat material
- Based upon decision discriminators, engineering practice and vendor input, select valve type
- Have bench scale valve fabricated
- Obtain bench scale test data at design conditions:
  - Seal performance (leak testing)
  - Mechanical operability (cycle testing)
  - Determine performance of internal cooling or insulation if needed
  - Perform detailed CFD & FEA model analysis
  - Perform (independent) Endurance Limit Analysis & non-linear analyses for creep, fatigue
  - Determine maintenance, inspection & in-service verification requirements

Tasks to Achieve TRL 6

- Test valves at Pilot scale
- Validate performance & environmental qualification of materials
- Develop & verify in-service inspection methods

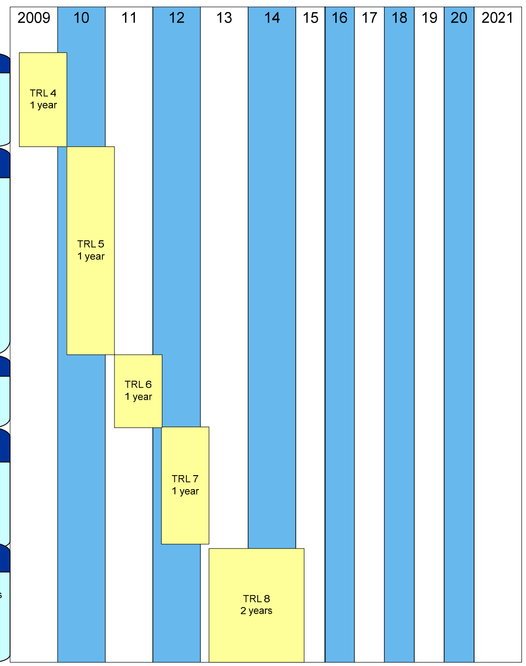
Tasks to Achieve TRL 7

- Fully integrated complete valve subsystem testing including actuator and control equipment
- Scale model testing & FEA & CFD Model validation
- Optimization
- Validate ISI / IST and maintenance Methods

Tasks to Achieve TRL 8

- Develop prototype valve for integration into CTF test loop
- Dynamically test valve mechanical operation, and environmental effects within test loop
- Validate model results, analyze valve performance at operational conditions
- Temperature & flow analysis validation
- Accoustic Testing

Validated TRL 8



Technology Down Selection

Assumed Configuration: Large I HX in Primary Loop and Max Reactor Outlet Temperature of 950 ^{0}C

Candidate Technologies – Seat and Packing Material

- Stellite and cobalt based alloys
- Stellite/chromium carbide composite
- Ceramics
- Carbon

Candidate Technologies – Valve Body Material

- Haynes 230 (N06230)
- Alloy 800H/AT (N08810/N08811)
- Inconel 617 (N06617)
- Hastelloy X (N06002)
- ASTM A297 Grade HK

Candidate Technologies – Valve Type

- Gate
- Ball
- Globe
- Angle valve
- Butterfly
- 2-way
- 3-way

Down Select Tasks – General

- Obtain normal, upset and faulted condition pressures and temperatures
- Obtain thermal properties of selected materials at design conditions
- Develop (if not available) thermal transient analysis for selected materials
- Analyze RAMI for valve materials / construction
- Obtain environmental effects data on valve materials
- Obtain environmental parameters for valve locations
- Ensure ASME III and 10CFR50.55(a) support material selection
- Determine packing, gasket and seal material performance
- Test for deterioration of material properties
- Test for thermal deformation of seat causing fail to seal (for valve seat)
- Stress analysis of selected components – comparison to allowable values
- Determine if active cooling is required and insulation needs
- Material Degradation

Decision Discriminators – General

- Actuator EQ
- Valve closing time
- Cost
- ASME codification requirements
- Ability to operate at design conditions without the need for active internal cooling or internal insulation.
- RAMI - Reliability, Accessibility, Maintainability and Inspectability

Decision Discriminators – Valve Seat

- Leak tightness at design conditions
- Wear and spalling resistance
- Wear product release rate

Decision Discriminators – Valve Body

- Creep fatigue, creep
- Carburization, chemical effects
- Radiation effects
- Thermal expansion properties
- Actual design stress compared to allowable stresses

Decision Discriminators – Valve Type

- Response to temperature transients (valve distortion)
- Leak tightness in He environment
- Pressure drop
- Ability to handle differential pressure requirement
- Operability at the design conditions

4.13 SSC-13 S-I Hydrogen Production System

TRL Rating Sheets, TRL 3 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-13.1	**Revision:** 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Reactive Distillation Column for HPS

Description:
A 15 MW Sulfur-Iodine Hydrogen Production System will be coupled with the NGNP producing 4.25 million liters of hydrogen per hour.

Area: ☐ NHSS ☐ HTS ☒ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Technology concept and application formulated	Proof of Concept	Component Verified at Bench Scale
TRL	**2**	**3**	**4**

Basis for Rating (Check box if continued on additional sheets) ☐

A laboratory experiment has been performed providing a proof of concept. Reactive distillation work in 2005 at prototypical temperatures and pressures showed greater than expected hydrogen production rates when iodine vapor concentration was low, but lower than expected rates with significant amounts of iodine present. Work ceased at that time due to funding and scheduling constraints. However, it is postulated that a larger column with more separation stages will allow for iodine-lean regions in the reaction zone of the column with expected concentrations of iodine in the feed.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Redesign and set up laboratory scale experiment. 2. Successfully operate multiple reactive distillation experiments. 3. Validate materials to be used for future testing.	GA	11/2008 – 04/2009	250

DDN(s) Supported: HPS-HID-01, -02, -03, -04	**Technology Case File:**
Subject Matter Expert Making Determination: Bob Buckingham	
Date: 10/31/08	**Originating Organization:** General Atomics

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-13.2	**Revision:** 1

☐ Area ☐ System ☒ Subsystem/Structure ☐ Component ☐ Technology

Title: Bunsen Reaction Section for HPS

Description:
A 15 MW Sulfur-Iodine Hydrogen Production System will be coupled with the NGNP producing 4.25 million liters of hydrogen per hour.

Area: ☐ NHSS ☐ HTS ☒ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Proof of Concept	Component Verified at Bench Scale	Technology validated in relevant environment
TRL	**3**	**4**	**5**

Basis for Rating (Check box if continued on additional sheets) ☐

The Bunsen reactor has been verified by laboratory scale testing, it is now being tested in the ILS experiment for the SI cycle.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Complete Bunsen reactor testing in ILS environment. 2. Validate all components of the system; piping, valves, pumps, drive motors, instrumentation and controls.	CEA	11/2008 – 09/2009	700

DDN(s) Supported: HPS-BUN-01, -02, -03, -04	**Technology Case File:**

Subject Matter Expert Making Determination: Bob Buckingham

Date: 10/31/08	**Originating Organization:** General Atomics

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC13.3	**Revision:**	1

☐ Area ☐ System ☒ Subsystem/Structure ☐ Component ☐ Technology

Title: Hydriodic Acid Decomposition Section for HPS

Description:
A 15 MW Sulfur-Iodine Hydrogen Production System will be coupled with the NGNP producing 4.25 million liters of hydrogen per hour.

Area: ☐ NHSS ☐ HTS ☒ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Proof of Concept	Component Verified at Bench Scale	Technology validated in relevant environment
TRL	**3**	**4**	**5**

Basis for Rating (Check box if continued on additional sheets) ☐

The HI decomposition section has been verified by multiple experiments at the lab scale. It is currently being tested as part of the ILS testing of the SI cycle. Reactive distillation work in 2005 at prototypical temperatures and pressures showed greater than expected hydrogen production rates when iodine vapor concentration was low, but lower than expected rates with significant amounts of iodine present. Work ceased at that time due to funding and scheduling constraints. However, it is postulated that a larger column with more separation stages will allow for iodine-lean regions in the reaction zone of the column with expected concentrations of iodine in the feed. Work on this specific component to validate this concept is a prerequisite for this ILS work to begin.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Complete HI decomposition testing at ILS scale. 2. Validate all components of the system; piping, valves, pumps, drive motors, instrumentation and controls.	GA	04/2009 - 09/2009	700

DDN(s) Supported: HPS-HID-01, -02, -03, -04, -05, -06, -07, -08, -09	**Technology Case File:**

Subject Matter Expert Making Determination: Bob Buckingham

Date: 10/31/08	**Originating Organization:** General Atomics

<table>
<tr><td colspan="6">TRL Rating Sheet</td></tr>
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<tr><td colspan="6">☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology</td></tr>
<tr><td colspan="6">Title: Hydrogen Production System (HPS)</td></tr>
<tr><td colspan="6">Description:
A 15 MW Sulfur-Iodine Hydrogen Production System will be coupled with the NGNP producing 4.25 million liters of hydrogen per hour.</td></tr>
<tr><td>Area:</td><td colspan="5">☐ NHSS ☐ HTS ☒ HPS ☐ BCS ☐ BOP</td></tr>
<tr><td colspan="2">ASSCT:</td><td colspan="2">Parent:</td><td colspan="2">WBS:</td></tr>
<tr><td colspan="6">Technology Readiness Level</td></tr>
<tr><td colspan="3"></td><td>Next Lower Rating Level</td><td>Current Rating Level</td><td>Next Higher Rating Level</td></tr>
<tr><td colspan="3">Generic Definitions (abbreviated)</td><td>Component Verified at Bench Scale</td><td>Technology validated in relevant environment</td><td>Similar subsystem in relevant env. for another application</td></tr>
<tr><td colspan="3">TRL</td><td>4</td><td>5</td><td>6</td></tr>
<tr><td colspan="6">Basis for Rating (Check box if continued on additional sheets) ☐</td></tr>
<tr><td colspan="6">All three sections of the SI cycle have been successfully integrated and tested at the laboratory scale. System piping, valves, pumps, drive motors, instrumentation and controls have all been validated.</td></tr>
<tr><td colspan="6">Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐</td></tr>
<tr><td colspan="3">Actions (list all)</td><td>Actionee</td><td>Schedule</td><td>Cost ($K)</td></tr>
<tr><td colspan="3">1. Design and build 70 kW pilot plant
2. Conduct long term tests for each of the three sections (1000 hrs)
3. Perform catalytic tests for acid decomposition sections (1000 hrs)
4. Produce 20,000 liters per hour of hydrogen</td><td>INL/SNL/GA</td><td>11/2008 – 09/2012</td><td>55,300</td></tr>
<tr><td colspan="3">DDN(s) Supported: HPS-SAD-01 through -15; HPS-FUS-01, -02, -03; HPS-BUN-01 through -07; HPS-HID-01 through -09; HPS-PPU-01, -02; HPS-PCN-01, -02, -03</td><td colspan="3">Technology Case File:</td></tr>
<tr><td colspan="6">Subject Matter Expert Making Determination: Bob Buckingham</td></tr>
<tr><td colspan="2">Date: 10/31/08</td><td colspan="4">Originating Organization: General Atomics</td></tr>
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TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-13.5	**Revision:** 1

☐ Area ☐ System ☒ Subsystem/Structure ☐ Component ☐ Technology

Title: Hydrogen Production System (HPS)

Description:
A 15 MW Sulfur-Iodine Hydrogen Production System will be coupled with the NGNP producing 4.25 million liters of hydrogen per hour.

Area: ☐ NHSS ☐ HTS ☒ HPS ☐ BCS ☐ BOP

ASSCT: 3.1	**Parent:** 3.0	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Technology validated in relevant environment	Similar subsystem in relevant env. for another application	Performance verific. of components under NGNP config. & relevant env.
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☐

All three sections of the SI cycle have been successfully tested and validated at the pilot plant scale.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Design and build 1.5 MW engineering scale plant 2. Conduct long term tests for each of the three sections (2,500 hrs) 3. Perform catalytic tests for acid decomposition sections (2,500 hrs) 4. Produce 425,000 liters per hour of hydrogen	INL/SNL/GA	09/2008 – 09/2015	91,000

DDN(s) Supported: HPS-SAD-01 through -15; HPS-FUS-01, -02, -03; HPS-BUN-01 through -07; HPS-HID-01 through -09; HPS-PPU-01, -02; HPS-PCN-01, -02, -03	**Technology Case File:**

Subject Matter Expert Making Determination: Bob Buckingham

Date: 10/31/08	**Originating Organization:** General Atomics

TRL Rating Sheet

Vendor: GA	Document Number: SSC-13.6	Revision: 1

☐ Area ☐ System ☒ Subsystem/Structure ☐ Component ☐ Technology

Title: Hydrogen Production System (HPS)

Description:
A 15 MW Sulfur-Iodine Hydrogen Production System will be coupled with the NGNP producing 4.25 million liters of hydrogen per hour.

Area: ☐ NHSS ☒ HTS ☒ HPS ☐ BCS ☐ BOP

ASSCT:	Parent:	WBS:

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Similar subsystem in relevant env. for another application	Performance verific. of components under NGNP config. & relevant env.	Prototype testing under operating environment
TRL	**6**	**7**	**8**

Basis for Rating (Check box if continued on additional sheets) ☐

All three sections of the SI cycle have been successfully tested and validated at the engineering plant scale.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

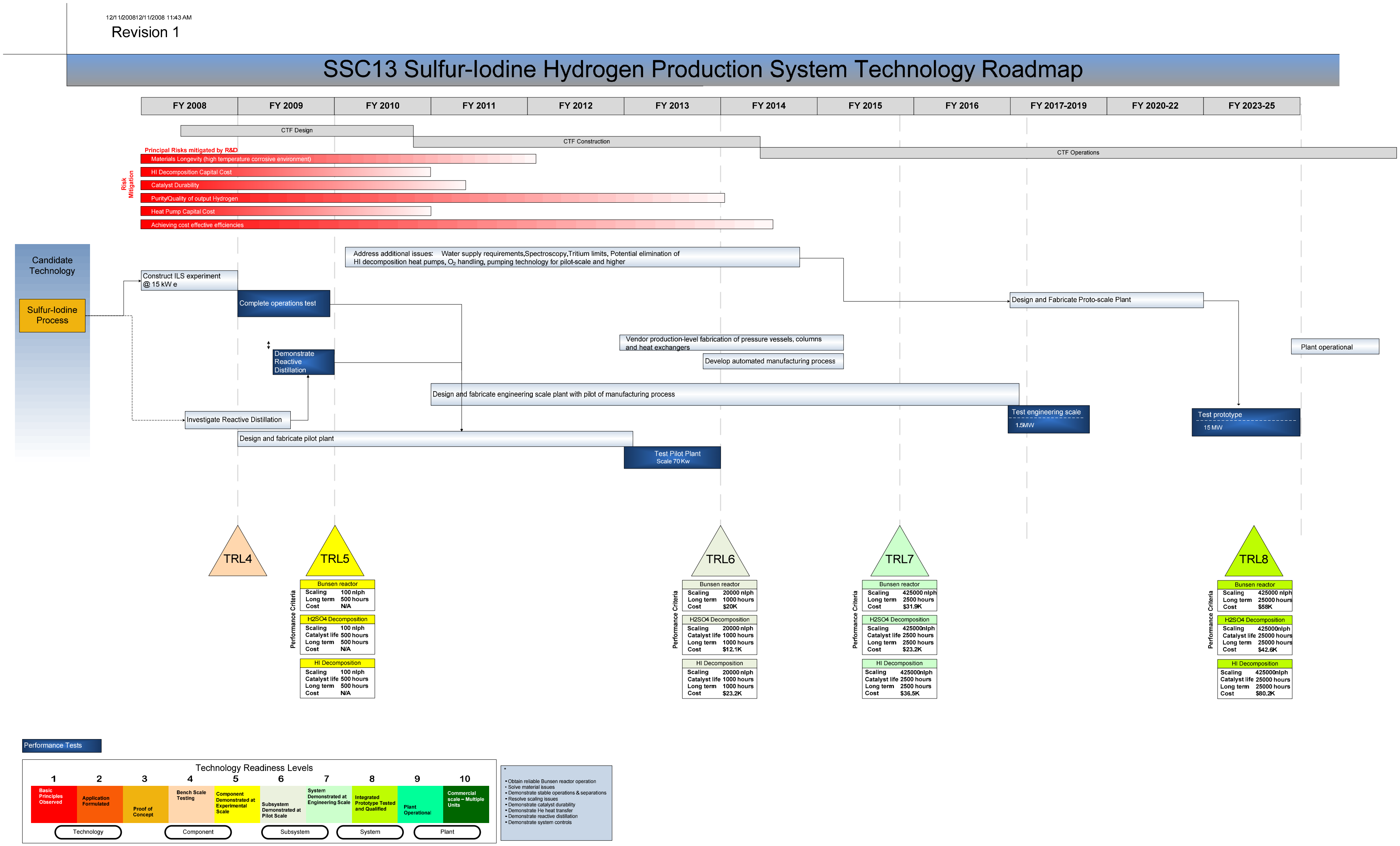
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Design and build 15 MW prototype plant to be coupled with the NGNP 2. Conduct long term tests for each of the three sections (25,000 hrs) 3. Perform catalytic tests for acid decomposition sections (25,000 hrs) 4. Produce 4,250,000 liters per hour of hydrogen	INL/SNL/GA	09/2012 – 09/2022	180,800

DDN(s) Supported: HPS-SAD-01 through -15; HPS-FUS-01, -02, -03; HPS-BUN-01 through -07; HPS-HID-01 through -09; HPS-PPU-01, -02; HPS-PCN-01, -02, -03

Technology Case File:

Subject Matter Expert Making Determination: Robert Buckingham

Date: 10/31/08 **Originating Organization:** General Atomics


12/11/200812/11/2008 11:43 AM
Revision 1
SSC13 Sulfur-Iodine Hydrogen Production System Technology Roadmap
FY 2008
FY 2009
FY 2010
FY 2011
FY 2012
FY 2013
FY 2014
FY 2015
FY 2016
FY 2017-2019
FY 2020-22
FY 2023-25
CTF Design
CTF Construction
CTF Operations
Risk Mitigation
Principal Risks mitigated by R&D
Materials Longevity (high temperature corrosive environment)
HI Decomposition Capital Cost
Catalyst Durability
Purity/Quality of output Hydrogen
Heat Pump Capital Cost
Achieving cost effective efficiencies
Candidate Technology
Sulfur-Iodine Process
Construct ILS experiment @ 15 kW e
Complete operations test
Demonstrate Reactive Distillation
Investigate Reactive Distillation
Design and fabricate pilot plant
Test Pilot Plant Scale 70 Kw
Address additional issues: Water supply requirements,Spectroscopy,Tritium limits, Potential elimination of HI decomposition heat pumps, O_2 handling, pumping technology for pilot-scale and higher
Vendor production-level fabrication of pressure vessels, columns and heat exchangers
Develop automated manufacturing process
Design and fabricate engineering scale plant with pilot of manufacturing process
Test engineering scale 1.5MW
Design and Fabricate Proto-scale Plant
Test prototype 15 MW
Plant operational
TRL4
TRL5
Performance Criteria
Bunsen reactor
Scaling 100 nlph
Long term 500 hours
Cost N/A
H2SO4 Decomposition
Scaling 100 nlph
Catalyst life 500 hours
Long term 500 hours
Cost N/A
HI Decomposition
Scaling 100 nlph
Catalyst life 500 hours
Long term 500 hours
Cost N/A
TRL6
Performance Criteria
Bunsen reactor
Scaling 20000 nlph
Long term 1000 hours
Cost $20K
H2SO4 Decomposition
Scaling 20000 nlph
Catalyst life 1000 hours
Long term 1000 hours
Cost $12.1K
HI Decomposition
Scaling 20000 nlph
Catalyst life 1000 hours
Long term 1000 hours
Cost $23.2K
TRL7
Performance Criteria
Bunsen reactor
Scaling 425000 nlph
Long term 2500 hours
Cost $31.9K
H2SO4 Decomposition
Scaling 425000nlph
Catalyst life 2500 hours
Long term 2500 hours
Cost $23.2K
HI Decomposition
Scaling 425000nlph
Catalyst life 2500 hours
Long term 2500 hours
Cost $36.5K
TRL8
Performance Criteria
Bunsen reactor
Scaling 425000 nlph
Long term 25000 hours
Cost $58K
H2SO4 Decomposition
Scaling 425000nlph
Catalyst life 25000 hours
Long term 25000 hours
Cost $42.6K
HI Decomposition
Scaling 425000nlph
Catalyst life 25000 hours
Long term 25000 hours
Cost $80.2K
Performance Tests
Technology Readiness Levels
1 Basic Principles Observed
2 Application Formulated
3 Proof of Concept
4 Bench Scale Testing
5 Component Demonstrated at Experimental Scale
6 Subsystem Demonstrated at Pilot Scale
7 System Demonstrated at Engineering Scale
8 Integrated Prototype Tested and Qualified
9 Plant Operational
10 Commercial scale – Multiple Units
Technology
Component
Subsystem
System
Plant
• Obtain reliable Bunsen reactor operation
• Solve material issues
• Demonstrate stable operations & separations
• Resolve scaling issues
• Demonstrate catalyst durability
• Demonstrate He heat transfer
• Demonstrate reactive distillation
• Demonstrate system controls

4.14 SSC-14 Fuel Handling and Storage System

TRL Rating Sheets, TRL 4 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-14.1	**Revision:** 1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Fuel Handling and Storage System (FHSS)

Description:
The FHSS is used to refuel the reactor and for all transfers of fuel and reflector elements between the reactor and local storage facilities and between the local storage facilities and the packaging and shipping facility. The system is also used to manipulate special tools for in-service inspection of reactor components. The major fuel handling and storage components (subsystems) include the fuel handling machine (FHM), the fuel transfer cask (FTC), the fuel handling equipment positioner (FHEP), the fuel handling equipment support structure (FHESS), the element hoist and grapple assembly (EHGA) in the local fuel storage facility, and the fuel sealing and inspection facility (FSIF). (Cont.)

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Proof of Concept	Components verified at bench scale	Components verified at experimental scale
TRL	**3**	**4**	**5**

Basis for Rating (Check box if continued on additional sheets) ☒

A large experience base exists from designing, building, testing and operating fuel handling equipment for the Peach Bottom and Fort St. Vrain (FSV) reactors. Although the Peach Bottom fuel handling machine was manually operated, important technology was developed in the areas of: (1) electrical power and signal cables for operation in 450°F helium with high gamma background; (2) lubricants for use in the same harsh environment; (3) electronic sensors for use on the grapple head; (4) grapple head floating plate technology for light touch in horizontal and vertical directions; and (5) general purpose manipulator technology adapted for special use in the reactor. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Conduct conceptual design of the FHSS components. The design effort will include a review of the current designs (developed in the early 1990's) and the current state of relevant technologies to ascertain the need for design changes to utilize current technology. Design improvements will be made based on the results of this review.	GA	18 months starting at the beginning of NGNP CD	1,900

DDN(s) Supported: C.21.01.04, C.21.01.07, C.21.01.08	**Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein

Date: 11-29-08	**Originating Organization:** General Atomics

Additional Description Sheet(s)

Description:

In-core fuel handling is performed by the FHM and the FTC working together. The functions of the various major FHSS components are summarized below:

- The FHM is a shielded, gas tight structure containing all the necessary mechanisms required to transfer fuel and reflector elements between the reactor core and the upper plenum
- The FTC is a shielded structure which transfers fuel and reflector elements between the fuel handling machine (inside the upper plenum) , and the FSIF and/or the Local Refueling and Storage Facilities (LRSF)
- The FHESS receives and supports fuel handling equipment over the reactor vessel during refueling
- The FHEP transfers and positions the FHM, FTC, FHESS, and auxiliary service cask between storage locations, reactor vessel and fuel/target processing facilities floor valves
- The EHGA robot is a remotely operated bridge robot in the LRSFs and FSIF which handle core elements, well plugs, and fuel elements
- The FSIF equipment loads spent fuel elements into shipping containers, seals the container lid, and inspects the resulting container integrity.

Operation of the FHSS is a key factor contributing to plant availability. The system must be highly reliable with sufficient redundancy to accommodate upset conditions and equipment failures. The equipment must minimize complexity and be readily maintainable, recognizing that it operates in a radioactive environment. These are all important requirements that require a comprehensive confirmation and endurance test program. The FHSS provides radiation protection to workers and public during refueling operations. The reactor containment is opened for refueling and the refueling equipment must be securely fastened and sealed to the pressure vessel. The equipment is designed to appropriate seismic requirements to maintain integrity with the reactor pressure vessel. Leakage of primary coolant from the reactor is prevented by maintaining the interior pressure slightly below atmospheric. In addition, the equipment is sealed to the reactor with elastomeric seals. In the event of upset conditions, such as an interior water leak, the equipment and seals are designed for the maximum pressure rise (approximately plus 25 psig). Machine controls and fail safe mechanisms are provided for the handling of fuel elements. Mislocating blocks, dropping or damaging blocks, or runaway machinery, etc., are concerns.

Additional Basis Sheet(s)

Basis:

The FSV FHM was designed and built in the late 1960's during the time that programmed machine tools were being developed for numerical control. This machine advanced from the Peach Bottom 1 technology in areas of: (1) computer control of multiple positioning systems in automatic mode or direct operator control in manual operation mode; (2) the use of electric motors, brakes, and position feedback instrumentation in a helium environment; (3) The use of a radiation-hardened television camera and lighting in helium; (4) programming techniques to safely operate the FHM within limits set by hard-wired interlocks and, (5) elementary inventory control, which was greatly enhanced in a 1989 control system upgrade.

The current design for the FHSS has evolved from the FSV technology. Years of experience with the FSV FHM have demonstrated both reliable features of the design and some features which could be improved. The current FHM design is based on the FSV FHM, but includes some mechanisms that differ from the FSV FHM:

- Shorter grapple probe
- Electrically controlled grapple mechanism rather than pneumatic
- Electrically controlled grapple head mechanism rather than pneumatic
- Increased handling mechanism linkage radial displacement
- Viewing system and electronic control system revised to incorporate more current technology
- Telescoping tube guide sleeve is transported and inserted by the FHM rather than an auxiliary service cask
- Vertical travel requirement is greater in order to operate in a deeper core

The FHSS also includes several new automated machines that must operate in concert. The simultaneous operation of these machines is necessary to refuel a reactor module within the allocated time.

The FTC and the EHGA robot are new designs required to operate in a helium environment. These machines incorporate proven technology where applicable. For example, the FTC will use grapple head, telescopic guide tubes, and isolation valve designs similar to those used in the FHM. The FHEP is similar to a commercially available, computer operated gantry crane with position control of the x, y, z, and load rotation axes. The EHGA robot and its end effectors are similar to the gantry robots applied by GA in the U.S. Army chemical weapons demilitarization development program. GA has developed the robotics for the remote handling of munitions in a lethal agent environment. The particular relevant expertise gained and "lessons learned" in the design, use and control of multiple gantry robots, end-effectors, and decontamination compatible hardware is available and applicable to the gantry robots to be used in the LRSFs and the FSIF. The computer control and element accountability system will utilize background data derived from the FSV project, commercial HTGR designs, the GA Demil program and industrial applications of computer controlled equipment. The FSV and Demil projects provide tested data bases for the FHSS computer architecture which include automated serialized accounting of fuel elements and target assemblies.

The baseline TRL assigned to the FHSS is 4 based on (1) the lowest TRL for the subsystems that comprise the FHSS and (2) the need to conduct tests to confirm the performance and environmental compatibility of instrumentation and control components and systems, and to firm up their design prior to overall system development and verification. A TRL of 4 is assigned to both the FHESS and the FSIF for the reasons given below. A TRL of 5 is assigned to the other FHSS subsystems based on the state of the technology as discussed above.

The FHESS with its multiple interfaces (i.e., the reactor isolation valves and neutron control assembly housing seals) is a first-of-a-kind unit. Although design of the FHESS is a routine structural task based on loads, deflections, and stability of the structure, consideration must also be given to the radiation shielding needed to prevent unnecessary personnel radiation exposure. Adequate vendor documentation is expected to be available for the seals and valves to warrant a TRL of 4, but testing is needed to validate the performance of these components.

Little design information is currently available for the FSIF and the equipment will be first-of-a-kind; however, the fuel handling and packaging mechanisms and procedures used in this facility will be based on those employed in FSV and in other HTGRs. Further, as noted above, the relevant expertise gained and "lessons learned" in the design, use, and control of multiple gantry robots, end-effectors, and decontamination compatible hardware is available and applicable to the gantry robots to be used in the FSIF. Thus, an initial TRL of 4 is judged appropriate for this FHSS subsystem.

It is also important to note that the conceptual designs of the current FHSS components were developed in the early 1990’s and were based on the technology available at that time. Further, the “technology development” activities defined for the FHSS in the technology development road map (TDRM) and supporting TRL rating sheets are primarily design verification tests. Thus, an important first step in NGNP FHSS technology development will be to review the current designs of the FHSS components and ascertain the extent to which previous design selections should be updated based on new technologies that have become available since the current designs were developed. It is not anticipated that any new technology will be need to be developed for the FHSS components; rather it is a matter of ensuring optimal utilization of currently available technology, particularly in the area of FHSS I&C.

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Perform a survey of the supply network for the types of equipment required for the NFSS and select vendors for the various components.	GA	6 months starting upon completion of action 1	350
3. Complete preliminary design of the FHSS	GA, FHSS component vendors	18 months starting at beginning of PD	3000
4. Perform testing as necessary to verify the accuracy and reliability of the instrumentation and control components under a variety of operating conditions and after frequent use. Test the fuel element identification equipment under a range of operating conditions including element motion, velocity, size of identification markings, lighting conditions, etc. Test other instrumentation under various operating speeds and environmental conditions to verify performance characteristics.	FHSS I&C vendor(s)	9 months starting at beginning of FD	900
5. Perform testing to demonstrate proper operation of the FHESS with its four built-in reactor isolation valves and inflatable seals. Test the inflatable seals that seat to the nuclear control assembly housings against offset (non-concentric) housing locations to simulate expected plant construction tolerances. Cycle valve operators and all seals to represent 10 refueling outages and demonstrate all interlocks. All testing will be conducted in ambient air.	FHESS vendor	1 year starting at beginning of FD	600
6. Perform tests of FSIF components to verify the automated packaging, sealing, and inspection processes (including leak-tightness testing capabilities).	FSIF component vendors	1 year starting at beginning of FD	900

<table>
<tr><td colspan="6">TRL Rating Sheet</td></tr>
<tr><td colspan="2">Vendor: GA</td><td colspan="2">Document Number: SSC-14.2</td><td colspan="2">Revision: 1</td></tr>
<tr><td colspan="6">☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology</td></tr>
<tr><td colspan="6">Title: Fuel Handling and Storage System (FHSS)</td></tr>
<tr><td colspan="6">Description:
The FHSS is used to refuel the reactor and for all transfers of fuel and reflector elements between the reactor and local storage facilities and between the local storage facilities and the packaging and shipping facility. The system is also used to manipulate special tools for in-service inspection of reactor components. The major fuel handling and storage components (subsystems) include the fuel handling machine (FHM), the fuel transfer cask (FTC), the fuel handling equipment positioner (FHEP), the fuel handling equipment support structure (FHESS), the element hoist and grapple assembly (EHGA) in the local fuel storage facility, and the fuel sealing and inspection facility (FSIF). In-core fuel handling is performed by the FHM and the FTC working together. These machines are positioned by the FHEP and are mounted on the FHESS, which is mounted on the reactor vessel.</td></tr>
<tr><td>Area:</td><td colspan="5">☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP</td></tr>
<tr><td colspan="2">ASSCT:</td><td colspan="2">Parent:</td><td colspan="2">WBS:</td></tr>
<tr><td colspan="6">Technology Readiness Level</td></tr>
<tr><td colspan="3"></td><td>Next Lower
Rating Level</td><td>Current
Rating Level</td><td>Next Higher
Rating Level</td></tr>
<tr><td colspan="3">Generic Definitions (abbreviated)</td><td>Components verified at bench scale</td><td>Components verified at experimental scale</td><td>Subsystems verified at pilot scale</td></tr>
<tr><td colspan="3">TRL</td><td>4</td><td>5</td><td>6</td></tr>
<tr><td colspan="6">Basis for Rating (Check box if continued on additional sheets) ☒</td></tr>
<tr><td colspan="6">A TRL of 5 for the FHSS is achieved upon completion of the required component level testing for the FHESS and the FSIF. The tests on the FHESS seals and valves have qualified the materials used for these components and have verified the functionality and endurance of these components.</td></tr>
<tr><td colspan="6">Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒</td></tr>
<tr><td colspan="3">Actions (list all)</td><td>Actionee</td><td>Schedule</td><td>Cost ($K)</td></tr>
<tr><td colspan="3">1. Perform speed, accuracy, and extended cyclic endurance and structural testing of the FHEP to verify the design and to ensure the reliability and accuracy of the FHEP to retrieve, transport and place large, heavy machines and structures. The testing shall include measurement of the four-axis acceleration and velocity capabilities of the FHEP under static and dynamic load conditions to acquire the data needed to validate process speed and performance predictions. (Cont.)</td><td>GA and FHEP vendor</td><td>18 months staring at beginning of FD</td><td>1900</td></tr>
<tr><td colspan="3">DDN(s) Supported: C.21.01.01, C.21.01.02, C.21.01.03, C.21.01.06</td><td colspan="3">Technology Case File:</td></tr>
<tr><td colspan="6">Subject Matter Expert Making Determination: John Saurwein</td></tr>
<tr><td colspan="2">Date: 11-29-08</td><td colspan="4">Originating Organization: General Atomics</td></tr>
</table>

Additional Basis Sheet(s)
Basis: Experimental-scale testing of the FSIF components has verified the automated packaging, sealing, and inspection processes to be used in this subsystem. Testing of the FHSS instrumentation and control (I&C) components in air and in helium has demonstrated the performance and environmental compatibility of these components and has demonstrated that the I&C, including software, meets design requirements and is compatible with the fuel handling mechanisms used in the FHSS.

Additional Action Sheet(s)

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
The safety interlocks of the FHEP control system will be validated in the course of these tests.			
2. Perform EHGA robot testing to validate that material handling operations for all fuel element related tasks are done within the cycle time allocation. Predicted recovery tasks will be functionally tested. Environmental endurance testing in both air and helium will be performed.	GA and EHGA vendor	18 months starting at beginning of FD	850
3. Perform full-scale rig test to acquire data for FHM on functional and performance limits in anticipated operating modes and operating conditions: Phase 1: Automated checkout of grapple head Phase 2: Automated checkout of element transfer mechanisms over a full core sector Phase 3: Automated cycle test in 250°F helium	GA and FHM vendor	18 months starting at beginning of FD	1250
4. Use a full-scale test rig and test article to conduct tests in air (Phase 1) and helium (Phase 2) to establish the operability and reliability of the FTC and its components under expected environmental conditions. Key components include the vertical drive system for the hoist grapple, horizontal transfer table drive, and the complete grapple system. Test Phase 3 will be a separate cyclic test of the automated hold-downs and remote connections.	GA and FTC vendor	18 months starting at beginning of FD	1250
5. Complete final design of the FHSS based on the results of all component testing. Issue final procurement specifications for all equipment.	GA	2 years starting 18 months into FD	1500

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-14.3	**Revision:** 1

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Fuel Handling and Storage System (FHSS)

Description:
The FHSS is used to refuel the reactor and for all transfers of fuel and reflector elements between the reactor and local storage facilities and between the local storage facilities and the packaging and shipping facility. The system is also used to manipulate special tools for in-service inspection of reactor components. The major fuel handling and storage components (subsystems) include the fuel handling machine (FHM), the fuel transfer cask (FTC), the fuel handling equipment positioner (FHEP), the fuel handling equipment support structure (FHESS), the element hoist and grapple assembly (EHGA) in the local fuel storage facility, and the fuel sealing and inspection facility (FSIF). In-core fuel handling is performed by the FHM and the FTC working together. These machines are positioned by the FHEP and are mounted on the FHESS, which is mounted on the reactor vessel.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Components verified at experimental scale	Subsystems verified at pilot scale	System verified at engineering scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☐

A TRL of 6 is achieved upon completion of all design verification testing of the FHSS subsystems including the FHM, FTC, FHEP, FHESS, EHGA, and the FSIP and either (1) the results of the tests confirm that the subsystems meet all functional and operational requirements or (2) design modifications have been made to the final design to correct any deficiencies detected during testing.

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Perform an integrated test of the FHSS to verify that all components of the system function together and that system operations can be performed safely and reliably within the allocated time. The testing will involve full-scale fuel handling and control equipment with simulated fuel elements in an environment representative of the operational environment.	GA, component vendors, and testing organization	2.5 years with completion at least 6 months before installation of equipment at NGNP	6000

DDN(s) Supported: C.21.01.05	**Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein

Date: 11-29-08	**Originating Organization:** General Atomics

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-14.4	**Revision:** 1			

☐ Area ☒ System ☐ Subsystem/Structure ☐ Component ☐ Technology

Title: Fuel Handling and Storage System (FHSS)

Description:
The FHSS is used to refuel the reactor and for all transfers of fuel and reflector elements between the reactor and local storage facilities and between the local storage facilities and the packaging and shipping facility. The system is also used to manipulate special tools for in-service inspection of reactor components. The major fuel handling and storage components (subsystems) include the fuel handling machine (FHM), the fuel transfer cask (FTC), the fuel handling equipment positioner (FHEP), the fuel handling equipment support structure (FHESS), the element hoist and grapple assembly (EHGA) in the local fuel storage facility, and the fuel sealing and inspection facility (FSIF). In-core fuel handling is performed by the FHM and the FTC working together. These machines are positioned by the FHEP and are mounted on the FHESS, which is mounted on the reactor vessel.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Subsystems verified at pilot scale	System verified at engineering scale	System tested and qualified
TRL	**6**	**7**	**8**

Basis for Rating (Check box if continued on additional sheets) ☐

A TRL of 7 is achieved when the following conditions are satisfied: (1) The integrated system test with simulated fuel elements in an environment representative of the operating environment, and (2) The results of the integrated system test demonstrate that all of the subsystems function together and that the FHSS is capable of performing all required operations safely and reliably in the allocated time.

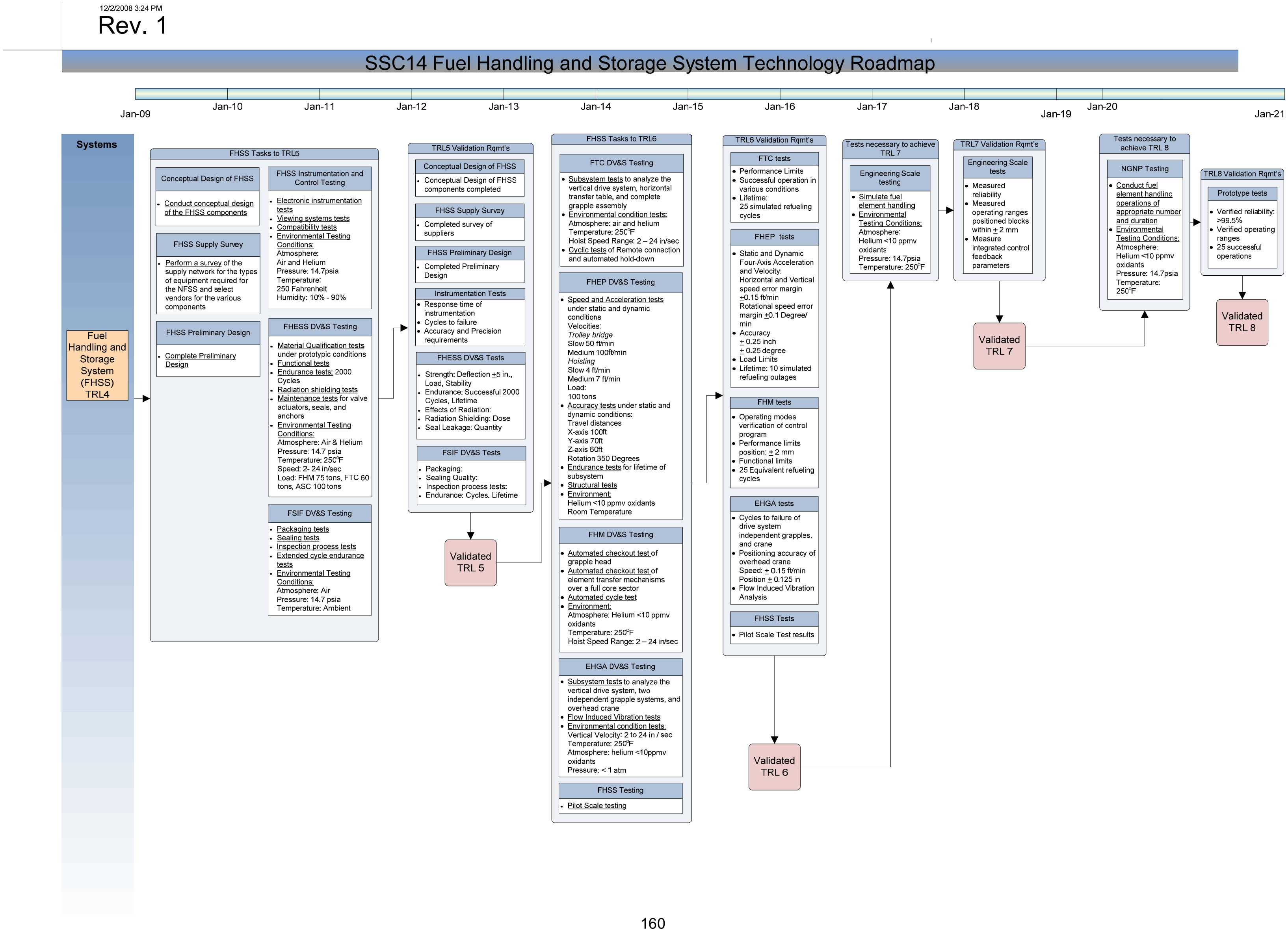
Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Conduct the appropriate number and duration of fuel handling operations in the actual operating environment (i.e., in the NGNP) to verify that the system meets reliability requirements.	GA, NGNP operator	During NGNP startup testing	TBD

DDN(s) Supported: None | **Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein

Date: 11-29-08 | **Originating Organization:** General Atomics



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Rev. 1
SSC14 Fuel Handling and Storage System Technology Roadmap
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Systems
Fuel Handling and Storage System (FHSS) TRL4
FHSS Tasks to TRL5
Conceptual Design of FHSS
Conduct conceptual design of the FHSS components
FHSS Supply Survey
Perform a survey of the supply network for the types of equipment required for the NFSS and select vendors for the various components
FHSS Preliminary Design
Complete Preliminary Design
FHSS Instrumentation and Control Testing
Electronic instrumentation tests
Viewing systems tests
Compatibility tests
Environmental Testing Conditions: Atmosphere: Air and Helium Pressure: 14.7psia Temperature: 250 Fahrenheit Humidity: 10% - 90%
FHESS DV&S Testing
Material Qualification tests under prototypic conditions
Functional tests
Endurance tests: 2000 Cycles
Radiation shielding tests
Maintenance tests for valve actuators, seals, and anchors
Environmental Testing Conditions: Atmosphere: Air & Helium Pressure: 14.7 psia Temperature: 250°F Speed: 2- 24 in/sec Load: FHM 75 tons, FTC 60 tons, ASC 100 tons
FSIF DV&S Testing
Packaging tests
Sealing tests
Inspection process tests
Extended cycle endurance tests
Environmental Testing Conditions: Atmosphere: Air Pressure: 14.7 psia Temperature: Ambient
TRL5 Validation Rqmt's
Conceptual Design of FHSS
Conceptual Design of FHSS components completed
FHSS Supply Survey
Completed survey of suppliers
FHSS Preliminary Design
Completed Preliminary Design
Instrumentation Tests
Response time of instrumentation
Cycles to failure
Accuracy and Precision requirements
FHESS DV&S Tests
Strength: Deflection ±5 in., Load, Stability
Endurance: Successful 2000 Cycles, Lifetime
Effects of Radiation:
Radiation Shielding: Dose
Seal Leakage: Quantity
FSIF DV&S Tests
Packaging:
Sealing Quality:
Inspection process tests:
Endurance: Cycles, Lifetime
Validated TRL 5
FHSS Tasks to TRL6
FTC DV&S Testing
Subsystem tests to analyze the vertical drive system, horizontal transfer table, and complete grapple assembly
Environmental condition tests: Atmosphere: air and helium Temperature: 250°F Hoist Speed Range: 2 – 24 in/sec
Cyclic tests of Remote connection and automated hold-down
FHEP DV&S Testing
Speed and Acceleration tests under static and dynamic conditions Velocities: Trolley bridge Slow 50 ft/min Medium 100ft/min Hoisting Slow 4 ft/min Medium 7 ft/min Load: 100 tons
Accuracy tests under static and dynamic conditions: Travel distances X-axis 100ft Y-axis 70ft Z-axis 60ft Rotation 350 Degrees
Endurance tests for lifetime of subsystem
Structural tests
Environment: Helium <10 ppmv oxidants Room Temperature
FHM DV&S Testing
Automated checkout test of grapple head
Automated checkout test of element transfer mechanisms over a full core sector
Automated cycle test
Environment: Atmosphere: Helium <10 ppmv oxidants Temperature: 250°F Hoist Speed Range: 2 – 24 in/sec
EHGA DV&S Testing
Subsystem tests to analyze the vertical drive system, two independent grapple systems, and overhead crane
Flow Induced Vibration tests
Environmental condition tests: Vertical Velocity: 2 to 24 in / sec Temperature: 250°F Atmosphere: helium <10ppmv oxidants Pressure: < 1 atm
FHSS Testing
Pilot Scale testing
TRL6 Validation Rqmt's
FTC tests
Performance Limits
Successful operation in various conditions
Lifetime: 25 simulated refueling cycles
FHEP tests
Static and Dynamic Four-Axis Acceleration and Velocity: Horizontal and Vertical speed error margin ±0.15 ft/min Rotational speed error margin ±0.1 Degree/ min
Accuracy ± 0.25 inch ± 0.25 degree
Load Limits
Lifetime: 10 simulated refueling outages
FHM tests
Operating modes verification of control program
Performance limits position: ± 2 mm
Functional limits
25 Equivalent refueling cycles
EHGA tests
Cycles to failure of drive system independent grapples, and crane
Positioning accuracy of overhead crane Speed: ± 0.15 ft/min Position ± 0.125 in
Flow Induced Vibration Analysis
FHSS Tests
Pilot Scale Test results
Validated TRL 6
Tests necessary to achieve TRL 7
Engineering Scale testing
Simulate fuel element handling
Environmental Testing Conditions: Atmosphere: Helium <10 ppmv oxidants Pressure: 14.7psia Temperature: 250°F
TRL7 Validation Rqmt's
Engineering Scale tests
Measured reliability
Measured operating ranges positioned blocks within ± 2 mm
Measure integrated control feedback parameters
Validated TRL 7
Tests necessary to achieve TRL 8
NGNP Testing
Conduct fuel element handling operations of appropriate number and duration
Environmental Testing Conditions: Atmosphere: Helium <10 ppmv oxidants Pressure: 14.7psia Temperature: 250°F
TRL8 Validation Rqmt's
Prototype tests
Verified reliability: >99.5%
Verified operating ranges
25 successful operations
Validated TRL 8

4.15 SSC-15 Primary Circuit and Balance of Plant Instrumentation

TRL Rating Sheets, TRL 4 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-15.1	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Reactor Control and Protection, Primary Circuit and Balance of Plant Instrumentation (SSC-15)

Description:
SSC-15 contains instrumentation equipment associated with the primary circuit and the balance of plant instrumentation, which will be placed in the primary helium circuit to detect leakage of radioactive materials, potentially affecting the public or plant personnel, and other instrumentation to provide defense-in-depth protection of reactor cooling functions. Instrumentation outside the reactor, but within the primary circuit or at particular points near the primary circuit boundary is considered Primary Circuit instrumentation. The Primary Circuit instrumentation provides detection of primary coolant leakage through measurement of pressure, temperature or radiation levels within the Reactor Building or in helium-to-helium heat exchanger piping which penetrates the Reactor Building. (Cont.)

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Application Formulated	Proof of Concept	Verified at Bench Scale
TRL	**2**	**3**	**4**

Basis for Rating (Check box if continued on additional sheets) ☒

The level 3 technical rating for SSC-15 is based on experience gained at the Fort St. Vrain nuclear generating station, using similar instrumentation for reactor control and protection. DDNs exist to verify these methods for the later MHR designs. Before a bench-scale rating level can be achieved, calculations to verify the preliminary control/protection requirements for NGNP multi-function plant operation must be completed. (Cont)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. From preliminary IHX piping and Reactor Building design information, select leak detection instrumentation locations in Reactor Building and hydrogen production facilities. Provide bench scale calculations to correlate leak magnitude and pressure/temperature changes in the Reactor Building. (Cont.)	GA	CD 0-12mo	20

DDN(s) Supported: C.31.01.01, C.34.01.02 **Technology Case File:**

Subject Matter Expert Making Determination: Dale Pfremmer

Date: 10/23/08 **Originating Organization:** General Atomics

Additional Description Sheet (s)

Description:

Also, SSC-15 includes moisture monitoring and pressure instrumentation for steam leakage detection, operator information, and as a protection-logic, reactor-trip parameter. SSC-15 includes plateout instrumentation to monitor and ascertain the level of radioactive plateout within the primary circuit as well. Helium flow-rate measurement is also included. Finally, Balance-of-Plant (BOP) measurements, comprised of steam flow rate, temperature, and pressure instrumentation contained in the steam-electric (BOP) equipment, complete the SSC-15 instrumentation group.

Additional Basis Sheet(s)

Basis:

This will provide a basis for later design efforts such as determination of helium flow measurement range and accuracy. Also, since available US gas-reactor operating experience is outdated, testing or other means of updating the database is required to achieve the level 4 rating that would precede the conceptual design of some of the instrumentation methods. This includes confirming application of instrumentation used outside the nuclear industry, or confirming application of instrumentation used in other high-temperature reactor development work, such as work in Japan, etc. Industrial proof-of-concept data can also improve and provide new bases for design of more modern instrumentation – for example, application of laser technology to moisture detection devices has come about since FSV. This instrumentation will undergo development and requires the technical rating process for application in the NGNP.

Helium flow rate measuring instrumentation, although not within the reactor design scope, is controlled and monitored through the reactor control and protection interfaces, and is included as part of the SSC-15 equipment. Pressure probes, piping and temperature sensors, located within the helium circulators, provide the helium flow rate instrumentation. Because of the integrated nature of this instrumentation, it must be included in the circulator design scope, with operational requirements derived through reactor control and protection requirements. This instrumentation is developed with the circulator design and will require verification of the development effort.

Reactor control/protection systems also use measurements comprised of steam flow rate, temperature, and pressure instrumentation contained in the steam-electric (BOP) equipment to coordinate nuclear control and electric-plant output, as well as to detect impairment of normal reactor heat rejection processes — ultimately using this information to determine if a reactor trip is required. For instance, the steam-turbine-trip parameter will be monitored by the reactor trip decision logic. This instrumentation is well established in nuclear electric plants, and so will not require verification prior to level 7.

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Obtain available bench scale data applicable for Primary Circuit radiation detection instrumentation and confirm, by test or analysis, capability to detect leaks through radiological measurements. Determine most likely means of placing such instrumentation in the primary circuit and provide bench-scale test verification of potential mounting schemes.	GA	CD 0-12mo	50
3. Verify bench scale instrumentation supplier data, and confirm that leakage, which could escape into the environment or endanger plant personnel if allowed to exceed specified levels, can be detected well below levels specified in RPS and IPS conceptual design documentation. Provide range and accuracy for instrumentation data base.	GA	CD 0-12mo	50
4. Contact circulator design team and verify incorporation of Helium Mass Flow Measurement in Circulator development effort. The circulator integrated instrumentation may also include safety-related primary helium temperature and pressure measurements.	GA Howden	CD 0-12mo CD 0-12mo	20 20
5. Provide bench-scale calculations for Plateout Probe instrumentation to determine fission product deposition levels. Acquire available plate-out technology information, such as OGL-1 plate-out measurement techniques, etc. Update planning for post-level-4 testing.	GA	CD 0-12mo	20
6. Include Steam Generator Moisture Ingress Detection Sensors in the bench scale verification effort. Survey and select from available commercial moisture monitoring equipment and perform tests to verify application of equipment to moisture detection design. Include new commercial technology such as Cavity Ring-Down Spectroscopy (CRDS) in evaluation. Update planning for post-level-4 testing.	GA Vendor	CD 0-12mo CD 0-12mo	50 20
7. Verify preliminary range, sensor accuracy, response, etc. for reactor control and protection instrumentation located in BOP. Include steam temperature, pressure and flow measurements. Perform bench scale reactor control, transient calculations. Update instrumentation reliability data from available nuclear-electric plant database. Include measurement redundancy, sensor fail-over techniques, signal transmission quality, etc.	GA	CD 0-12mo	50

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-15.2	**Revision:** 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Reactor Control and Protection, Primary Circuit and Balance of Plant Instrumentation (SSC-15)

Description:
SSC-15 contains instrumentation associated with the primary circuit and the balance of plant. Some of the instrumentation will be placed in the primary helium circuit or reactor building to detect leakage of radioactive materials, potentially affecting the public or plant personnel. Balance-of-Plant (BOP) measurements, comprised of steam flow rate, temperature, pressure, etc provide defense-in-depth protection of reactor cooling functions. Instrumentation within the primary circuit or included in the secondary boundary provided by the Reactor Building, is considered Primary Circuit instrumentation. (Cont.)

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Proof of concept	Verified at bench scale	Verified at experimental scale
TRL	**3**	**4**	**5**

Basis for Rating (Check box if continued on additional sheets) ☒

The current level 4 technical rating for SSC-15 is based on completion of activities required to achieve a level 4 technical rating. These activities provided a bench scale assessment of primary circuit and balance of plant instrumentation for the proposed NGNP design. Available commercial instrumentation and instrumentation used in earlier nuclear plants was reviewed. This provides information to start conceptual design activities. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Complete NHSS conceptual instrumentation design and coordinate with interfacing design areas – Reactor Building, BOP, etc. Provide preliminary views of each installed system and operational requirements for radiation detection, pressure, temperature, etc. measurement. Document design issues. (cont)	GA	CD 12-36mo	300

DDN(s) Supported: C.31.01.01, C.34.01.02	**Technology Case File:**

Subject Matter Expert Making Determination: Dale Pfremmer

Date: 10/23/08 **Originating Organization:** General Atomics

Additional Description Sheet(s)

Description:

SSC-15 instrumentation provides detection of primary coolant leakage through measurement of pressure, temperature or radiation levels within the Reactor Building or in helium piping which penetrates the Reactor Building. SSC-15 includes moisture monitoring and pressure instrumentation to detect steam in-leakage, provide operator information, and as a protection-logic input to the reactor-trip function. SSC-15 also includes helium flow rate instrumentation and radioactive plateout monitoring instrumentation.

Additional Basis Sheet(s)

Basis:

Achievement of the level 5 rating will require conceptual design selections from the available devices. Critical components within the instrumentation assemblies will be determined and testing at the component level will be performed. Industrial proof-of-concept data, provided by vendors, will also be reviewed to determine if further testing or other means of updating the database is required to achieve the level 5 rating. It is expected that most of this type of testing will involve advanced instrumentation systems, such as the moisture monitoring and plateout probe systems.

Conceptual design activities will also provide a range of plant operations, and analysis, to determine the helium flow rate measuring system requirements. These will be provided to the circulator development team, since pressure probes, piping and temperature sensors, etc located within the helium circulators, provides the helium flow rate instrumentation. Although not within the reactor design scope, helium flow rate is controlled and monitored through the reactor control and protection interfaces. Specification of requirements for this instrumentation is included under the SSC-15 equipment design activities. However, all testing activities will be completed under the circulator development scope.

Reactor control/protection analysis during conceptual design will also provide measurement requirements for steam-electric (BOP) equipment, such as steam flow rate, temperature, and pressure instrumentation. The SSC-15 effort will coordinate nuclear control and BOP electric-plant design requirements. This instrumentation is well established in nuclear electric plants, and so will not require verification testing, other than that provided in BOP development activities.

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Determine potential suppliers of Steam Generator Moisture Ingress Detection Sensors, based on selection from available commercial moisture monitoring equipment. Specify experimental scale tests to verify components of equipment for moisture detection design. Refer to uncertainties in industrial proof-of-concept data provided by vendors or other uncertainties requiring updates to the available database. Further testing is required in these cases.	GA Vendor	CD 12-24mo CD 12-24mo	50 50
3. Determine potential suppliers for Plateout Probe instrumentation to determine fission product deposition levels. Determine experimental scale testing to verify NGNP application.	GA Facility	CD 12-24mo CD 12-24mo	50 50
4. Complete testing (2) and (3) above. Verify application of components tested, and document resolution of design issues determined through test results. Resolve by analysis or other means, all design issues which do not require testing. Document results to confirm level 5 technical rating. Provide recommendation for testing at the next technical level.	GA	CD 24-36mo	100

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-15.3	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Reactor Control and Protection, Primary Circuit and Balance of Plant Instrumentation (SSC-15)

Description:
SSC-15 contains instrumentation associated with the primary circuit and the balance of plant. Some of the instrumentation will be placed in the primary helium circuit or reactor building to detect leakage of radioactive materials, potentially affecting the public or plant personnel. Balance-of-Plant (BOP) measurements, comprised of steam flow rate, temperature, pressure, etc provide defense-in-depth protection of reactor cooling functions. Instrumentation within the primary circuit or included in the secondary boundary provided by the Reactor Building, is considered Primary Circuit instrumentation. (Cont.)

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Verified at bench scale	Verified at experimental scale	Verified at pilot scale
TRL	**4**	**5**	**6**

Basis for Rating (Check box if continued on additional sheets) ☒

The current level 5 technical rating for SSC-15 is based on completion of activities required to achieve a level 5 technical rating. These activities provided component testing of new instrumentation required in NGNP, and analytical assessment supporting design application of the conventional instrumentation contained in the primary circuit and balance of plant. This provides information to start preliminary design activities. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1.Complete preliminary final instrumentation design and coordinate with interfacing design areas – Reactor Building, BOP, etc. Provide preliminary views of each installed system to confirm instrumentation installation points and verify operating conditions. Document design issues. (cont)	GA	FD 0-42mo	500

DDN(s) Supported: C.31.01.01, C.34.01.02	**Technology Case File:**

Subject Matter Expert Making Determination: Dale Pfremmer

Date: 10/23/08	**Originating Organization:** General Atomics

Additional Description Sheet(s)

Description:

SSC-15 instrumentation provides detection of primary coolant leakage through measurement of pressure, temperature or radiation levels within the Reactor Building or in helium piping which penetrates the Reactor Building. SSC-15 includes moisture monitoring and pressure instrumentation to detect steam in-leakage, provide operator information, and as a protection-logic input to the reactor-trip function. SSC-15 also includes helium flow rate instrumentation and radioactive plateout monitoring instrumentation.

Additional Basis Sheet(s)

Basis:

Subsystem testing will be determined by review of vendor development efforts. If subsystem testing is necessary, this type of testing is only expected for advanced instrumentation systems, such as the moisture monitoring and plateout probe systems. No subsystem testing is expected for the conventional instrumentation contained in the primary circuit and balance of plant. Achievement of the level 6 technical rating will be provided by analytical confirmation for this type of instrumentation.

Likewise, coordination with the circulator development team will determine the need for helium flow rate instrumentation testing. If necessary, this testing will be accomplished with other circulator subsystem testing, and will be conducted under the circulator development scope. It is likely that seismic testing of the helium flow rate measurement system will be more convenient if performed at the subsystem level.

SSC-15 will review nuclear control and electric-plant instrumentation development, but will require no testing to advance to a level 6 technical rating.

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Review vendor development of Steam Generator Moisture Ingress Detection Sensors and Plateout Probe instrumentation. Specify necessary subsystem testing, complete tests, and verify results. (No subsystem testing expected.) Advance analytical results to confirm level 6 technical rating.	GA Vendor Facility	FD 0-36mo FD 36-42mo FD 36-42mo	40 200 160
3. Review circulator subsystem testing activities to determine that helium flow measurement system is satisfactory to confirm the level 6 technical rating. Repeat or add testing, including seismic testing, if necessary. Provide supporting analysis. Document results to support verification of reactor flow rate measurement for the safety-related protection system.	GA Howden	FD 0-36mo FD 36-42mo	10 20
4. Review BOP electric-plant instrumentation development to verify accuracy, range, time of response, etc of BOP temperature, pressure, flow rate, etc. instrumentation. Provide supporting analysis and document results to verify reactor control capabilities and confirm ability of PCDIS to accomplish required actions following a reactor trip.	GA BOP	FD 0-42mo FD 0-42mo	50 20

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-15.4	**Revision:** 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Reactor Control and Protection, Primary Circuit and Balance of Plant Instrumentation (SSC-15)

Description:
SSC-15 contains instrumentation associated with the primary circuit and the balance of plant. Some of the instrumentation will be placed in the primary helium circuit or reactor building to detect leakage of radioactive materials, potentially affecting the public or plant personnel. Balance-of-Plant (BOP) measurements, comprised of steam flow rate, temperature, pressure, etc provide defense-in-depth protection of reactor cooling functions. Instrumentation within the primary circuit or included in the secondary boundary provided by the Reactor Building, is considered Primary Circuit instrumentation. (Cont.)

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Verified at experimental scale	Verified at pilot scale	Verified at engineering scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☒

The current level 6 technical rating for SSC-15 is based on completion of activities required to achieve a level 6 technical rating. These activities determined specific subsystem testing or, as an alternative, provided analytical confirmation of the technical level of the instrumentation. This provided information to complete the final design and perform any necessary system level instrumentation testing. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Complete final design. Issue final P&ID drawings for Primary Circuit and Balance of Plant Instrumentation. Coordinate with interfacing design areas – Reactor Building, BOP, etc. to verify pre-installation acceptance test planning and documentation to be completed. (Cont.)	GA	FD 42-84mo	300

DDN(s) Supported: C.31.01.01, C.34.01.02	**Technology Case File:**

Subject Matter Expert Making Determination: Dale Pfremmer

Date: 10/23/08	**Originating Organization:** General Atomics

Additional Description Sheet(s)

Description:

SSC-15 instrumentation provides detection of primary coolant leakage through measurement of pressure, temperature or radiation levels within the Reactor Building or in helium piping which penetrates the Reactor Building. SSC-15 includes moisture monitoring and pressure instrumentation to detect steam in-leakage, provide operator information, and as a protection-logic input to the reactor-trip function. SSC-15 also includes helium flow rate instrumentation and radioactive plateout monitoring instrumentation.

Additional Basis Sheet(s)

Basis:

Acceptance testing for vendor developed instrumentation and helium flow rate instrumentation will be completed. Acceptance testing will be conducted at vendor facility. Seismic testing will be completed, and instrumentation tested to assure compliance with SSE and OBE requirements. After delivery, tests will be repeated on-site to validate operation and compliance with as-built specifications. System mounting compatibility will also be confirmed. Results will be reviewed and testing will be repeated if equipment modifications are necessary. Advancement to the level 7 technical rating will be supported by analytical results.

Likewise, SSC-15 coordination with the circulator development team will be provided to determine engineering scale helium flow rate instrumentation testing. This testing will be accomplished with other circulator subsystem testing, and will be conducted under the circulator development scope. Seismic testing of helium flow rate measurement system will be completed (or level 6 seismic testing may be repeated, if necessary).

SSC-15 will monitor BOP instrumentation development testing to assure accuracy, reliability, maintainability, etc. of helium flow measurement and confirm defense-in-depth protection of nuclear/electric-plant reactor cooling capability. BOP will provide pre-installation checkout of instrumentation and documentation to advance to a level 7 technical rating.

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Assure updated analysis is provided to define accuracy, reliability, maintainability, etc. of all radiological leak detection instrumentation and for Steam Generator Moisture Ingress Detection Sensors and Plateout Probe instrumentation.	GA	FD 42-60mo	100
3. Fabricate instrumentation and monitor vendor acceptance testing. Complete seismic testing, including repeat of operational testing to assure compliance with SSE and OBE operational requirements. Document to confirm qualification of safety-related protection instrumentation.	GA Vendors GA	FD 60-78mo FD 72-80mo FD 80-84mo	100 1,600 200
4. Deliver instrumentation and repeat vendor acceptance tests on-site to validate operation.	GA	FD 82-84mo	50
5. Verify instrumentation mounting and cable installation capability.	GA Howden	FD 42-46mo FD 46-78mo	40 100
6. Provide circulator flow measurement test requirements. Combine helium flow rate measurement testing with circulator pre-installation acceptance testing. Provide updated analysis to assure accuracy, reliability, maintainability, etc. of helium flow measurement instrumentation is satisfactory for level 7 technical rating.	GA Howden	FD 72-78mo FD 72-78mo	50 60
7. Complete fabrication of circulator systems. Determine seismic testing which needs to be repeated (if not done previously at level 6) to assure compliance with SSE and OBE operational requirements. Document to confirm qualification of safety-related helium flow rate instrumentation.	GA BOP	FD 60-72mo FD 60-72mo	50 50
8. Review BOP electric-plant instrumentation pre-installation testing. Assure updated analysis is provided to define accuracy, reliability, maintainability, etc. of temperature, pressure, flow rate, etc. instrumentation. Assure seismic testing requirements have been completed. Document for qualification of safety-related protection instrumentation.			

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-15.5	**Revision:** 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Reactor Control and Protection, Primary Circuit and Balance of Plant Instrumentation (SSC-15)

Description:
SSC-15 contains instrumentation associated with the primary circuit and the balance of plant. Some of the instrumentation will be placed in the primary helium circuit or reactor building to detect leakage of radioactive materials, potentially affecting the public or plant personnel. Balance-of-Plant (BOP) measurements, comprised of steam flow rate, temperature, pressure, etc provide defense-in-depth protection of reactor cooling functions. Instrumentation within the primary circuit or included in the secondary boundary provided by the Reactor Building, is considered Primary Circuit instrumentation. (Cont.)

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Verified at pilot scale	Verified at engineering scale	Tested and Qualified
TRL	**6**	**7**	**8**

Basis for Rating (Check box if continued on additional sheets) ☒

The current level 7 technical rating for SSC-15 is based on completion of activities required to achieve a level 7 technical rating. These activities provided necessary pre-installation system testing of critical instrumentation required in NGNP, or analytical assessment to confirm the technical level of safety-related plant instrumentation. The level 7 effort provided confirmation to install this instrumentation. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Install primary circuit and balance of plant instrumentation – coordinate with Reactor Building, Circulator System, BOP, etc. to assure cable separation, instrumentation identification, wiring continuity, etc and provide documentation to validate installation process. (Cont.)	GA Vendors	FD 84-96mo FD 84-96mo	200 1,500

DDN(s) Supported: none	**Technology Case File:**

Subject Matter Expert Making Determination: Dale Pfremmer

Date: 10/23/08	**Originating Organization:** General Atomics

Additional Description Sheet(s)

Description:

SSC-15 instrumentation provides detection of primary coolant leakage through measurement of pressure, temperature or radiation levels within the Reactor Building or in helium piping which penetrates the Reactor Building. SSC-15 includes moisture monitoring and pressure instrumentation to detect steam in-leakage, provide operator information, and as a protection-logic input to the reactor-trip function. SSC-15 also includes helium flow rate instrumentation and radioactive plateout monitoring instrumentation.

Additional Basis Sheet(s)

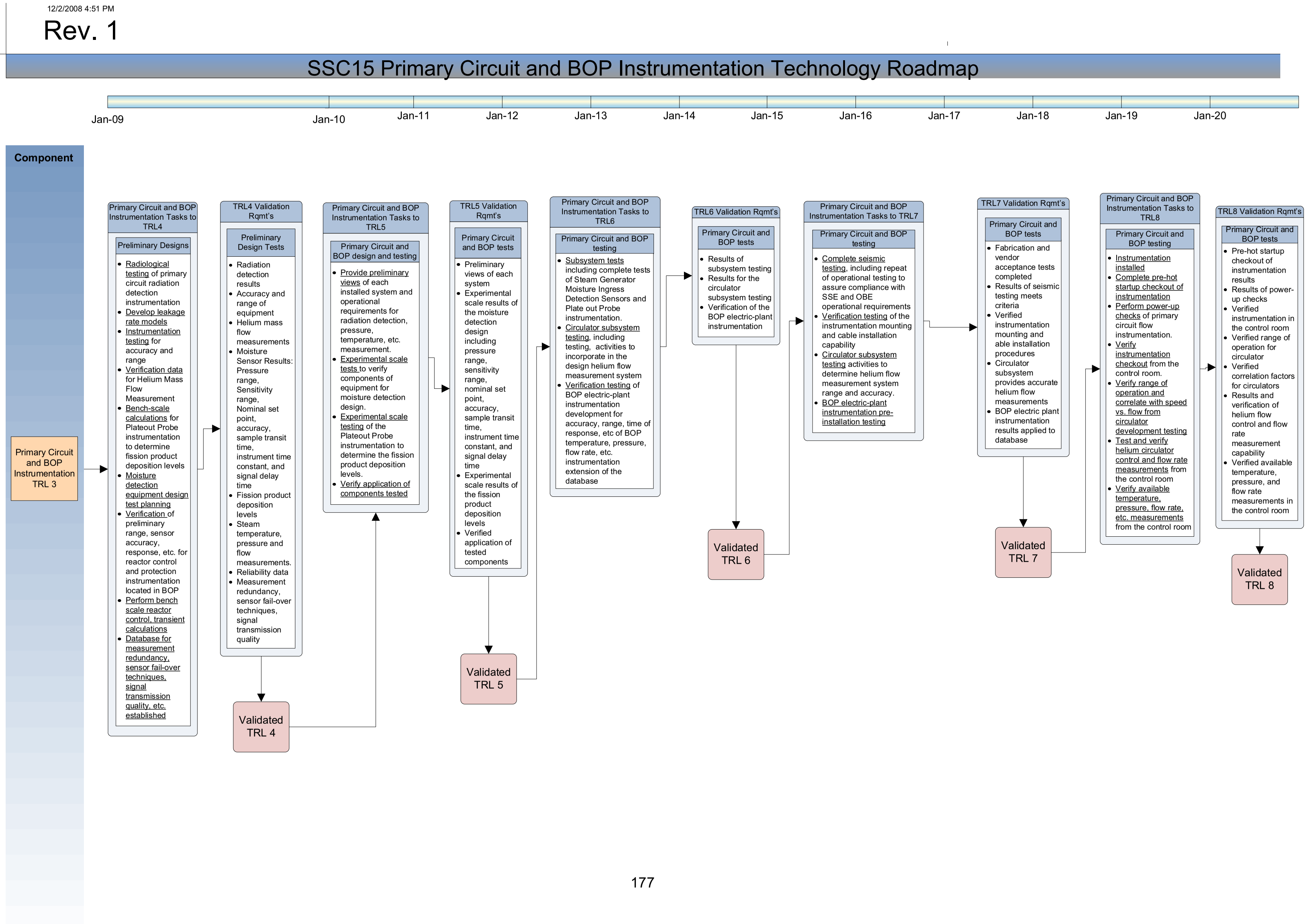
Basis:

Instrumentation checkouts to confirm hot startup readiness will be completed to achieve a level 8 technical rating. This includes checkout of primary circuit instrumentation, BOP instrumentation, and helium flow rate instrumentation. Advancement to the level 8 technical rating will complete qualification of safety-related and non-safety instrumentation for the reactor protection and investment protection functions.

Helium flow rate instrumentation checkout will be incorporated in pre-hot start circulator system checking, by pressurization of the vessel to (TBD) and subsequent operation of the circulators.

SSC-15 advancement to a level 8 technical rating will include validation of instrumentation functions from the control room.

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Complete pre-hot startup checkout of instrumentation. Perform power-up checks of primary circuit instrumentation. Verify instrumentation checkout from the control room.	GA Vendors	FD 84-96mo FD 84-108mo	100 200
3. Operate circulators and test helium flow rate instrumentation. Verify range of operation and correlate with speed vs. flow from circulator development testing. Verify helium circulator control and flow rate measurements from the control room.	GA	FD 96-108mo	100
4. Monitor BOP electric-plant instrumentation during BOP pre-hot startup readiness testing and verify available temperature, pressure, flow rate, etc. measurements from the control room. Confirm operator information and control functions associated with instrumentation.	GA	FD 96-108mo	100
5. Provide documentation supporting qualification of primary circuit and BOP instrumentation to confirm level 8 technical rating.	GA	FD 96-108mo	200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Rev. 1
SSC15 Primary Circuit and BOP Instrumentation Technology Roadmap
Jan-09
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Component
Primary Circuit and BOP Instrumentation TRL 3
Primary Circuit and BOP Instrumentation Tasks to TRL4
Preliminary Designs
• Radiological testing of primary circuit radiation detection instrumentation
• Develop leakage rate models
• Instrumentation testing for accuracy and range
• Verification data for Helium Mass Flow Measurement
• Bench-scale calculations for Plateout Probe instrumentation to determine fission product deposition levels
• Moisture detection equipment design test planning
• Verification of preliminary range, sensor accuracy, response, etc. for reactor control and protection instrumentation located in BOP
• Perform bench scale reactor control, transient calculations
• Database for measurement redundancy, sensor fail-over techniques, signal transmission quality, etc. established
TRL4 Validation Rqmt's
Preliminary Design Tests
• Radiation detection results
• Accuracy and range of equipment
• Helium mass flow measurements
• Moisture Sensor Results: Pressure range, Sensitivity range, Nominal set point, accuracy, sample transit time, instrument time constant, and signal delay time
• Fission product deposition levels
• Steam temperature, pressure and flow measurements.
• Reliability data
• Measurement redundancy, sensor fail-over techniques, signal transmission quality
Validated TRL 4
Primary Circuit and BOP Instrumentation Tasks to TRL5
Primary Circuit and BOP design and testing
• Provide preliminary views of each installed system and operational requirements for radiation detection, pressure, temperature, etc. measurement.
• Experimental scale tests to verify components of equipment for moisture detection design.
• Experimental scale testing of the Plateout Probe instrumentation to determine the fission product deposition levels.
• Verify application of components tested
TRL5 Validation Rqmt's
Primary Circuit and BOP tests
• Preliminary views of each system
• Experimental scale results of the moisture detection design including pressure range, sensitivity range, nominal set point, accuracy, sample transit time, instrument time constant, and signal delay time
• Experimental scale results of the fission product deposition levels
• Verified application of tested components
Validated TRL 5
Primary Circuit and BOP Instrumentation Tasks to TRL6
Primary Circuit and BOP testing
• Subsystem tests including complete tests of Steam Generator Moisture Ingress Detection Sensors and Plate out Probe instrumentation.
• Circulator subsystem testing, including testing, activities to incorporate in the design helium flow measurement system
• Verification testing of BOP electric-plant instrumentation development for accuracy, range, time of response, etc of BOP temperature, pressure, flow rate, etc. instrumentation extension of the database
TRL6 Validation Rqmt's
Primary Circuit and BOP tests
• Results of subsystem testing
• Results for the circulator subsystem testing
• Verification of the BOP electric-plant instrumentation
Validated TRL 6
Primary Circuit and BOP Instrumentation Tasks to TRL7
Primary Circuit and BOP testing
• Complete seismic testing, including repeat of operational testing to assure compliance with SSE and OBE operational requirements
• Verification testing of the instrumentation mounting and cable installation capability
• Circulator subsystem testing activities to determine helium flow measurement system range and accuracy.
• BOP electric-plant instrumentation pre-installation testing
TRL7 Validation Rqmt's
Primary Circuit and BOP tests
• Fabrication and vendor acceptance tests completed
• Results of seismic testing meets criteria
• Verified instrumentation mounting and able installation procedures
• Circulator subsystem provides accurate helium flow measurements
• BOP electric plant instrumentation results applied to database
Validated TRL 7
Primary Circuit and BOP Instrumentation Tasks to TRL8
Primary Circuit and BOP testing
• Instrumentation installed
• Complete pre-hot startup checkout of instrumentation
• Perform power-up checks of primary circuit flow instrumentation.
• Verify instrumentation checkout from the control room.
• Verify range of operation and correlate with speed vs. flow from circulator development testing
• Test and verify helium circulator control and flow rate measurements from the control room
• Verify available temperature, pressure, flow rate, etc. measurements from the control room
TRL8 Validation Rqmt's
Primary Circuit and BOP tests
• Pre-hot startup checkout of instrumentation results
• Results of power-up checks
• Verified instrumentation in the control room
• Verified range of operation for circulator
• Verified correlation factors for circulators
• Results and verification of helium flow control and flow rate measurement capability
• Verified available temperature, pressure, and flow rate measurements in the control room
Validated TRL 8

4.16 SSC-16 RPS, IPS, and PCDIS

TRL Rating Sheets, TRL 4 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-16.1	**Revision:** 2

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Reactor Control and Protection, RPS, IPS AND PCDIS (SSC-16)

Description:
SSC-16 contains the primary components of the Reactor Control and Protection systems. This necessarily includes determining and verifying the Plant Control Room layout, the operational and safety interfaces, remote shutdown facilities, plant-wide distribution of control and protection functions, and the overall plant control architecture for effective, reliable plant operation. SSC-16 includes development of the reactor control and protection algorithms, which require verification at preliminary and latter stages of the design. Since it is quite likely that the plant control architecture and the operator interface will employ modern digital hardware and software, SSC-16 also includes the necessary testing and qualification to assure reliability and safety with this type of equipment.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Proof of concept	Verified at bench scale	Verified at experimental scale
TRL	**3**	**4**	**5**

Basis for Rating (Check box if continued on additional sheets) ☒

The initial level 4 technical rating for SSC-16 relies primarily on work to develop a similar control and protection configuration for the New Production Reactor (NPR) program in the early 90s at General Atomics. This work established the control architecture for the NPR plant using modern digital hardware and software. Conceptual designs were completed for NPR protection and control systems. The NPR work and other MHR control development efforts justify an initial technical rating of 4 because the NPR project completed trade-off studies to define top level requirements for control room layout, plant control architecture, utilization of digital equipment and software for operator interactions, capability for multi-function plant control and safety, etc. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Complete conceptual design engineering. Determine plant control and protection scheme. Determine preliminary testing. Determine development simulator scope and requirements. Develop models. Document. (Cont.)	GA	CD 0-36mo	2,400

DDN(s) Supported: C34.02.02.01, C.34.02.02.02	**Technology Case File:**

Subject Matter Expert Making Determination: Dale Pfremmer

Date: 10/23/08 | **Originating Organization:** General Atomics

Additional Basis Sheet(s)
Basis: The approach provided by this work will be followed as the basis for starting the NGNP conceptual design effort. During the initial phase of the NGNP conceptual design, development (SSC-16) of plant control algorithms, calculations to verify the preliminary control/protection design specifically for NGNP multi-function plant operation, etc. must be completed. This requires development of a real-time simulator, which in-turn supports level 5 testing to verify preliminary operator interaction and control methods. The simulator supports acceptance testing of RPS, IPS and PCDIS equipment and software, and will be used at a higher technical rating to test the as-built, interconnected Reactor Control and Protection systems equipment. Other testing to complete level 5 readiness will confirm reliability assumptions provided by digital equipment hardware and software manufacturers.

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Use vendor supplied equipment to perform experimental scale testing of safety, protection, and control failover methods, signal noise tolerance, etc in simulated equipment operating and placement configurations. Test digital equipment robustness, susceptibility to common-mode failure, etc. considering single and multiple failure cases to confirm the reliability design for RPS, IPS and PCDIS equipment, under scenarios of operation, maintenance, etc. If necessary, provide experimental scale verification of the 2-out-of-4 redundancy scheme for safety and protection equipment. Combine test results with conceptual design verification of electric power availability, and other BOP functions critical to Reactor Control and Protection reliability. Document level 5 rating for safety-related digital computer equipment and non-safety protection and control equipment. Provide recommendations for later pilot scale testing.	GA Vendor 1 Vendor 2 Vendor 3	CD 24-36mo CD 24-36mo CD 24-36mo CD 24-36mo	600 500 500 500
3. Test the preliminary operator interface using the real-time simulator at the experimental scale to evaluate critical aspects of interactive plant operation, control automation strategy, information recovery strategy, etc. Include requirements for operability and safety from NGNP participants in other design areas as well, to confirm the overall conceptual design features of the operator displays. Document necessary plant control interface testing requirements for testing activities in other NGNP design areas needed by the reactor Control and Protection systems at the next technical rating levels (level 6 or level 7). For example, circulator motor control testing will be required to verify assumptions made during conceptual design to develop PCDIS reactor flow control algorithms.	GA Vendor 4	CD 24-36mo CD 24-36mo	600 400
4. Update conceptual design Reactor Control and Protection systems analysis results to confirm preliminary design readiness. Obtain preliminary review of licensability, and document issues.	GA	CD 30-36mo	900

TRL Rating Sheet

Vendor:	GA	**Document Number:**	SSC-16.2	**Revision:**	1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Reactor Control and Protection, RPS, IPS AND PCDIS (SSC-16)

Description:
SSC-16 contains the primary components of the Reactor Control and Protection systems. This necessarily includes determining and verifying the Plant Control Room layout, the operational and safety interfaces, remote shutdown facilities, plant-wide distribution of control and protection functions, and the overall plant control architecture for effective, reliable plant operation. SSC-16 includes development of the reactor control and protection algorithms, which require verification at preliminary and latter stages of the design. Since it is quite likely that the plant control architecture and the operator interface will employ modern digital hardware and software, SSC-16 also includes the necessary testing and qualification to assure reliability and safety with this type of equipment.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Verified at pilot scale	Verified at experimental scale	Verified at pilot scale
TRL	**4**	**5**	**6**

Basis for Rating (Check box if continued on additional sheets) ☒

The level 5 rating for SSC-16 is based on completion of activities required to achieve a level 5 technical rating. This work included testing of operator interfaces to control plant processes using experimental scale updates of the real-time simulator. Additional experimental scale equipment tests were performed to provide reliability data not included in the original equipment supplier data. Reactor control and protection analysis results confirmed preliminary design readiness. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Complete preliminary final design engineering. Verify plant control and protection scheme. Verify PCDIS subsystems. Oversee and obtain testing results. Update development simulator requirements for system checkout testing. Finalize development models. Document.	GA	FD 0-42mo	1,300

DDN(s) Supported: C34.02.02.01, C.34.02.02.02, C.31.02.01.01

Technology Case File:

Subject Matter Expert Making Determination: Dale Pfremmer

Date: 10/23/08 **Originating Organization:** General Atomics

Additional Basis Sheet(s)
Basis: To advance to a level 6 technical rating, additional testing must be completed to confirm data and control signal transfer rates, and other aspects of the design. Preliminary Design (PD) plant-total instrumentation and control equipment estimates from each of the BOP, NHSS, etc. design areas will be needed to establish test requirements. Vendor supplied equipment will be used. RPS, IPS, and PCDIS data-highway communication capacity, considering the PD data-highway hierarchy within the combined structure of these systems and their interfacing plant systems will be tested. Equipment tests to verify storage, formatting, and on-line retrieval of stored data for use in trend displays, tech spec information displays, safety-console information displays, and other critical operator information displays, will be included. Also, tests to verify the reliability of Reactor Control and Protection equipment operating in locations outside the control room must be included. The level 6 rating will require circulator test data and updates of the control development simulator facility to test the PCDIS reactor flow control algorithms. RPS, IPS and PCDIS acceptance tests (at level 7) will also be based, in part, on these tests.

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Confirm information transfer rates by pilot scale testing of representative digital equipment configurations using vendor supplied hardware and software to drive communication functions. Test preliminary specification of data-highway(s) transmission capacity and information hierarchy. Resolve issues of transfer speed, data loss, synchronization, etc. to confirm readiness to begin RPS, IPS, and PCDIS final design equipment specifications.	GA Vendor 1	FD 24-42mo FD 30-42mo	200 1,400
3. Develop pilot scale facilities for RPS, IPS, and PCDIS plant-distributed control and instrumentation equipment testing, using vendor supplied equipment. Address operating lifetime, on-line maintenance access, and other issues requiring placement specific test data not available from prospective equipment vendors. Where necessary, provide separate pilot scale test configurations for RPS/IPS protection systems and PCDIS control systems to separate safety licensing issues during this testing. Verify channel separation, isolation from non-safety equipment, failed-channel operation, etc. for RPS and IPS to obtain preliminary confirmation of licensability necessary to issue final design procurement specifications for vendor supplied equipment. Issue requests for necessary Reactor Control and Protection testing required in other NGNP design areas, such as communication signal noise environment, temperature/humidity/pressure environment, motion/vibration environment, electrical quality, cooling quality, etc. needed by the Reactor Control and Protection systems to issue the final design specifications. Combine the test results, resolve issues, and document overall results of pilot scale equipment testing to confirm final design readiness.	GA Vendor 1 Vendor 2 GA Vendor 4 GA	FD 30-42mo FD 12-42mo FD 12-42mo FD 30-42mo FD 36-42mo FD 12-42mo	400 1,200 500 200 800 900
4. Procure checkout interfaces for development simulator.			
5. Recommend testing procedures to obtain data not available from previous circulator development tests in order to confirm or update previous PCDIS design assumptions with respect to circulator flow vs. circulator motor speed over the NGNP operating range. Update control development simulator utilizing test specific representation of circulator motor/speed control. Repeat helium flow control algorithm development tests to assure that helium flow control by	GA	FD 36-42mo	1,000

means of variable frequency circulator motor speed controllers will not invoke limit-cycling or cause unexpected interaction with commands from the PCDIS. Resolve issues and update PCDIS algorithm design documentation. 6. Provide reactor control and protection analysis results to confirm final design readiness.			

TRL Rating Sheet

Vendor: GA	**Document Number:** SSC-16.3	**Revision:** 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Reactor Control and Protection, RPS, IPS AND PCDIS (SSC-16)

Description:
SSC-16 contains the primary components of the Reactor Control and Protection systems. This necessarily includes determining and verifying the Plant Control Room layout, the operational and safety interfaces, remote shutdown facilities, plant-wide distribution of control and protection functions, and the overall plant control architecture for effective, reliable plant operation. SSC-16 includes development of the reactor control and protection algorithms, which require verification at preliminary and latter stages of the design. Since it is quite likely that the plant control architecture and the operator interface will employ modern digital hardware and software, SSC-16 also includes the necessary testing and qualification to assure reliability and safety with this type of equipment.

Area:	☒ NHSS	☐ HTS	☐ HPS	☐ BCS	☐ BOP

ASSCT:	**Parent:**	**WBS:**

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Verified at experimental scale	Verified at pilot scale	Verified at engineering scale
TRL	**5**	**6**	**7**

Basis for Rating (Check box if continued on additional sheets) ☒

The level 6 rating for SSC-16 is based on completion of activities required to achieve a level 6 technical rating. This work included pilot scale testing of plant-distributed equipment, signal communications, life time under operating conditions, etc. and compliance with safety-related regulatory requirements for channel separation, etc. The PCDIS control design was also updated to include information derived from circulator development testing. (Cont)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Complete final design engineering. Oversee and support procurement of IPS, RPS, and PCDIS equipment. Support and oversee all acceptance testing. Provide development simulator for testing. Validate plant control and protection scheme. Document. (Cont.)	GA	FD 42-84mo	1,500

DDN(s) Supported: C.31.02.01.01, C.33.01.01.01	**Technology Case File:**

Subject Matter Expert Making Determination: Dale Pfremmer

Date: 10/23/08	**Originating Organization:** General Atomics

Additional Basis Sheet(s)
Basis: Reactor control and protection analysis results verified software embedded reactor control algorithms and confirmed final design readiness. This provided the basis for the Reactor Control and Protection systems procurement specifications, as well as parallel development of a full-scope training simulator. To advance to a level 7 rating, additional testing must be completed to finalize the operator and hardware interfaces for RPS, IPS, and PCDIS. Software validation acceptance test procedures must be developed and completed, and engineering scale testing must be performed to validate the as-built Reactor Control and Protection systems hardware and software, and to confirm RPS, IPS, and PCDIS installation readiness. The level 7 rating will also require seismic testing of SSC-16 systems before installation can be completed.

Additional Action Sheet(s)

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Procure RPS, IPS and PCDIS equipment.	GA Vendor 1 Vendor 2	FD 42-78mo FD 60-78mo FD 60-78mo	800 30,000 16,000
3. Develop on-site engineering scale equipment test configurations, and procedures, to confirm installation checkout capabilities, online and offline maintenance capabilities, etc. using duplicate equipment supplied by vendor. Complete these tests and verify that all adjustments are made by the vendor(s) before delivery.	GA Vendor 1 Vendor 2	FD 42-60mo FD 48-60mo FD 48-60mo	900 1,000 800
4. Configure (or duplicate) the control development simulator to provide final engineering scale testing of RPS, IPS and PCDIS equipment. Determine testing to validate software design for combined RPS, IPS and PCDIS operator control and plant information interfaces. Determine plant control and protection systems testing requirements and prepare combined (and separate) RPS, IPS and PCDIS acceptance test procedures to be performed by the vendor(s) and verified before acceptance of equipment. Validate as-built Reactor Control and Protection systems software and equipment. Complete pre-delivery acceptance tests and post-delivery on-site acceptance tests to confirm installation readiness.	GA Vendor 1	FD 60-78mo FD 72-84mo	500 200
5. Complete seismic qualification tests and issue final report to confirm installation readiness.	GA Vendor 5	FD 78-84mo FD 78-84mo	100 3,000
6. Provide final reactor control and protection analysis results to confirm installation readiness and finalize Reactor Control and Protection systems licensing acceptance process.	GA	FD 80-84mo	200

TRL Rating Sheet

Vendor: GA	Document Number: SSC-16.4	Revision: 1

☐ Area ☐ System ☐ Subsystem/Structure ☒ Component ☐ Technology

Title: Reactor Control and Protection, RPS, IPS AND PCDIS (SSC-16)

Description:
SSC-16 contains the primary components of the Reactor Control and Protection systems. This necessarily includes determining and verifying the Plant Control Room layout, the operational and safety interfaces, remote shutdown facilities, plant-wide distribution of control and protection functions, and the overall plant control architecture for effective, reliable plant operation. SSC-16 includes development of the reactor control and protection algorithms, which require verification at preliminary and latter stages of the design. Since it is quite likely that the plant control architecture and the operator interface will employ modern digital hardware and software, SSC-16 also includes the necessary testing and qualification to assure reliability and safety with this type of equipment.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ BCS ☐ BOP

ASSCT:	Parent:	WBS:

Technology Readiness Level

	Next Lower Rating Level	Current Rating Level	Next Higher Rating Level
Generic Definitions *(abbreviated)*	Verified at pilot scale	Verified at engineering scale	Tested and qualified
TRL	**6**	**7**	**8**

Basis for Rating (Check box if continued on additional sheets) ☒

The level 7 rating for SSC-16 is based on completion of activities required to achieve a level 7 technical rating. This work completed all pre-delivery and post-delivery acceptance testing of RPS, IPS, and PCDIS equipment and validation of the installed software. Final reactor control and protection analysis results confirmed installation readiness and provided the final licensing acceptance process for Plant Control and Protection systems. (Cont.)

Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☒

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
1. Install Plant Control and Protection systems and complete reconnection checkout procedures for equipment moved from pre-installation checkout locations or which have been reconnected (from the level 7 simulation configuration) for plant operation. (Cont.)	GA Vendor 1 Vendor 2	FD 84-96mo FD 84-96mo FD 84-96mo	1,500 1,000 500

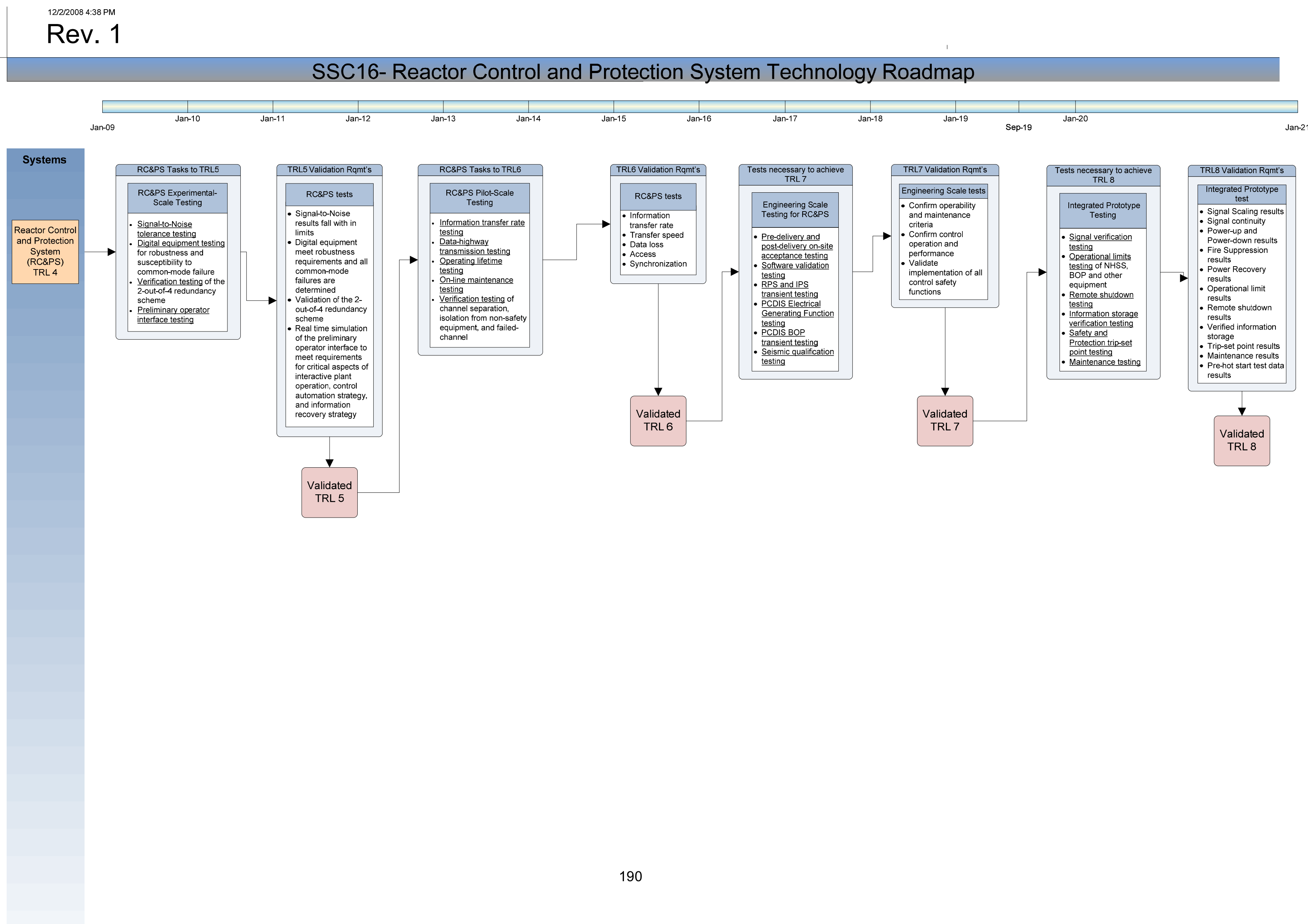
DDN(s) Supported: none | **Technology Case File:**

Subject Matter Expert Making Determination: Dale Pfremmer

Date: 10/23/08 | **Originating Organization:** General Atomics

Additional Basis Sheet(s)
Basis: To advance to a level 8 rating, the Plant Control and Protection systems must be installed and reconnected (from validation test configurations to operational configurations), using moderate test procedures to validate this process. The level 8 rating will also require flow and equipment operation testing as required by other systems and by other Reactor Control and Protection SSCs. Therefore, other systems must be installed and connected to test the Reactor Control and Protection systems.

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Verify signal communications, signal scaling and continuity, power-up and power-down features, fire-suppression and other equipment protection features, power failure recovery features, etc. Complete QA check-off procedures to validate final configuration of hardware and software. Complete tests of NHSS, BOP, etc. equipment (dependent on Reactor Control and Protection systems) as allowed within limits of prior-to-hot-startup operational capabilities. Repeat QA and testing for issues requiring resolution. Document SSC-16 final status for hot startup readiness. 2. Check vessel pressurization equipment and pressurize vessel to (TBD). Operate circulators, and verify helium flow control capabilities. Complete other operation and instrumentation tests, including operator information and procedures, control room supervisory information and procedures, and information storage verification tests; remote shutdown facility tests; BOP and Hydrogen Plant control system checks; safety and protection trip-setpoint tests; maintenance tests; etc. to verify hot-startup readiness. Update the SSC-16 status. Include off-line analysis, from the simulator, comparing expected control and protection test results and actual results, to confirm hot startup readiness. Verify regulatory acceptance of Reactor Control and Protection systems hot startup readiness at level 8.	GA Vendor 1 Vendor 2	FD 96-108mo FD 96-108mo FD 96-108mo	1,100 400 200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Rev. 1
SSC16- Reactor Control and Protection System Technology Roadmap
Jan-09
Jan-10
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Jan-17
Jan-18
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Sep-19
Jan-20
Jan-21
Systems
Reactor Control and Protection System (RC&PS) TRL 4
RC&PS Tasks to TRL5
RC&PS Experimental-Scale Testing
Signal-to-Noise tolerance testing
Digital equipment testing for robustness and susceptibility to common-mode failure
Verification testing of the 2-out-of-4 redundancy scheme
Preliminary operator interface testing
TRL5 Validation Rqmt's
RC&PS tests
Signal-to-Noise results fall with in limits
Digital equipment meet robustness requirements and all common-mode failures are determined
Validation of the 2-out-of-4 redundancy scheme
Real time simulation of the preliminary operator interface to meet requirements for critical aspects of interactive plant operation, control automation strategy, and information recovery strategy
Validated TRL 5
RC&PS Tasks to TRL6
RC&PS Pilot-Scale Testing
Information transfer rate testing
Data-highway transmission testing
Operating lifetime testing
On-line maintenance testing
Verification testing of channel separation, isolation from non-safety equipment, and failed-channel
TRL6 Validation Rqmt's
RC&PS tests
Information transfer rate
Transfer speed
Data loss
Access
Synchronization
Validated TRL 6
Tests necessary to achieve TRL 7
Engineering Scale Testing for RC&PS
Pre-delivery and post-delivery on-site acceptance testing
Software validation testing
RPS and IPS transient testing
PCDIS Electrical Generating Function testing
PCDIS BOP transient testing
Seismic qualification testing
TRL7 Validation Rqmt's
Engineering Scale tests
Confirm operability and maintenance criteria
Confirm control operation and performance
Validate implementation of all control safety functions
Validated TRL 7
Tests necessary to achieve TRL 8
Integrated Prototype Testing
Signal verification testing
Operational limits testing of NHSS, BOP and other equipment
Remote shutdown testing
Information storage verification testing
Safety and Protection trip-set point testing
Maintenance testing
TRL8 Validation Rqmt's
Integrated Prototype test
Signal Scaling results
Signal continuity
Power-up and Power-down results
Fire Suppression results
Power Recovery results
Operational limit results
Remote shutdown results
Verified information storage
Trip-set point results
Maintenance results
Pre-hot start test data results
Validated TRL 8

5 TECHNOLOGY DEVELOPMENT SCHEDULES

Figure 5-1 provides an overall technology development schedule that shows all of the technology development activities identified in the TDRMs and Test Plans for all of the critical SSC. This schedule was compiled from the schedule information provided in the Test Plans. A second schedule that includes just the testing identified in the Test Plans as potentially being performed in the CTF is provided in Figure 5-2. It is important to note that most, if not all of the tests for which the CTF has been identified as a potential location for the test could be done elsewhere should the CTF not be available. However, assuming that the CTF is built and is available, it would be a logical location for performing the tests identified in Figure 5-2.

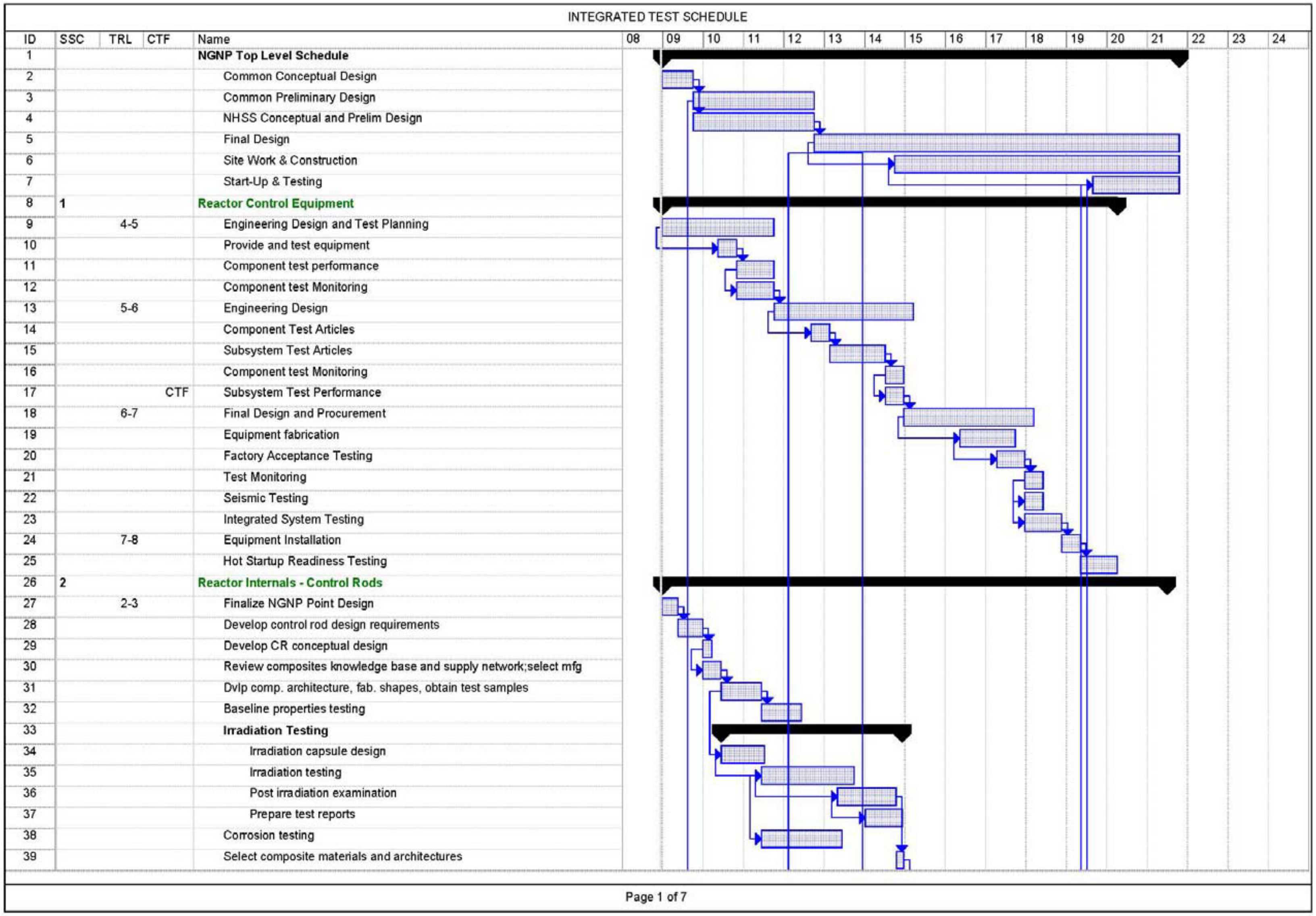

Figure 5-1. Overall Schedule for NGNP Technology Development

INTEGRATED TEST SCHEDULE

ID	SSC	TRL	CTF	Name
40		3-4		Develop final comp. architecture and fab. shapes
41				Baseline properties testing for engineering data base
42				Irradiation testing for engineering data base
43				Corrosion testing for engineering data base
44				Complete CR composites materials models
45		4-5		Engineering analysis
46		5-6		Engineering analysis
47		**6-7**		**CR vibration testing**
48				Develop test specification
49				Fab. prototype control rod
50				Set up test facility and prepare test procedures
51			CTF	Conduct CR vibration testing and prepare report
52				CR shock absorber test
53				**CR structural integrity test**
54				Develop test specification
55				Fab. prototype control rod
56				Set up test facility and prepare test procedures
57			CTF	Conduct CR structural integrity test and prepare report
58		7-8		Integrated NCA testing at NGNP
59	3			**High Temperature Ducts & Insulation**
60		2-3		Condition of Service Documented; safety, code and standards, mat'ls
61		3-4		Analytical Modeling
62		4-5		Component & Coupon testing
63		5-6		Subsystem Testing & Analysis
64		6-7		Scale System-Level Vessels tests
65		7-8	CTF	CTF testing (if required)
66	4			**Reactor Core & Graphite**
67	4a		**CTF**	**Reactor Core**
68			CTF	Core Element Dynamic Strength
69			CTF	Core Element Failure Modes
70			CTF	Fuel Element Flow Channel Data
71			CTF	Control Rod Flow Channel Data
72			CTF	Core Column Vibration Data
73			CTF	Core Crossflow Test Data
74			CTF	Core Fluctuation Test Data
75			CTF	Bottom Reflector/Core Support Pressure Drop/Flow Mixing Data
76			CTF	Metallic Plenum Element & Top Reflector Pressure Drop/Flow Distribu
77	4b			**Graphite**
78				Graphite Coke Source Qualification

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Figure 5-1. Overall Schedule for NGNP Technology Development (2 of 7)

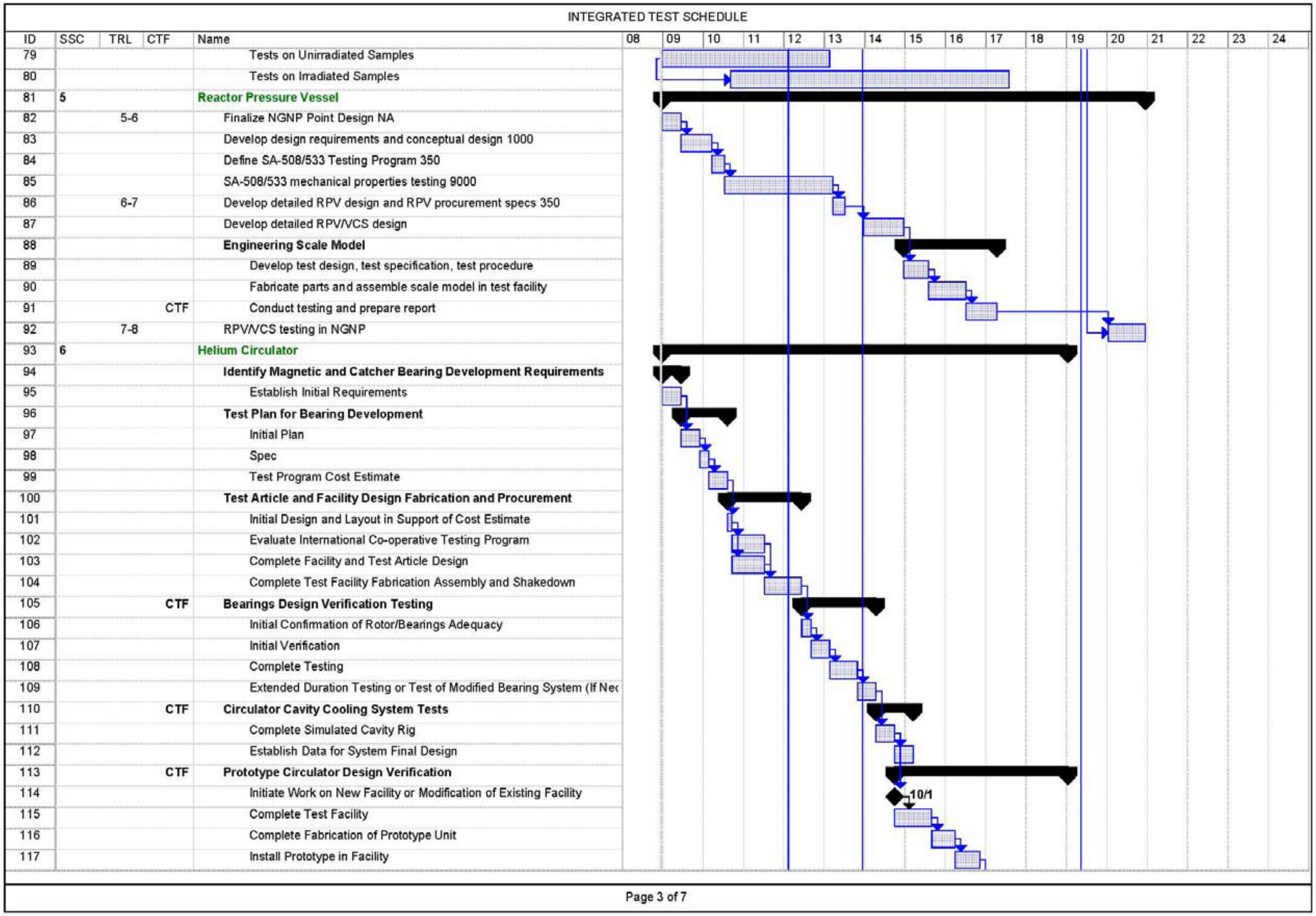

Figure 5-1. Overall Schedule for NGNP Technology Development (3 of 7)

INTEGRATED TEST SCHEDULE

ID	SSC	TRL	CTF	Name
118				Initial Confirmation that Circulator Meets Requirements
119				Complete Initial Test Verification
120				Extended Duration Testing
121				Test Modified MC If Necessary
122				Proof Test Additional Circulators in Support of First Plant Operation
123	7			**Intermediate Heat Exchanger**
124		2-3		Modelling to confirm IHX feasibility
125				Environmental aging and thermal cycling testing on Alloy 617
126				Verify diffusion bonding and chemical etching processes
127		3-4		Construct and test bench-scale PCHE module
128		4-5		Mechanical properties and environmental testing of samples from bench-s
129				IHX vessel design and experimental-scale testing
130		5-6		Computer modeling to complete IHX preliminary design
131		6-7	CTF	Design support testing on engineering-scale mockup(s)
132				Finalize IHX design and issue procurement specs
133		7-8		Construct prototype IHX
134			CTF	Test prototype IHX in NGNP operating environment
135	8			**Shutdown Cooling System Heat Exchanger**
136		4-5		**Design & Analysis**
137				Thermal Design & Analysis of SCHE
138				Structural Analysis
139				Selection of Shroud Material and Shroud Insulation
140		5-6		**Structural & Properties Tests**
141				Shroud and its Insulation Properties Tests
142				Shroud Leak Test
143				Vibration and Acoustical Tests
144				Related SG testing
145				Inlet Flow and/or Temperature Distribution Tests
146				Stability Oriface
147				SCHE Final Design
148				Build a Full Scale SCHE (est=$6M)
149		6-7	CTF	Airflow Testing for Full-Scale SCHE
150		7-8	CTF	Full-Scale SCHE Test in Helium
151	9			**Reactor Cavity Cooling System**
152		4-5		Emissivity of RCCS Panel Surfaces
153		5-6		Wind Tunnel Testing of Scale Model RCCS I/O Structure
154		6-7		RCCS Cooling Panel Heat Transfer and Friction Factor and Integrated Tes
155		7-8		Integrated RCCS Performance
156	10			**Steam Generator**

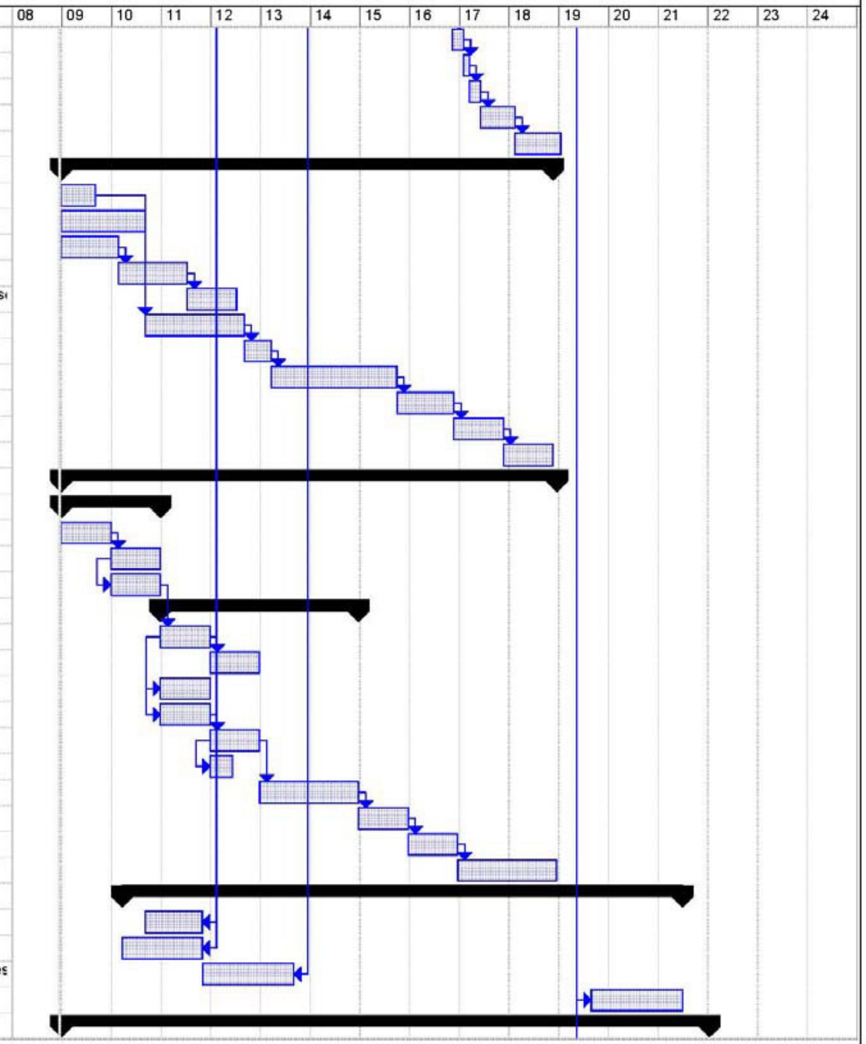

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Figure 5-1. Overall Schedule for NGNP Technology Development (4 of 7)

INTEGRATED TEST SCHEDULE

ID	SSC	TRL	CTF	Name
157		3-4		SG Design & Analysis
158				High Temperature Materials Properties (950C)
159		5-6		Base Material and bimetallic weld test (750C)
160				Air Flow Testing and Oriface testing
161				Lead In Lead Out testing
162				Tube Support System Test
163				Insulation testing
164				Large Helical Coil Fabrication testing
165		6-7		SG Final Design and Fabrication
166				Air Flow Testing
167		7-8		Testing in Helium
168	11			**TurboMachinery (for combined cycle PCS)**
169		4-5-6		**Full-scale Turbo-compressor**
170				TC, generator materials validation for helium
171				Shaft seals
172				aerodynamics performance testing
173		4-5-6		**Generator**
174				Insulation and lead-outs testing
175				generator operations testing with models
176				generator build
177		4-5-6		**EM Bearings**
178				electromagnetic suspension system
179				controls testing
180				rotor testing
181				friction tests
182				diaphragm coupling
183				**Full-Scale Testing & Integration**
184		6-7		Turbine and compressor builds
185		7-8		TM integration
186				PCU Prototype Testing
187	12			**High Temperature Valves**
188		3-4		Bench-scale TDP (matl): conditions of service, safety, code & standards
189		4-5		Analytical Modeling of Components Qualification
190		5-6		Pilot-scale component testing
191		6-7		Complete Valve Sub-System Integrated Experimental Scale Model Test
192		7-8	CTF	Integrated CTF Testing
193	13			**S-I Hydrogen Production System**
194		5-6		**Task 1: Pilot Plant**
195				Conceptual Design

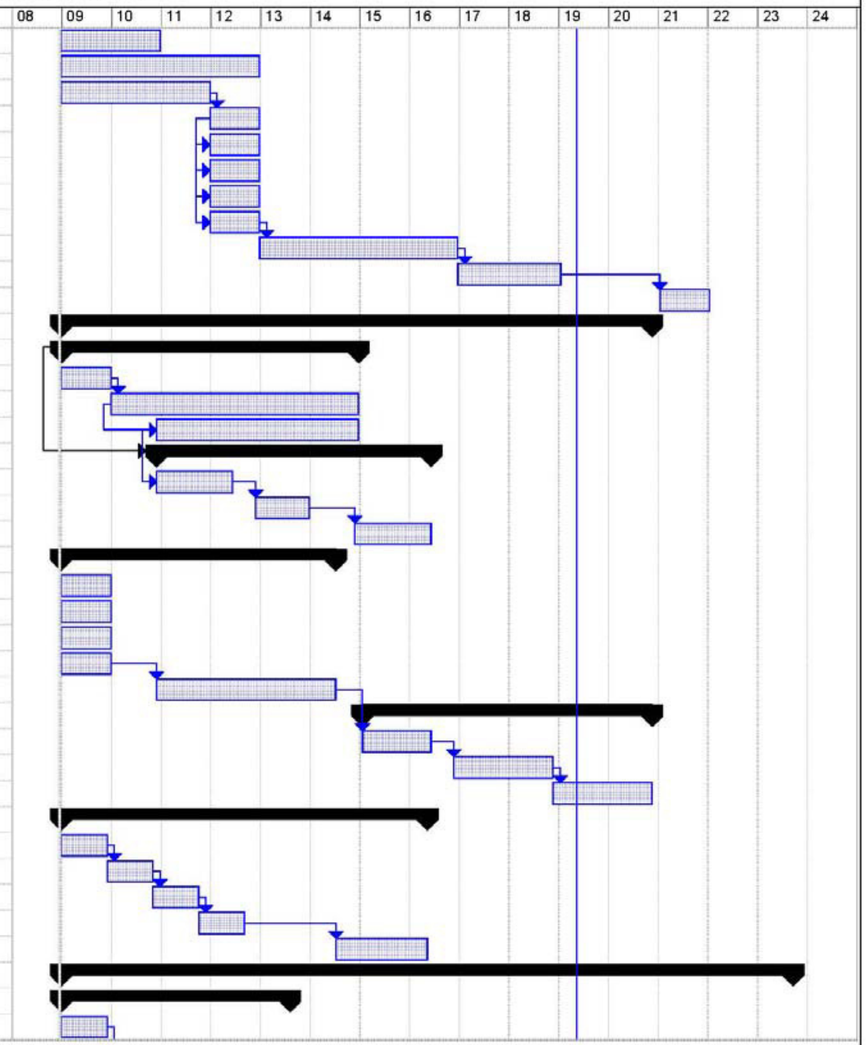

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Figure 5-1. Overall Schedule for NGNP Technology Development (5 of 7)

INTEGRATED TEST SCHEDULE

ID	SSC	TRL	CTF	Name
196				Preliminary Design
197				Validation
198				Final Design
199				Construction
200				Operation
201		6-7		**Task 2: Engineering Demonstration Plant**
202				Conceptual Design
203				Preliminary Design
204				Validation
205				Final Design
206			CTF	Construction
207			CTF	Operation
208				**Task 3: Prototype Plant**
209				Conceptual Design
210				Preliminary Design
211				Validation
212				Final Design
213				Construction
214				Operation
215	14			**Fuel Handling and Storage System**
216		4-5		Prepare conceptual design for NGNP FHSS
217				Survey supply network and select vendors
218				Complete FHSS preliminary design
219				Perform I&C testing
220				Perform testing of Fuel Handling Equip. Support Structure
221				Perform testing of Fuel Sealing and Inspection Facility
222		5-6		Perform testing of Fuel Handling Equipment Positioner
223				Perform Testing of Equipment Hoist and Grapple Assembly
224				Perform testing of Fuel Handling Machine
225				Perform testing of Fuel Transfer Cask
226				Complete final design of FHSS and procurement specs
227		6-7		Integrated FHSS
228				Prepare test specification
229				Fabricate prototype equipment and test rig
230				Set up test facility and equipment
231				Conduct testing
232				Evaluate test data and complete test report
233		7-8		FHSS testing in NGNP
234	15			**Primary Circuit and BOP Protection Instrumentation**

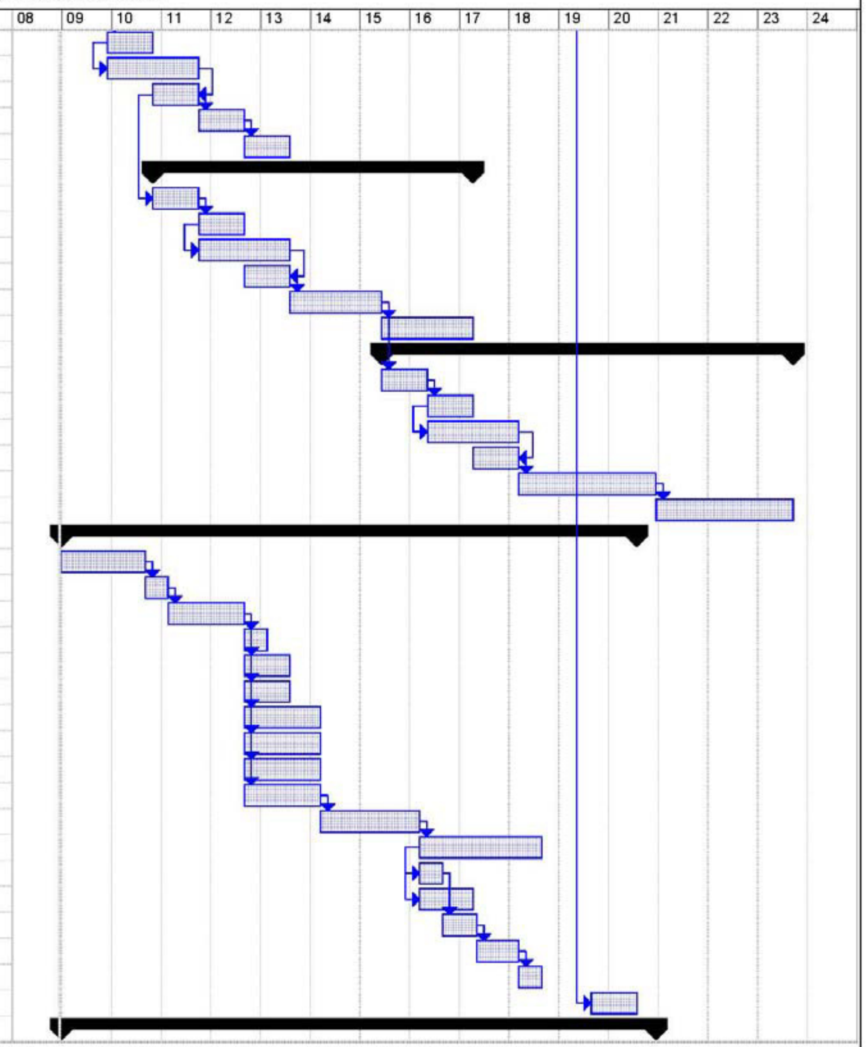

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Figure 5-1. Overall Schedule for NGNP Technology Development (6 of 7)

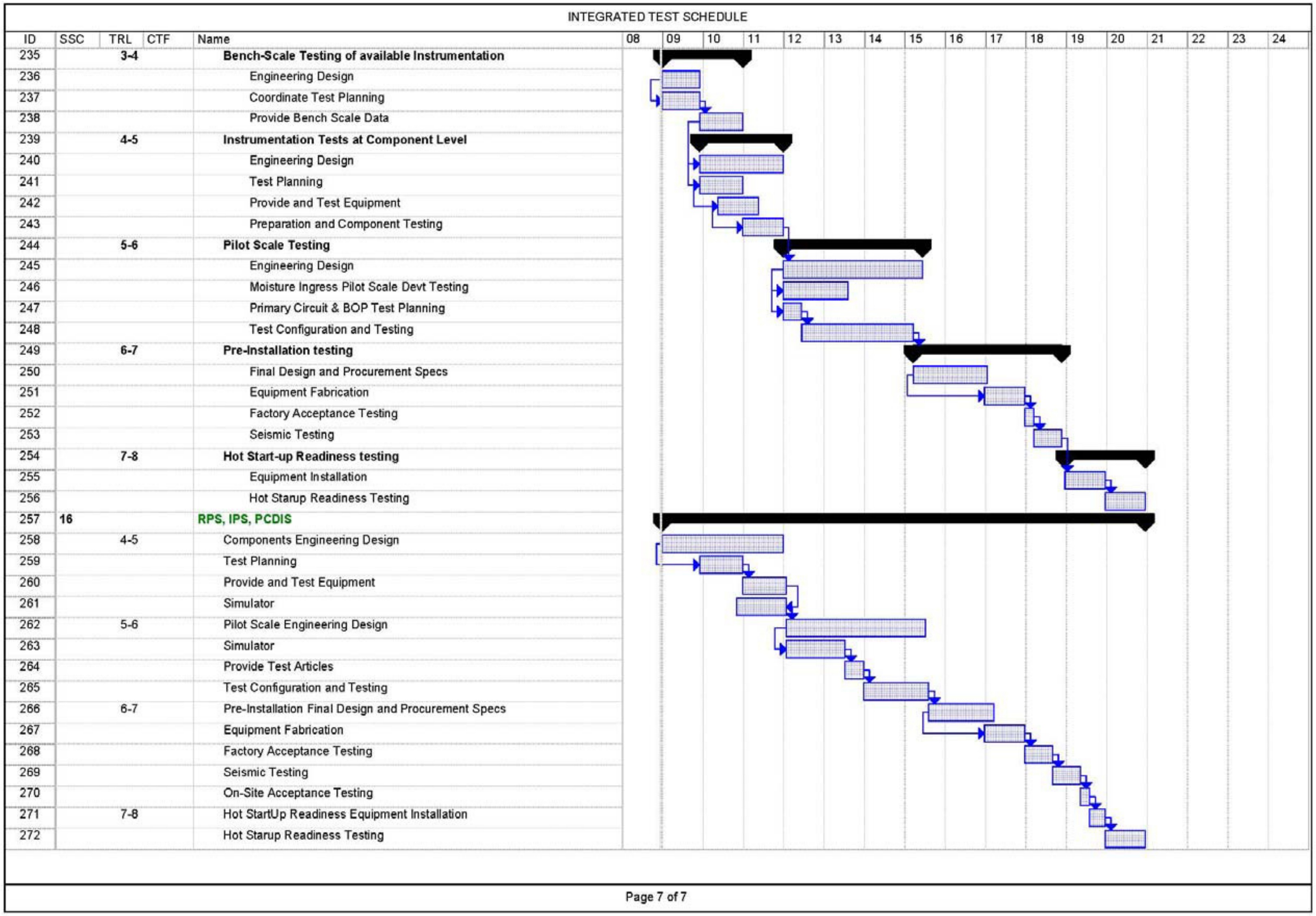

Figure 5-1. Overall Schedule for NGNP Technology Development (7 of 7)

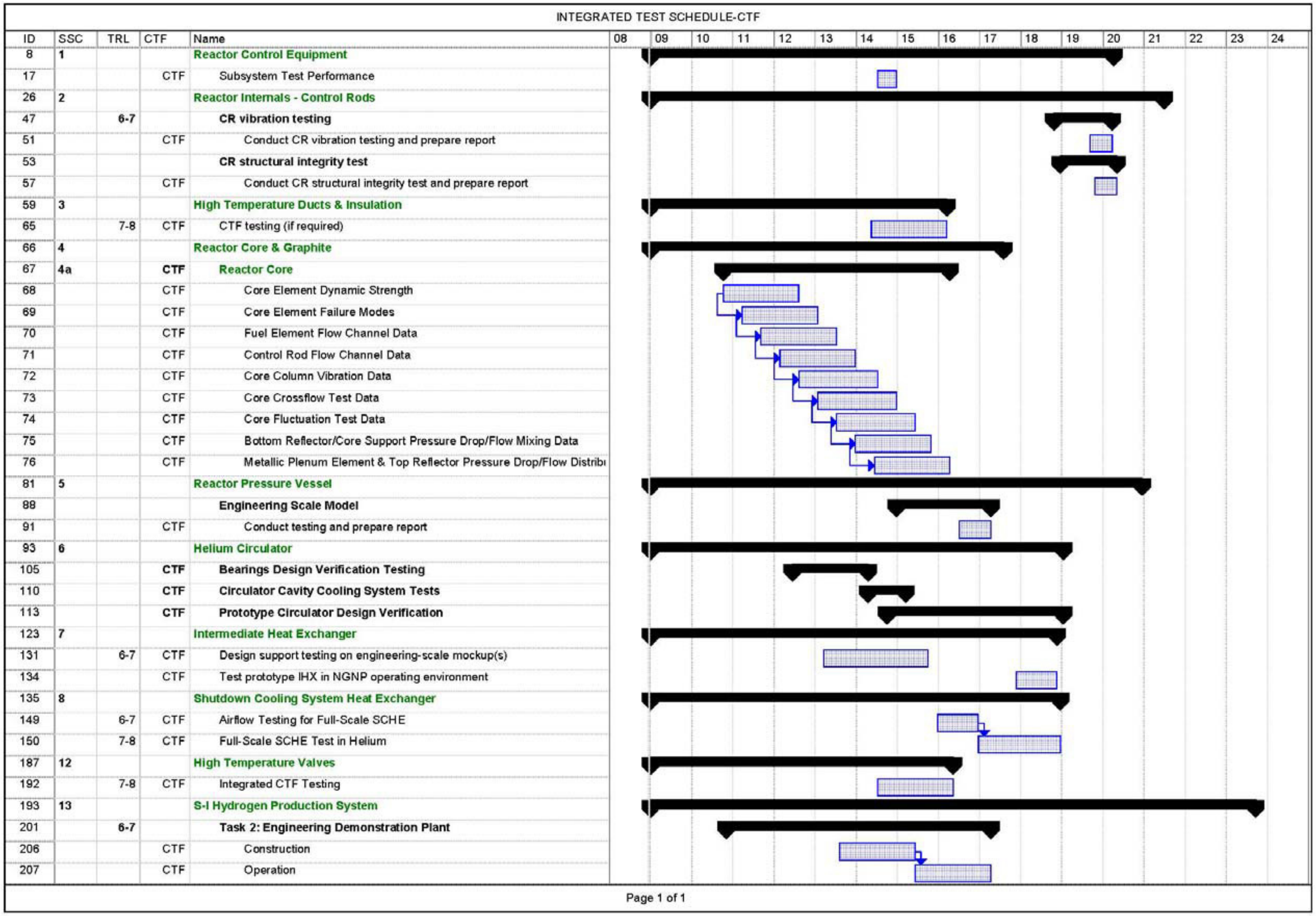

Figure 5-2. Schedule for Potential Testing in CTF

6 SURVEY OF INTERATIONAL INTEREST IN CTF

GA conducted the following initiatives to assess the needs and requirements of the international gas-cooled reactor community for component testing in a high-temperature helium loop:

- Distributed a questionnaire at the HTR2008 Conference soliciting input from HTR2008 Conference attendees on potential international interest in conducting tests in the CTF, and the probable requirements for such tests
- Supplemented the above effort by sending out an electronic mailing of the same questionnaire distributed at the HTR2008 Conference to international parties that GA considers likely to have an interest in the CTF.
- Solicited input from GA's Utility Advisory Board
- Solicited input from GA's NGNP international team members (e.g., Fuji Electric, Rolls-Royce, and KAERI)
- Solicited input from JAEA

The effort to obtain input from HTR2008 Conference attendees was totally unsuccessful. Although, GA's questionnaire was included in the packet provided by the conference organizers to every conference attendee, only one questionnaire was returned to GA and the respondent did not offer any input with regard to potential testing in the CTF. The supplemental mailing was made to the parties shown in Table 6-1 on November 17.

Table 6-1. Mailing List for CTF Questionnaire

First Name	Organization Name	Country/Region	Email Address
Derek Buckthorpe	AMEC	UK	Derek.Buckthorpe@amec.com
Taiju Shibata	JAEA	Japan	shibata.taiju@jaea.go.jp
Micheal Futterer	European Commission Joint Research Centre	Netherlands	michael.fuetterer@jrc.nl
Changheui Jang	Korea Advanced Institute of Science and Technology	Korea	chjang@kaist.ac.kr
Suyuan Yu	Tsinghua University	China	suyuan@inet.tsinghua.edu.cn
Jie Wang	Tsinghua University	China	wangj@d103.inet.tsinghua.edu.cn
M.S. Yao	Tsinghua University	China	yaoms@mail.tsinghua.edu.cn
Yuliang Sun	Tsinghua University	China	sunyul@inet.tsinghua.edu.cn
Xavier Raespart	CEA	France	xraepsaet@cea.fr
Ursula Ohlig	Research Centre Juelich	Germany	u.ohlig@fz-juelich.de
Mabrouk Methnani	IEAE	Austria	M.Methnani@iaea.org
Ferhat Aziz	National Nuclear Energy Agency	Indonesia	ferhat@batan.go.id
Han de Haas	NRG	Netherlands	dehaas@nrg-nl.com
Eben Mulder	PBMR	South Africa	Eben.Mulder@pbmr.co.za
Osman Kadiroglu	Hacettepe University, Nuclear Engineering	Turkey	okk@nuke.hacettepe.edu.tr
Jun Lim	NUTRECK	Korea	jlim@peacer.org
Takeshi Matsuo	Mitsubishi Heavy Industries	Japan	Takeshi06226_matsuo@mhi.co.jp
Hee Cheon No	Korea Advanced Institute of Science and Technology	Korea	hcno@kaist.ac.kr
Martin Gronek	HOCHSCHULE ZITTAU/GOERLITZ	Germany	MGronek@hs-zigr.de
Torsen Rottenbach	HOCHSCHULE ZITTAU/GOERLITZ	Germany	T.Rottenbach@hs-zigr.de
Frank Worlitz	HOCHSCHULE ZITTAU/GOERLITZ	Germany	F.Worlitz@hs-zigr.de
Wolfgang Hoffelner	Paul Scherrer Institute	Switzerland	wolfgang.hoffelner@psi.ch
Céline Cabet	CEA	France	celine.cabet@cea.fr

At this writing, a response has been received only from Michael Futterer. Mr. Futterer indicated that he has distributed GA's request for input to the members of the European HTR-TN network, but has not received any feed-back as yet. Mr. Futterer also noted that the Europeans have started planning a new project on helium technology for 2010 and that he will get back to GA if he receives any input from HTR-TN members.

The only member of GA's Utility advisory board to respond was Dan Keuter of Entergy. Mr. Keuter noted that Entergy is interested in supporting the NGNP and HTGR in general, but that Entergy is not planning to directly use the CTF. He further noted that he sees the CTF being used by the reactor and component vendors, but not by the end users and operators such as Entergy.

GA's effort to obtain input from its international NGNP team members and from JAEA was much more successful. The following discussions present the input provided by JAEA, Fuji Electric, KAERI, and Rolls-Royce.

6.1 JAEA Input

The potential needs and requirements of JAEA for the VHTR, and likely the similar needs and requirements for the NGNP, for a high-temperature helium test loop are in the areas of:

- Helium circulator
- High temperature valve
- Reactor vessel and internals
- Control and instrumentation

The needs and requirements of these areas are described in the sections below.

6.1.1 Helium circulator

The helium circulation duty requirement for the NGNP is about 60-times greater than that of the state-of-the-art helium circulators operational in the HTTR. This indicates a significant need for technology development. The technology requirements for the NGNP helium circulator will likely include:

1. Specification of detailed design conditions to meet the NGNP functional and operational requirements
2. Helium circulator technology review and design selection

3. Development of a detailed design including fluid dynamic, rotordynamic and thermal/structural designs, and the necessary component technology development for gas seals and bearings and control system

4. Prototypical scale testing

A full-dimensional scale model of the helium circulator should be built and used for design and functional tests using the CTF. Preliminary design needs and appropriate test conditions for the helium circulator design validation are given in Table 6-2.

Table 6-2. Preliminary Needs and Conditions for He Circulator Design Validation

	NGNP Reactor	Test
Scale	600 MWt reactor	full scale mechanical
Design type	Axial or radial	Axial or radial
Helium flow (kg/sec)	250	35
Gas inlet temperature (^{o}C)	500	500
Gas inlet pressure	7	1
Pressure rise (%)	2	2
Circulator power (MWe)	12	2
Rotational speed (rpm)	~ 5000	~ 5000
Rotor diameter (m)	~ 1	~ 1
Motor drive	He submerged Induction motor	He submerged Induction motor
Shaft seals	He dry gas seal	He dry gas seal
Journal/radial bearings	Magnetic/catcher bearings or oil bearings optional	Magnetic/catcher bearings or oil bearings optional

If a circulator with water or oil bearings is used, the probability of water or oil ingress accidents is high. A circulator with gas or magnetic bearings should be used.

6.1.2 High Temperature Valves

For potential process applications including hydrogen production, the heat in the helium coolant of the NGNP at temperatures of about 950°C is transferred to a secondary high temperature helium loop through an intermediate heat exchanger (IHX). This interface between the primary and secondary systems requires fast-acting isolation valves operating in the high temperature environment. The valves are required to prevent the release of radioactive materials to the hydrogen production system and the environment in case of a rupture of the IHX tubes, and to protect the primary system from chemicals and combustible materials in case of a failure of the process heat exchanger. This need for the isolation valves is illustrated in Figure 6-1.

In JAEA, a large-scale model of a high-temperature isolation valve, as shown in Figure 6-2, has been fabricated, and the gas tightness of the valve in multiple open and shut cycles has been measured in temperatures around 900°C in a test loop as shown in Figure 6-3. The future technical issues include minimization of the valve seat deformation caused by thermal expansion in high temperature, development of modified or new coating materials to maintain the seat-face roughness following multiple rounds of close and shut actions in cold and hot conditions. The lifetime of the valve seat and coating materials to secure leak tightness performance must be extended and confirmed.

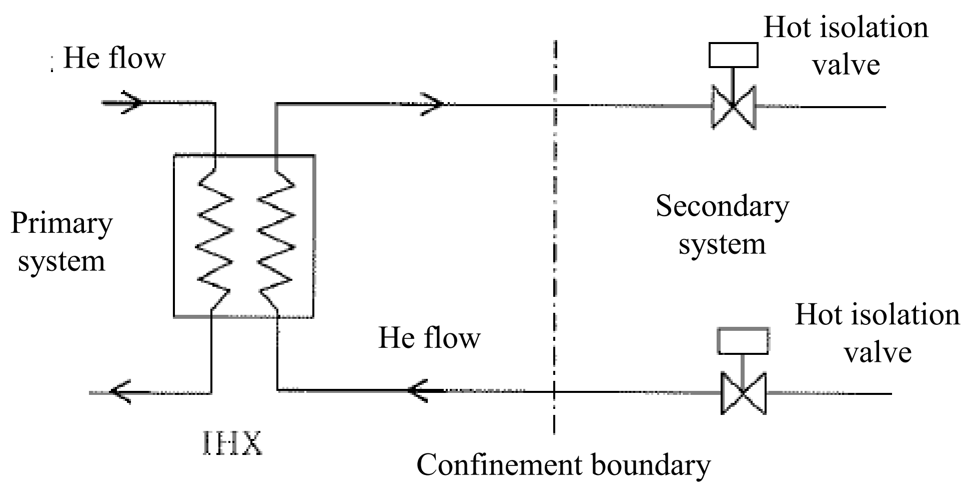

Figure 6-1. High temperature isolation valve requirements

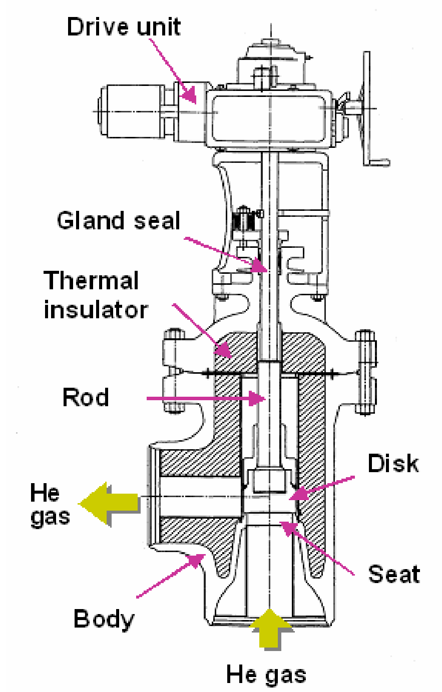

Figure 6-2. The isolation valve model built and tested for the HTTR

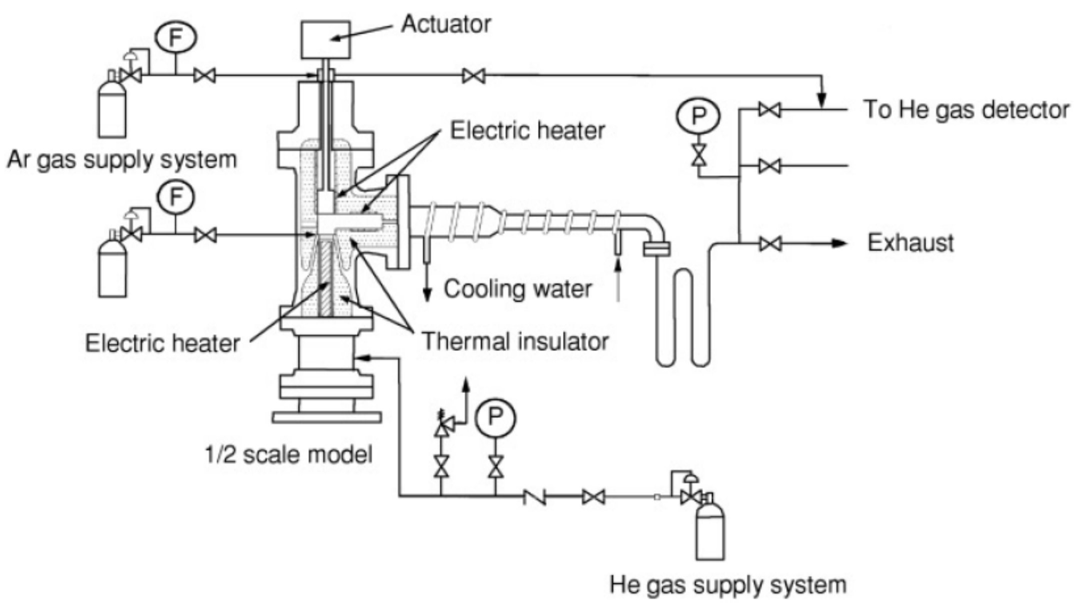

Figure 6-3. The test loop for the HTTR 905°C isolation valve

Development of a detailed thermal and structural analytical model is necessary to evaluate mechanical strength performance under long-term high-temperature operation. This should be followed by the verification testing. In addition, significant design scale up and optimization must be done to meet the technical performance and additional cost requirements for the NGNP.

Table 6-3 compares the test conditions of the high temperature isolation valve for the HTTR and the design need and test conditions likely to be required by the NGNP.

Table 6-3. High Temperature Isolation Valves

	HTTR Reactor	NGNP
Design type	Angle valve	Angle valve
Fluid	Helium	Helium
Mass flow rate (kg/sec)	2.5	90
Volume flow rate (m3/sec)	1.5	44
Inlet temperature (^{o}C)	905	900
Inlet pressure	4.1	5
Seat		
bore ID	0.204	0.5
Bore O.D.	0.244	0.7
Material	Hastelloy X	Hastelloy X
Body		
O.D.	0.59	2
Height	1.5	6
Material	Cast steel SCPH32	Cast steel SCPH32

6.1.3 Reactor Vessel and Internals

The experience gained in the developmental phases and ensuing reactor operations in the HTTR project points to the need for a large engineering-scale, high-temperature mockup of the full NGNP reactor to enable the performance of the detailed design to be confirmed for the critical thermal and fluid dynamic issues described below.

1) Distribution of helium flow at the nominal temperature of 900°C at high pressure in the mockup reactor internals. Measurement of by-pass or leakage flow is important. This investigation would satisfy the critical need to validate the design methods for concentrating and mapping the effective coolant flows to the fueled core regions, eliminating the hot streaks, cooling the control rods and drive mechanisms, and maintaining the uniform operating temperatures of the reactor pressure vessel.

2) Process of air ingress and the effectiveness of possible mitigation methods in the reactor internals at temperatures up to 900°C at near atmospheric pressure. The question of air ingress may be an important issue in the detailed design development and licensing for the NGNP. A rupture in the primary system piping is an event that should not result in significant safety consequences in the NGNP. In such a loss-of-coolant event, the reactor would be shut down inherently and the decay heat removed passively with the ultimate reactor temperature rise being less than the design limit. Still, graphite oxidation damage to the fuel and core in the event of a major air-ingress through the breached primary pressure boundary remains an important concern to reactor safety.

Two major air ingress events should be studied using the mockup reactor model including the graphite core. One is rupture of a control rod or refueling standpipe atop the reactor pressure vessel and the other is rupture of one or more main coolant pipes on the lower body of the reactor pressure vessel. Experiments and benchmarked analyses should be performed to understand the complex air ingress sequences and mechanisms in the depressurized reactor. Possible air ingress mitigation methods should be devised and validated by testing.

3) Process of water ingress from steam generator and shutdown cooling heat exchanger at elevated temperature and pressure.

6.1.4 Control and instrumentation

The availability and reliability of temperature and neutron measuring instrumentation in high temperature are of concern based on JAEA’s experience in the HTTR and other past test reactor experience.

6.1.4.1 Fuel temperature measurement

The following items are necessary for in-core fuel temperature measurements during normal operation of the NGNP. The integrity and applicability of the measurement system of the items in the following high-temperature ranges should be confirmed in a high-temperature test loop, including:

- N typed thermocouples: 0 ~ 1200 ºC
- B typed thermocouples: 0 ~ 1500 ºC
- Various melting wires: 900 ~ 1400 ºC

6.1.4.2 Neutron instrumentation system

The following are for the neutron instrumentation system (NIS) used for post accident monitoring (PAM). The integrity of the NIS under the conditions expected in a depressurization accident should be confirmed for monitoring the sub-criticality after reactor shutdown, assuming that the NIS would be located within the NGNP reactor pressure vessel. The items and the test condition are:

Fission chamber or alternatives: 600~800°C at the measurement location

6.2 Fuji Input

6.2.1 Background

The VHTR has the potential to provide high-temperature coolant at about 950°C, but it is clear that the coolant flow fraction through the fuel elements in the core region is required to be over 85% to keep fuel temperatures within acceptable limits during normal reactor operating conditions. Carefully designed seal mechanisms are needed to attain such a highly-effective coolant flow fraction in the core region, and are also needed to keep steel structures insulated from hot reactor outlet coolant.

Another challenge for a VHTR is an ingenious solution that allows for use of SA-508/533 steel for the Reactor Pressure Vessel (RPV). Thermal analysis results suggest that it will be necessary to employ direct vessel cooling to ensure with high confidence that peak vessel operating temperatures are below the ASME code limit of 371°C for SA-508/533 steel. In the Japanese GTHTR300 design, a direct Vessel Cooling System (VCS) routes compressor-discharged helium at 140°C through flow paths in the Permanent Side Reflector (PSR). This VCS is necessary and sufficient to keep RPV temperature below the ASME code limit of 371°C during normal operating conditions, but the VCS has little effect on vessel cooling during a low-pressure conduction cooldown event. However, the coolant flow paths in the PSR have an advantage over other coolant flow path options, such as a double core barrel and channel box arrangement, from the standpoint of keeping fuel temperatures lower during a LPCC event.

Therefore, the combination of a VCS and the coolant flow paths in the PSR has the potential to be a significant design improvement relative to the GT-MHR during both normal operation and an LPCC event. However, there could be substantial difficulties associated with designing the core bottom structure to incorporate these design features because the inlet structure for the VCS coolant, the inlet structure for the PSR coolant flow paths, and the thermal insulation for the hot reactor outlet coolant must be located in close proximity. Thus, it will likely be a challenge to demonstrate that the VCS and coolant flow paths in the PSR can be made compatible with the core bottom structure and seal mechanisms.

6.2.2 Requirement for a scale model test in the envisioned large-scale CTF

The following 3 items should be demonstrated by a scale-model mock-up of the VHTR core bottom structure as shown in Figure 6-4. The appropriate scale for these experiments is 1/3 to 1/1 because the real gap width is about 2-3 mm, which would be difficult to reproduce accurately at a scale smaller than 1/3.

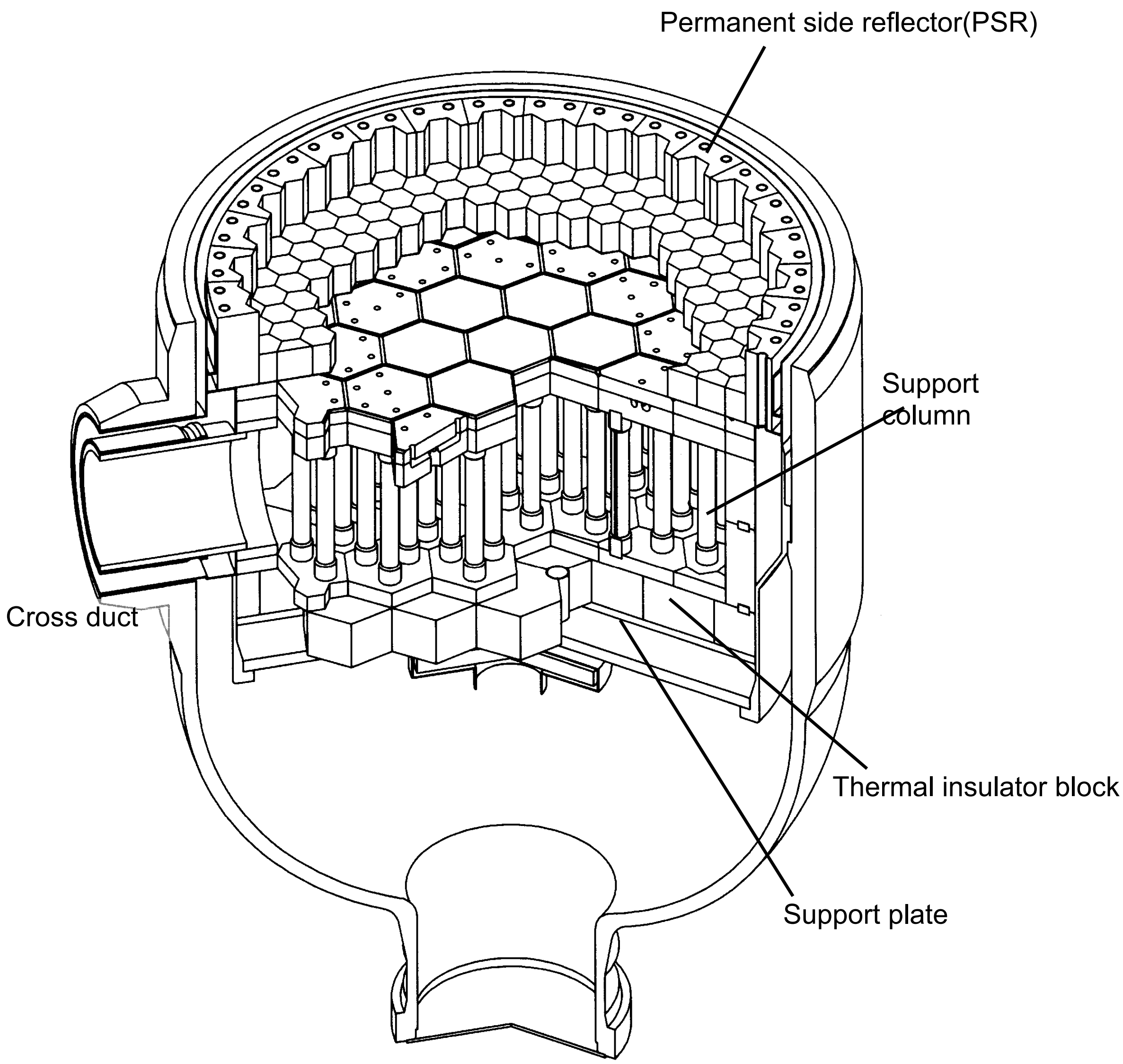

Figure 6-4. Typical core bottom structure of VHTR

Seal test for the bypass coolant flow path

The helium coolant returning to the reactor vessel is provided from the outer annular path in the cross-duct and is channeled to the inlet of the coolant paths in the PSR. The coolant must flow through the core barrel up to the PSR. This flow path is formed by the steel and graphite structures. There is a potential for helium to bypass the core if it leaks to the core hot plenum through gaps and clearances between the steel and graphite blocks. Testing is needed to verify the effectiveness of the seals that are designed to prevent gap flow between the graphite blocks. Figure 6-5 illustrates the experimental apparatus for a seal test for gap flow between bottom blocks. Figure 6-6 illustrates the experimental apparatus for a seal test for the coolant path between the PSR and the hot plenum, in addition to gap flow between bottom blocks.

Heat-up test for support plate

Thermal insulator blocks are installed below the hot plenum to prevent heat-up of the support plate and core barrel. It is concern that hot coolant in the hot plenum may flow through the gaps between the insulator blocks and contact with the upper face of the support plate. The potential for hot spots on the support plate should be checked by providing heated helium from the top of a mock-up assembly. Figure 6-7 illustrates the experimental apparatus for a heat-up test (a flow meter in each column is unnecessary in this case).

Effect to flow distribution of each fuel column due to the pressure distribution in hot plenum

Pressure variations within the hot plenum could cause significant non-uniform flow in the fuel columns. Figure 6-7 illustrates the experimental apparatus for a flow distribution test (a heater is unnecessary in this case).

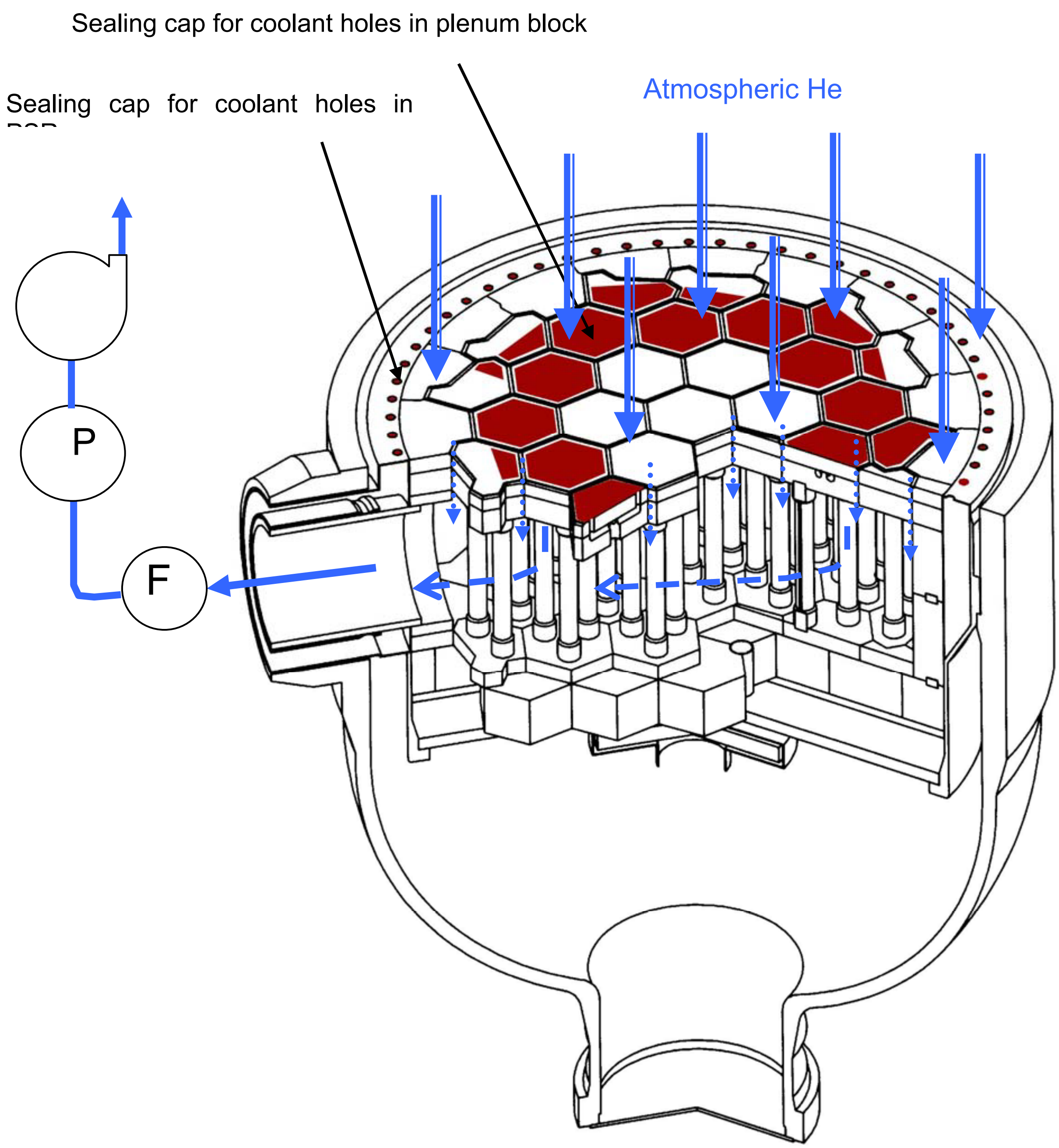

Figure 6-5. Experimental apparatus for fuel block bypass flow seal test

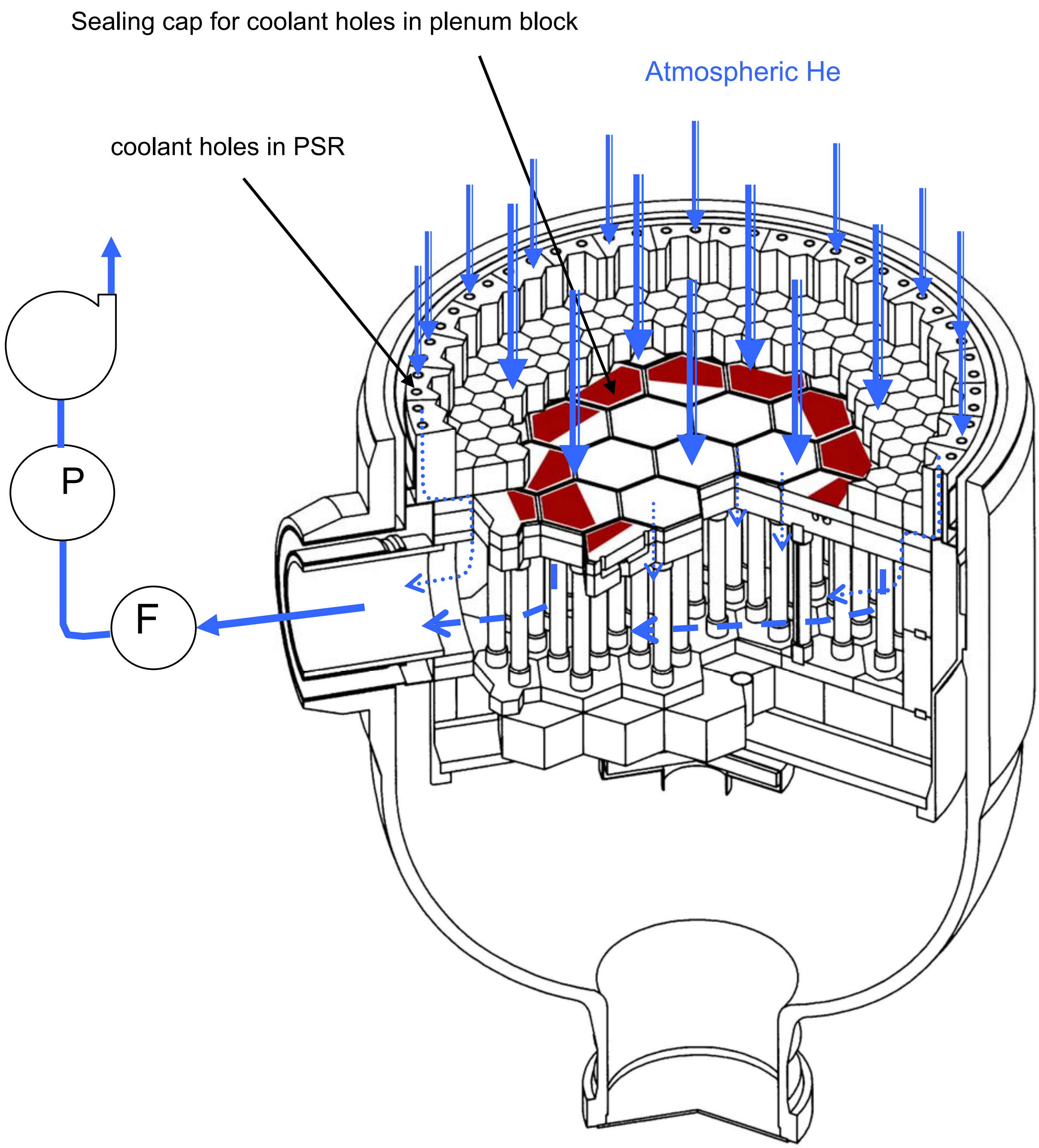

Figure 6-6. Experimental apparatus of PSR bypass flow seal test

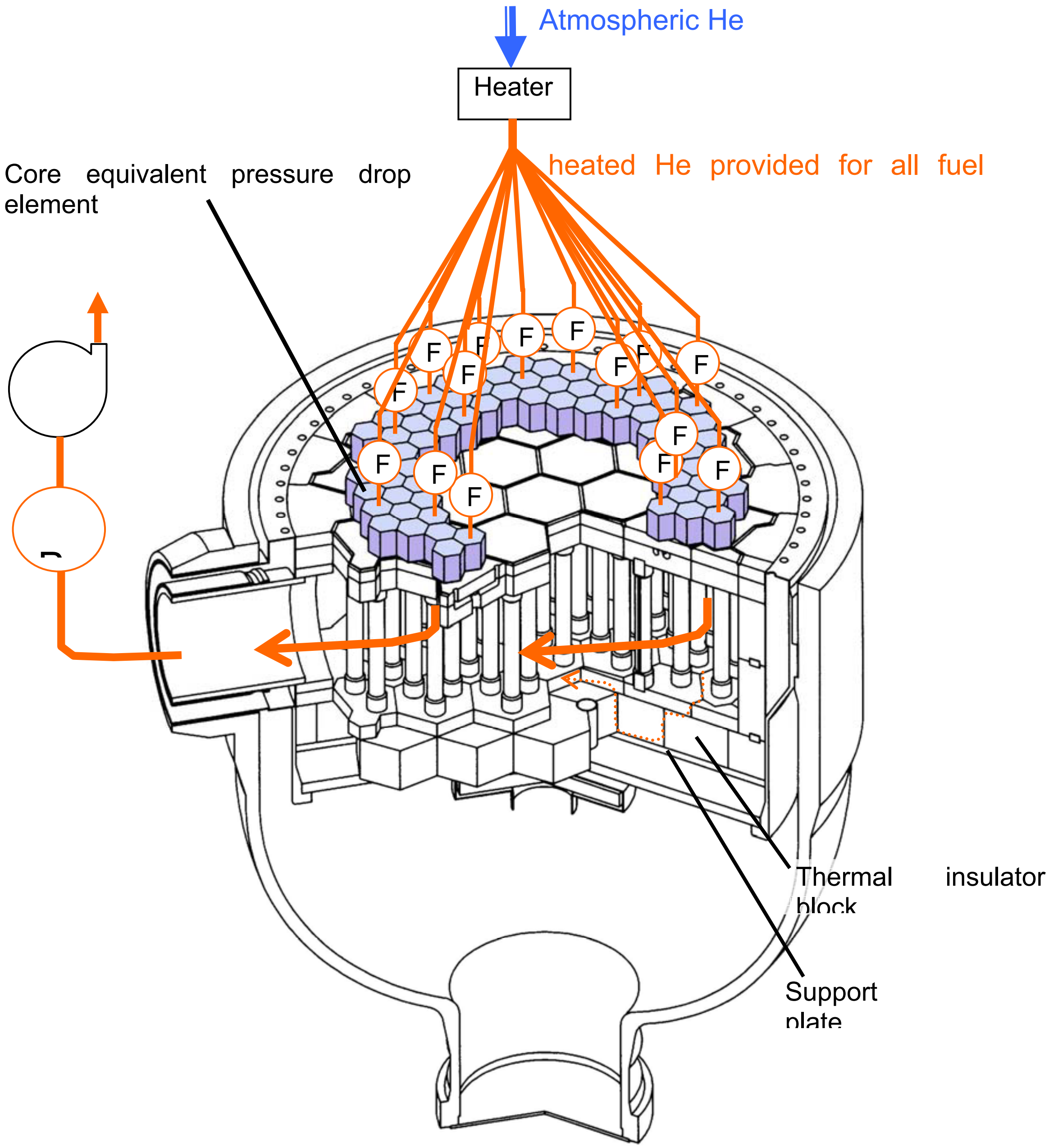

Figure 6-7. Experimental apparatus for heat-up test

6.3 KAERI Input

KAERI indicated potential interest in using the CTF for testing of heat exchangers and a reactor vessel cooling system.

6.3.1 Heat Exchanger (IHX and Process Heat Exchanger)

Types of Tests

- Steady and transients tests
- Thermal-hydraulic performance tests
- Structural integrity tests
- Tritium permeation tests

Test Conditions

- Thermal power: 1 ~ 2 MWt up to single module (40~50MWt)
- Primary side coolant: He
- Secondary side coolant
  - Sulfuric acid and/or sulfuric acid gas, steam for PHE tests
  - He, He-N2 mixture or molten salt for IHX tests
- Pressure difference between loop: variable from 0 to 70 bar
- Primary/secondary temperature: 0~980°C/0~950°C

6.3.2 Cooled Vessel Concept with Vessel Cooling System

Types of Tests

- Steady and transients tests
- Thermal-hydraulic performance tests
- RCCS performance tests

Test Conditions

- Thermal power: 10MWt
- Vessel material: SA508 and/or SA533
- Vessel cooling system
- Water- and Air-cooled RCCS

6.4 Rolls-Royce Input

Rolls-Royce identified several possible areas where the CTF could be used to generate test data for gas technologies. Rolls-Royce's input is provided verbatim in the following questionnaires that were graciously completed by Rolls-Royce staff who have been participating with GA on the NGNP Project. As noted in Rolls-Royce's responses below, their suggestions concerning possible uses of the CTF to support gas technology development do not imply any commitment by Rolls-Royce to perform tests in the CTF, and any interest by Rolls-Royce in performing such tests would be contingent on Rolls-Royce's participation in the NGNP project or on a corporate decision by Rolls-Royce to become involved in some other aspect of gas-reactor development.

Questionnaire - Component Test Facility (CTF)

Testing that Rolls-Royce believes could potentially be performed in the CTF
Testing of Turbomachinery aerodynamics for different working fluids depending on future requirements of GenIV nuclear plants. Cascade testing of a range of aerofoils would be of interest to de-risk and validate results against design tools. Examples of testing would be to measure tip losses and turning losses for various blade incident angles. It would also be of interest to carry out testing for measuring pressure losses and flow distributions of components such as diffusers, pipes and manifolds that are associated with the gas turbine. Advanced Compact Heat Exchangers is another area of interest to Rolls-Royce. A test facility capable of testing various advanced heat exchanger designs using novel working fluids such as Helium and Helium/Nitrogen mixtures. Testing would for example include heat transfer, pressure losses and flow distribution, integrity of heat exchanger to cyclic loading, assessing nitriding rates/risk in the heat exchanger section at high-pressure / temperature and leakage testing.

Approximate requirements and conditions for potential tests
High temperature and high pressure helium and/or helium/Nitrogen including cooling facilities. Capability would be required to supply very clean flow with precise and straight flow distribution. Intrusive Instrumentation for pressure and temperature measurements would be required i.e. fine controlled traversing pressure and temperature probes Depending on future requirements for GenIV plants loops with various working fluids would be needed. Capability to vary the mass flow ratio of mixtures.

Comments
Cascade testing of a single row of aerofoils would be significantly easier than a full component testing and would help de-risk component for a Helium or Helium/Nitrogen gas turbine in the NGNP or any future commercial plant where the PCS configuration is either a CCGT or pure GT cycle. Heat Exchanger testing would be of primary interest if the cycle used in the NGNP or any future commercial plant would be an indirect cycle to do testing related to the IHXs or a pure GT to do testing associated with intercoolers and recuperators. Rolls-Royce interest in testing in the CTF would be contingent on its participation in the NGNP project or on a corporate decision to become involved in some other aspect of gas-reactor development

Questionnaire - Component Test Facility (CTF)

Testing that Rolls-Royce believes could potentially be performed in the CTF
High-temperature Electrics (e.g. Active Magnetic Bearing coils) Electrical Lead-outs (Cabling, cable insulation and sealing/insulation of electrical penetrations for direct cycle with alternator and electromagnetic bearings inside the helium primary coolant) Seals (Labyrinth and Dry Gas) Gas Bearings Active Magnetic Bearings (and Back-up Bearings) Materials (Composites)

Approximate requirements and conditions for potential tests
High-pressure helium Alternative gases and mixtures for potential indirect cycles (nitrogen, helium/nitrogen and perhaps carbon dioxide) Realistic operating temperatures Seal and bearing tests require drive motors, to demonstrate operation at realistic shaft speeds These minor component technologies might be demonstrated in smaller installations, not in the main helium loop.

Comments
Penetrations, High-temperature Electrics, Electrical Lead-outs, Gas bearings and Seals are of general significance to combined cycle and gas turbine power conversion systems for future commercial high-temperature reactors. Active Magnetic Bearings (AMBs) and Back-up bearings are of more specific relevance to direct cycle systems. Rolls-Royce already has a program for development of low-power electrical systems to operate within environments of up to 400ºC. Rolls-Royce interest in testing in the CTF would be contingent on its participation in the NGNP project or on a corporate decision to become involved in some other aspect of gas-reactor development

Questionnaire - Component Test Facility (CTF)

Testing that Rolls-Royce believes could potentially be performed in the CTF
Gas Turbine Emergency Bypass Valve System Loss of electrical load requires a rapid response to prevent overspeed of the gas turbine and of the alternator. A large proportion of working fluid must be diverted from the compressor outlet to bypass the turbine. The valve throat area required is very large. Flow velocity through the valve will be very high, and the flow must be expanded smoothly to avoid damaging acoustic effects. Valve exhaust ducting is required to diffuse the flow to the point of merging with the turbine exhaust flow. The valve size and the large pressure difference across the closed valve imply an extremely high actuation force requirement. The system for achieving this will require testing. Valve inlet ducting requires testing to confirm avoidance of flow instability (such as an orbiting inlet vortex). The geometry of merging flow is likely to require testing.

Approximate requirements and conditions for potential tests
High-pressure helium, to provide realistic resistance to actuation. Alternative working fluids of higher density, such as nitrogen, helium/nitrogen mixture and perhaps carbon dioxide. Through-flow response to valve initial opening, although not necessarily at full scale. Realistic operating temperatures. Space for installation of this component, its associated ducting and its actuation system.

Comments
An emergency bypass valve is considered to be essential for combined cycle and gas turbine power conversion systems, which offer significant advantages to future commercial high-temperature reactor systems. The actuation force requirement will be reduced by a pressure-balancing system. This will be part of the system to be tested. Rolls-Royce interest in testing in the CTF would be contingent on its participation in the NGNP project or on a corporate decision to become involved in some other aspect of gas-reactor development.

Questionnaire - Component Test Facility (CTF)

Testing that Rolls-Royce believes could potentially be performed in the CTF
Gas Turbine Control Bypass Valve System During system start-up, the turbine and compressor are motored. The procedure for transferring to generation of shaft power may involve controlled closing of a large turbine bypass valve. This may be the same valve as the emergency bypass valve. If the Emergency Bypass Valve and the Control Bypass Valve are separate valves, then the Control Bypass Valve provides the means of managing recovery from operation of the Emergency Bypass Valve. Controllability and stability of operation are to be demonstrated.

Approximate requirements and conditions for potential tests
As for the Emergency Bypass Valve, which may be the same valve.

Comments
The actuation force requirement will be reduced by a pressure-balancing system. This will be part of the system to be tested. The start-up procedures have not yet been defined for combined cycle and gas turbine cycle power conversion systems. Procedures may be devised which would relax the requirement for controllability of the bypass valve. The Emergency Bypass Valve is required to open rapidly. The Control Bypass Valve is required to close slowly and under control. These contrasting requirements may require separate control systems, requiring two valves rather than one combined valve. Rolls-Royce interest in testing in the CTF would be contingent on its participation in the NGNP project or on a corporate decision to become involved in some other aspect of gas-reactor development

Questionnaire - Component Test Facility (CTF)

Testing that Rolls-Royce believes could potentially be performed in the CTF
Steam Generator testing of a typical full size module would be extremely valuable if NGNP uses this type of system. CTF could simulate the full range of stresses involved in start up and operation of the system, and allow the SG to be instrumented to a greater extent than is possible in a nuclear plant. Typical nuclear SGs have a very long life so that creep and fatigue effects become important. They can be large and therefore difficult to replace.

Approximate requirements and conditions for potential tests
It would be important to simulate flow conditions through the SG to ensure that vibrations etc are duplicated. These SGs are typically very heavy and the unit would need to be suspended in a realistic manner. Stresses can be generated during start up and shut down. Chemical impurities would need to be simulated. It may be difficult to accelerate this testing so that timescales will be long, and it may therefore be beneficial to start early.

Comments
Testing will be simpler if the design is modular, so that only one module need be tested. It may be possible to build on UK experience of SG design for gas reactors in order to help to de-risk these components. Rolls-Royce has currently undertaken no work on NGNP specific SG design issues. Rolls-Royce interest in testing in the CTF would be contingent on its participation in the NGNP project or on a corporate decision to become involved in some other aspect of gas-reactor development.

7 C0MMENTS AND RECOMMENDATIONS ON INL CTF F&ORS

The scope of this task included review and comment on the Component Test Facility Functional and Operational Requirements as defined in INL document INL/EXT-08-14150, Rev. 0 dated April 28, 2008. GA reviewed the F&ORs and also arranged for JAEA to review and comment on them as a subcontractor to GA NGNP team member Fuji Electric. GA's and JAEA's comments are presented below.

7.1 GA Comments

General Comments

1) The scope and mission statements could be made more substantive if they would address more completely the potential applications for NGNP. At present, only the National Hydrogen Initiative in mentioned. It would help to have a statement to the effect: "The CTF mission is to support development, qualification, risk reduction and licensing for the NGNP and its potential applications, which include hydrogen production, electricity production and coal conversion and process steam for mineral recovery, industrial and petrochemical applications."

Specific Comments

Section 3.1) The administrative & support functions should include a machine shop and welding shop to support experimental set-up and operation.

Section 3.1) The administrative & support functions should include a loading dock and receiving/inspection facility.

Section 3.1.3.2) The list of components for the secondary loop are the same as for the primary loop. Rather than having another IHX, the secondary loop would have a process heat exchangers (e.g. reformers). In addition, a secondary loop would couple the gas-turbine, compressor, recuperator, pre-cooler and intercooler.

Section 3.1.3.2) This section should require heat removal capability for testing of some power system components.

Section 3.1.3.7) It is not clear what a coolant or HT fluid test is. The requirement should be more specific or give examples such as fluid compatibility and fluid stability testing.

Section 3.1.3.8) This statement says that direct-cycle power conversion testing capability must be provided, whereas 2.6, Item 13 states that "unless otherwise stated, planned test are for an indirect cycle". This appears to be a conflict.

Section 3.1.3.19) As stated, this functional requirement could be interpreted as having to incorporate a shake table into the HT flow facility. It is suggested that it the requirement be restated as “vibration and seismic monitoring capability”.

Section 3.1.3.22) This requirement is too general to be useful. It is recommended that it be deleted.

Section 3.1.3.24) This requirement is too general to be useful. It is recommended that it be deleted or revised to give specific examples of the type of mechanical properties testing that should be provided.

Section 3.1.4.2) It would be more helpful if this requirement was more specific as to the type of analysis required such as chemical, metallurgical, microscopy, etc.

Section 3.1.5.5) Rather than “enable heat rejection”, the facility should provide a heat rejection facility.

7.2 JAEA Comments

JAEA reviewed the reference documents listed below within the context of the types of tests that JAEA envisions will be performed in the CTF (as discussed in Section 5.1).

1. High Level Requirements – “High Temperature Gas Reactor (HTGR)-Component Test Facility (CTF),” INL/MIS-08-14156 (PLN-2763), Rev 0, 4/28/08
2. Technical and Functional Requirements – “High Temperature Gas Reactor (HTGR)-Component Test Facility (CTF),” INL/EXT-08-14150, Rev 0, 4/28/08
3. The Component Test Facility – “A National User Facility for Testing of High Temperature Gas-Cooled Reactor (HTGR) Components and Systems, Paper HTR2008-58250
4. International CTF Users Requirements Study prepared by JAEA for GA (Section 5.1)

JAEA previously identified its anticipated CTF needs and requirements for the VHTR in Reference 4 and considers these to likely also be the needs and requirements of the NGNP. These CTF needs are in the following component and integrated system areas:

- Helium circulator
- High-temperature valve
- Reactor vessel and internals
- Controls and instrumentation

The following functional requirements are identified or revised to enable the testing of the above components and systems.

Functions 3.1.3.1 and 3.1.3.2 in INL/EXT-08-14150 are judged to satisfy the functional and operational requirements for testing of the helium circulator and the high-temperature valve.

Function 3.1.3.4 should be revised to include the reactor pressure vessel (RPV) and an extended period of testing capability as required for the testing of the reactor vessel and internals. Because this functional test is on a scale-model of the NGNP RPV and associated components and systems, it is important that an extended period of testing capability be allowed in the CTF to meet the design data needs (e.g., accident analysis method validation) of the NGNP. This includes testing under the thermal and hydraulic transient conditions in pressurized and depressurized events.

Function 3.1.3.4 should be modified to read as follows (with changes indicated by italicized bold text).

> **3.1.3.4 Function**: Enable testing of scaled models of the NGNP reactor vessel and associated components/systems
>
> The CTF will have the necessary equipment to enable concurrent reactor component or integrated testing (e.g., ***reactor pressure vessel [RPV],*** control rod drive mechanism [CRDM], graphite blocks, graphite reflectors, reactor blocks, core structure, plenum, graphite core, and reactor cavity cooling system [RCCS]) with other testing listed in this section (3.1.3). It will need integrated reactor component ***or system*** testing capability for up to ***one or more weeks test duration***. In addition, the CTF needs the capability for shutdown cooling and control tests.

Function 3.1.3.5 should include the associated instrumentation testing and calibration capability for the control and protection hardware and software, and the instrumentation for the full-scale helium circulator for the NGNP. This is considered necessary because the helium circulator for the NGNP is expected to be equipped with magnetic bearings and catcher bearings, which will require extensive testing.

Function 3.1.3.9 satisfies the functional and operational requirements for the controls and instrumentation.

8 REFERENCES

[GA 2008a] Labar, M., R. Phelps, and J. Saurwein, "NGNP Steam Generator Alternatives Study," General Atomics Report 911120, April 2008

[GA 2008b] Labar, M., and J. Saurwein, "NGNP Power Conversion Alternatives and Selection Study," General Atomics Report 911131, October 2008

[GA TDP 2007] Hanson, D., "NGNP Umbrella Technology Development Plan," GA Report PC-000543, Rev. 0, July 2007

[INL 2007] "NGNP Engineering White Paper: High Temperature Gas Reactor – Component Test Facility," INL Report INL/EXT-07-13146, Rev. 1, November 2007

[INL 2008a] Balls, V., D. Duncan, and S, Austad, "The Component Test Facility – A National User Facility for Testing of High Temperature Gas-Cooled Reactor (HTGR) Components and Systems," Paper HTR2008-58250, Proceedings of the 4th International Topical Meeting on High Temperature Reactor Technology HTR2008, September 28-October 1, 2008, Washington, DC USA

[INL 2008b] "High Temperature Gas-Cooled Reactor (HTGR) Component Test Facility (CTF) – Technical and Functional Requirements," INL Report INL/EXT-08-14150, Rev. 0, April 2008

[INL 2008c] "Technical Readiness Level Plan," INL Report INL/EXT-08-14251, draft, May 2008

[PCDSR 2007] "NGNP and Hydrogen Production Preconceptual Design Studies Report," GA Report 911107, Rev. 0, July 2007

APPENDIX A
TEST PLAN FOR HOT DUCT (SSC-3)

Revision FINAL

Engineering Services for the Next Generation Nuclear Plant (NGNP) with Hydrogen Production

Test Plan for Helium Duct and Insulation

Prepared by URS – Washington Division

For General Atomics

URS Washington Division

Study Report:

Test Plan for

Signature of Preparer ....

Reviewer: .... William McTigue

Title: .... Consulting Engineer

Signature of Reviewer ....

Approved .... HanKwon Choi

Title .... Project Director

Signature of Approver ....

Title: .... Consulting Engineer

Signature of Reviewer ....

Approved .... HanKwon Choi

Title .... Project Director

Signature of Approver ....

Project Number: 29105-3000

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List of Acronyms

CFD – Computational Fluid Dynamics

FEA – Finite Element Analysis

CTF – Component Test Facility

HT – High Temperature

TRL – Technology Readiness Level

ALARA – As Low as Reasonably Achievable

TID – Total Integrated Dose

MHTGR – Modular High Temperature Gas Reactor

RAMI – Reliability, Availability, Maintainability, Inspectability

DBA – Design Basis Accident

NPH – Natural Phenomena Hazards

PCS – Power Conversion System

RCPB – Reactor Coolant Pressure Boundary

1.0 INTRODUCTION

The Next Generation Nuclear Plant (NGNP) Hot Duct and Insulation Test Plan specifies the scope, approach, methodology, goals, resources, and schedule of each step of the Technology Development Plan to drive the Technological Readiness Level (TRL) of the hot duct and associated insulation from a TRL of two (TRL-2) to an eight (TRL-8). TRL-8 may require dynamic testing of a full size prototype at the NGNP Component Test Facility (CTF) planned for construction at the Idaho National Laboratories (INL). The necessity of CTF testing will be based on the findings of this TRL development plan.

Unless indicated otherwise, "hot duct" herein refers to the primary system hot leg ducts connecting the reactor vessel to the IHX and steam generator (assumed to be the preferred power conversion system (PCS) component) and enclosed in the cross vessels that establish the reactor coolant pressure boundary (RCPB); the annular space between the hot duct and cross vessel is the reactor coolant cold leg. From a technology development perspective, the hot duct is considered to be the limiting high temperature duct and insulation application for the GA NGNP design for the following reasons:

- highest operating temperature and potential for hot streaming
- exposure to graphite dust and other particulates
- design and inspection challenges for vessel nozzle connections
- hot to cold leg leak detection
- total integrated dose and ALARA

The insulation and graphite together are referred to as the thermal barrier. This test plan concentrates on the insulation since the graphite application is well established. Some TRL advancement tasks to a large extent mirror other technology development tasks for other portions of NGNP design. Existing experimental data may be sufficient to advance some aspects of the design without the physical testing outlined in this test plan. Material property verification (coupon tests) is a good example of this. Several reactor components consider making use of the same materials so existing coupons experimental test data may substitute for the coupon tests for the hot duct development.

1.1 TRL Level Breakdown

The TRL levels can be better understood and more efficiently applied to the hot duct if they are broken down by components, sub-systems and systems. Each level has its own highest applicable TRL as indicated in Figure 1.

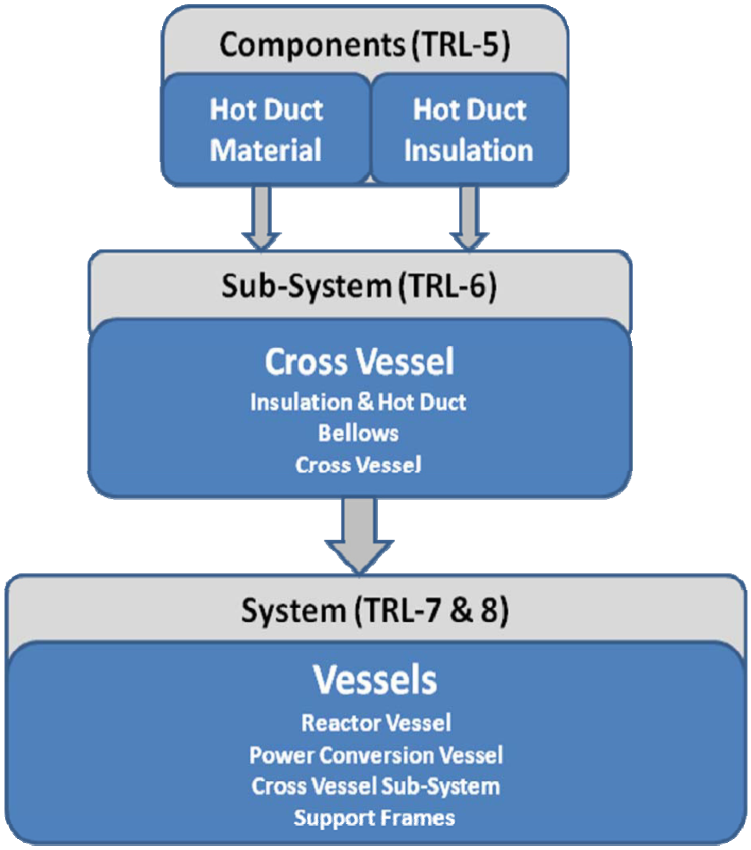


Figure 1: TRL Level by System Breakdown

A TRL-2 was initially assigned to the hot duct and insulation because a proposed configuration for the insulated duct has been formulated and the technical challenges associated with containment of high temperature and pressure helium gas are understood. Additionally, published data indicates that there are commercially available insulating materials and duct alloys that are viable candidates. However, critical functions and/or characteristics for a duct/insulation system have not been proven for the NGNP service conditions.

Table 1 below summarizes the steps required for each TRL advancement, where in this plan they are detailed and the estimated time for completion.

Table 1: TRL Task Summary and Estimated Duration

TRL	Task Description	Section	Estimated Duration
3	Safety Class Determination	2.1	6 Months
	Relevant Standards and Codes Applicability	2.2	
	Establish Conditions of Service	2.3	
	Insulation and Duct Material Selection	2.4	
4	Upfront CFD Flow and Temperature Analysis	3.1	1 Year
	Upfront FEA Stress Analysis	3.2	
	Hot to Cold Leg Leak Detection Method	3.3	
	Insulation Connection Method	3.4	
5	Hot Duct and Insulation Material Properties Tests	4.1	1 Year
	Component Level Test	4.2	
	Life Cycle Cost Analysis	4.3	
	RAMI Analysis	4.4	
	Acoustic and Flow Induced Vibration Analysis	4.5	
	Endurance Limit Analysis	4.6	
	Creep Analysis	4.7	
	ALARA Analysis	4.8	
	LIMIT Analysis	4.9	
6	FEA Simulation Optimization	5.1	1 Year
	CFD Simulation Optimization	5.2	
	Sub-System Level Test	5.3	
	Thermal Expansion Analysis	5.4	
7	Testing of Integrated System	6.1	1 Year
	Risk Assessment for CTF Testing	6.2	
8	CTF Testing (if required)	7.0	2 Years

1.2 Objectives

This test plan is inclusive of all steps necessary to advance the hot duct design from TRL-2 to TRL-8. As a result, steps are added that include all necessary information gathering, research, material selection and simulations that must be accomplished before physical testing can begin.

The test objectives are many, including all tasks necessary to advance to TRL-8. The main objectives are listed here.

- Compile applicable values and requirements from the codes listed in Section 2.1
- Establish Conditions of Service that the hot duct assembly must be designed to endure including all design basis accidents
- Select the appropriate materials of construction
- Establish the most prudent design for the hot duct including connection to the adjoining vessels and connections between the various components of the hot duct assembly
- Select a method of leak detection between the hot and cold ducts

- Perform FEA and CFD simulations to advance the design before testing begins
- Perform accelerated erosion and corrosion, acoustic, fatigue, endurance, creep, ALARA and LIMIT analyses to confirm duct integrity
- Verify selected material properties through physical testing
- Qualify the design for intended service

This test plan represents the overall approach to demonstrating the capability of the hot duct to meet specified performance requirements over its design lifetime. The test plan can be revised as execution of the technology development plan progresses. QA requirements, which apply to all test plan elements, are listed separately.

1.3 Approach

The Test Plan task begins with identifying the design bases and determining the Conditions of Service under normal, upset and faulted conditions. The properties of candidate materials for the hot duct and insulation are compiled over the range of operating conditions.

A high fidelity 3D model is then constructed that represents the design, configuration and geometry of the hot duct. A number of analyses will be performed which simulate the various aspects of operation, allowing the behavior of the design in response to loads to be visualized. Detailed analytical simulations will be performed in at least six areas, outlined below in Section 3. Results will be combined to evaluate the response of the system to all loads applied simultaneously under all operating cases. The final models will feature all design aspects of the hot duct.

The analytical models will be validated using test results obtained from various stages of testing. Several other analyses will be completed including creep and endurance limits, LIMIT, ALARA, RAMI, risk assessment, thermal expansion, acoustic response, response to DBA and NPH, and Hazard Analyses.

After the material for the duct components has been chosen and initial upfront FEA and CFD analyses completed, component level testing will be completed. Once the component materials have been proven the sub-system including the cross vessel will be tested. The final test is a scale test of the entire system. A Risk Assessment will help determine whether integrated CTF testing should be undertaken. Results of model and coupon tests will be compiled to predict behavior and analytical models will be validated.

2.0 TASKS TO ACHIEVE TRL-3

TRL-3 is in essence a proof of concept usually consisting of laboratory scale tests. Due to the fact that the components to a large extent have been proven in other similar applications, the co-axial duct concept has been proven. However, conditions of service must be documented, safety class determined, code and standard applicability resolved, and initial materials selected.

2.1 Safety Class

A component's safety classification influences its criteria for design, fabrication, testing and inspection, and may determine leakage detection requirements in the particular case of the hot duct. Safety classification is therefore considered to be relevant to technology development and test planning of the hot duct assembly.

The hot duct assembly for the 350 MWt Modular High Temperature Gas Reactor (MHTGR) design is classified as non-safety related in [PSID, 1992]. The MHTGR cross vessel that encloses the hot duct is classified as safety-related and designed to ASME III criteria as part of the vessel system and reactor coolant pressure boundary. In response to NRC comments pertaining to cross vessel failures, [PSID, 1992] summarizes the results of fracture mechanics evaluations that support a low probability of catastrophic failure of the cross vessel (response to NRC comment R 5-49), and states that leak detection capability and application of leak-before-break technology similar to light-water reactor (LWR) designs are not required for the MHTGR (response to NRC comment R 5-18).

Safety classification of MHTGR structures, systems and components (SSCs) is identified as a licensability issue in [PSER, 1996], in part because MHTGR SSCs were considered safety related only if they were required for accident dose consequence mitigation. Current NRC licensing policy for advanced reactors, e.g., as given in [NUREG-1860], suggests that safety classification criteria for the NGNP will be similar to that of current LWRs.

Design differences between current LWR designs and NGNP support a much lower safety significance for the hot duct than the analogous LWR reactor coolant system hot leg that is nuclear safety Class 1 as part of the reactor coolant pressure boundary. However, defense-in-depth and deterministic licensing basis considerations suggest the possibility that the hot duct may have to be classified as safety-related to support plant licensing. The following is quoted from Section 3.2 of the preliminary NRC evaluation of the MHTGR in [PSER, 1996]:

The gross failure of the cross vessel is a consideration for the design and licensing of the NGNP. In the preliminary licensing review of the 350 MWt MHTGR, GA determined the probability of gross failure of the cross vessel to be less than 10E-08 per plant-year. NRC stated that it could not confirm the gross vessel failure probability estimates [PSER, 1989 §5.2.5].

More recent conceptual design studies for the NGNP include consideration of air ingress events, e.g., that could result from gross failure of the cross vessel, and show promising results with respect to the safety significance of graphite oxidation resulting from such events [Richards, 2008]. The design criteria, inspection requirements and primary system leakage detection capability of the hot duct are factors influencing the integrity of the primary coolant system, and pose potential challenges to the cross vessel design. Therefore, this test plan considers applicability of codes and standards and definition of design and inspection criteria to be the first steps in the hot duct assembly's technology development.

"It is the staff's position that to ensure that the margins of integrity of the MHTGR steel reactor vessel are at a level comparable to that for LWR steel reactor vessels, some combination of plant systems design and additional safety analyses must be pursued to lower the expected frequency of [ASME Code] Service Level C and D occurrences to values consistent with LWRs (i.e., Table I of SRP 3.9.3). This reference to plant system design involves the questions of safety classification and [Regulatory Treatment of Non-Safety Related Systems] RTNSS discussed in Section 4.2.5 above."

Although a failure of the NGNP hot duct may be shown to be acceptable from a consequence standpoint, the factors summarized above include considerations of defense-in-depth and maintaining a low probability of events. These considerations affect NGNP licensing and suggest the need to revisit the hot duct assembly safety classification and its implications for design, fabrication, inspection, testing and leak detection requirements.

2.2 Relevant Standards and Code Applicability

The initial test plan presented herein considers the approach presented in GA's preliminary design information for the 350 MWt Modular High Temperature Gas Reactor (MHTGR). The 350 MWt MHTGR hot duct was not considered to be part of the RCPB, and was therefore not classified as an ASME Code Section III, Class 1 component. The final design and licensing approach applied to the hot duct will strongly influence the test protocol, so this test plan considers applicability of code criteria and its effect on technology development and test criteria.

As an initial step in the test plan process, applicable codes and standards will be identified. It is not the intention of this test plan to identify all relevant codes, but rather establish code applicability. Generally, the following apply:

- ASME Boiler and Pressure Vessel Code
- Nuclear Codes and Standards
- Quality Assurance Requirements

The codes for metallic materials are well established. However, the code challenge may be in determining how to extend the codes to even higher temperatures. It is not yet known what hot duct or insulation materials will be used, but code qualified materials subjected to high temperatures (950°C) are few. Insulation protects the hot duct inside and the cross vessel; if this insulation can be shown to reduce the hot duct temperature below 760°C the established codes can be applied. The cold leg return gas is between 490 – 590°C and should not introduce and code qualification challenges.

2.3 Establish Conditions of Service

Section 3 of Vollman [Vollman, 2008] provides data for reactor vessel system conditions that will be used for hot duct technology development including the following.

- Long term and transient temperatures
- Neutron flux
- Impurities in primary helium
- Noise level

- Pressure transients

2.3.1 Normal, Fault and Upset Conditions

Temperatures including ambient and outer duct max temperatures, flow magnitude, operating pressures and working fluid properties must be compiled over the range of operating conditions including normal, upset, emergency and faulted conditions.

2.3.2 Design Basis Accidents and Natural Phenomena Hazards

Relevant Design Basis Accidents (DBA) including phenomena hazards must be known and quantified. This information will affect how many additional scenarios must be simulated to ensure the design withstands all possible DBAs before a demonstrationprototype is built.

2.3.3 Design Life

The design life of the hot duct is 60 years.

2.4 Insulation and Duct Material Selection

The insulation and duct material selection process will consider many factors. The materials will be selected based on the criteria listed below over the range of operating conditions including normal, upset, emergency and fault conditions.

- tensile strength
- fracture toughness
- creep and relaxation data
- high and low cycle fatigue criteria
- high temperature endurance limit
- fabrication limitations and tolerances
- thermal expansion
- welding compatibility with vessels
- welds and heat affected zone material properties
- material and fabrication cost
- chemical and radiation resistance
- dissimilar material interactions (insulation/duct)
- differential thermal expansion
- erosion and corrosion characteristics

2.4.1 Existing Material Data

Vollman [Vollman, 2008] has summarized some material candidates based on initial material examination done for the NGNP at Oak Ridge National Lab. The study has determined that the strongest material candidates for the duct are Alloy 800 (AT/HT), Haynes 230, and Hastelloy X. The hot duct requires a refractory lining to protect it from high temperatures regardless of the duct material chosen. Vollman also summarized relevant refractory candidate materials. Once again, these materials will be used as a starting point but other materials will also be considered for the refractory lining. The initial candidates are Harbison-Walker Greenlight-45-L and Greenlight 45-LGR. These identified materials will be considered along with other materials that may be identified.

2.4.2 Material Selection Process

The initial material selections that Vollman determined and any others identified as candidates will be down-selected based on the FEA and CFD simulations to determine whether they do indeed exhibit the needed properties before the physical tests are performed.

3.0 TASKS TO ACHIEVE TRL-4

Advancement to TRL-4 typically involves bench scale component verification. Since the hot duct is a component based on technology that to a large extent has been demonstrated in similar situations, analytical modeling is an acceptable alternative.

3.1 Initial Thermal Expansion Analysis

The purpose of this analysis is to determine both the radial and axial thermal expansion of the hot duct assembly and adjoining vessels. Both the internal interaction of the hot duct components and the interaction of the hot duct with the vessels must be considered.

Outputs from this analysis will be clearances between the hot duct components at all applicable temperatures at each tolerance extreme. Any external forces resulting from thermal expansion (either from the vessels on the hot duct or from the hot duct on the vessels) will be quantified, using industry standard pipe stress analysis methods for use in the FEA analysis.

The assumptions listed here for this analysis are expected to be limited to vendor data for thermal expansion. This data will be confirmed through the FEA analysis later in this test plan.

This calculation will be performed using a worksheet employing code qualified methods. Radial thermal expansion will be calculated with tolerances and tolerance stack-up taken into consideration.

The accept/reject criteria for this analysis will be based on whether the required hot duct component clearance (based on tolerances and thermal growth requirements) is acceptable. Additional criteria involve the quantity of force, if any, imposed on the adjoining vessel nozzles compared with allowable nozzle loads under combined loading.

Resource requirements for this analysis will be limited to qualified Engineers (usually one originator and one checker), the appropriate spreadsheet software and adequate computers on which to perform the analysis.

3.2 Upfront CFD Flow and Temperature Analysis

3.2.1 Objectives and Desired Outputs

Determine velocity profiles (and any potential flow induced vibrations), temperature distributions and heat transfer coefficients for the hot duct and insulation. Results from this analysis will be used to validate the selected materials or to further down-select from any remaining candidate materials.

3.2.2 Simulation Description

The model used for the analysis will be built to best approximate the component level test apparatus so simulation results can be validated later in the development plan. The conditions of service will be assigned as boundary conditions in the CFD model including ambient temperatures, flow magnitude and differential pressures. Hot duct and insulation material assignments for the simulation will be based on the initial material selections. The only material properties of concern for this analysis are the thermal properties since they will affect heat transfer within the model. The working fluid is known and can be assigned the appropriate properties. Figure 2 shows an example of a CFD result plot of the hot duct. This particular plot represents flow through the cross vessel.

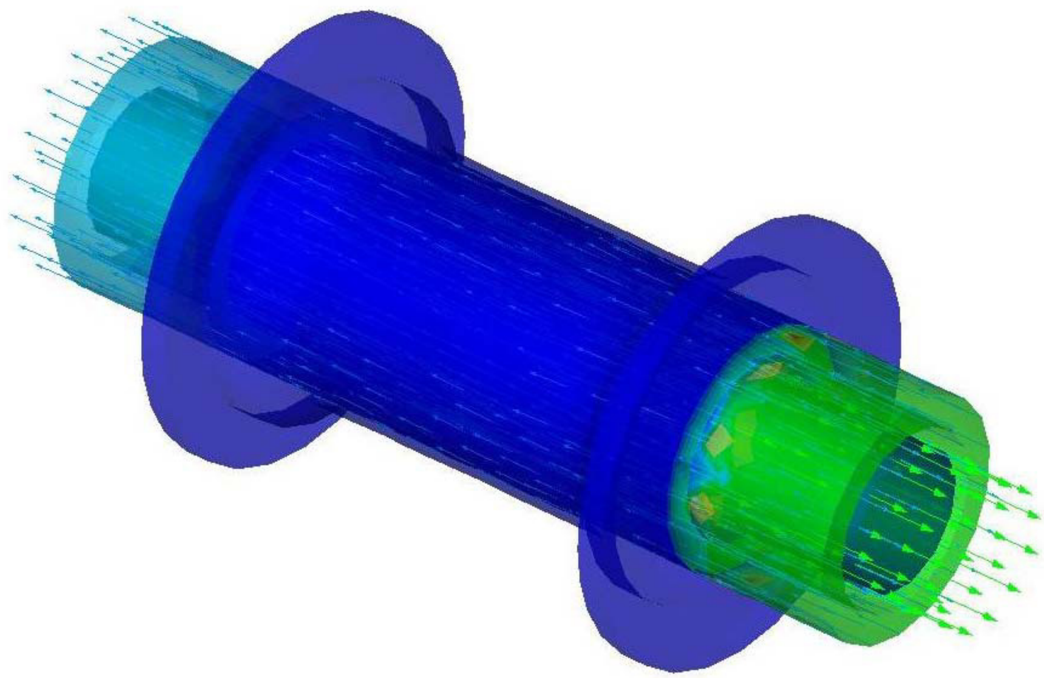


Figure 2: Example of a CFD Results Plot

3.2.3 Accept/Reject Criteria

These criteria will be based on the applicable codes and standards as well as any other pertinent design goals set forth by the project. The simulation results must verify temperatures do not exceed hot duct and insulation code allowable values.

3.2.4 Resource Requirements and Proposed Test Location

Resource requirements for this simulation will be limited to qualified Engineers (usually one originator and one checker), the appropriate computational fluid dynamics software and adequate computers (per QA requirements) on which to perform the analysis or analyses. The proposed test location is listed below.

URS-Washington Division Denver Office
7800 E Union Ave
Denver, CO 80237

Attn: Dave Carroccia
303-843-2038
dave.carroccia@wgint.com

3.3 Upfront FEA Stress Analysis

3.3.1 Purpose, Scope, Desired Outputs

This initial FEA analysis will be focused on the interactions between the insulation, duct and graphite. The thermal profile from the CFD analysis will be imported into the FEA model so thermal stresses can be calculated. All applicable loads will be applied simultaneously to ensure the hot duct can withstand the worst case conditions. Elevated temperature material properties will be used and the stress values compared to acceptable code acceptance criteria using commercially available material properties.

3.3.2 Assumptions and Approach

The hot duct, insulation and graphite will be modeled along with their connection hardware. This initial model will then be constrained and thermal loads from CFD will be applied along with pressure loads.

3.3.3 Accept/Reject Criteria

Stress values from the worst case loading scenario will be compared against allowable code values.

3.3.4 Resource Requirements

Resource requirements for this simulation will be limited to qualified Engineers (usually one originator and one checker), the appropriate finite element analysis software and adequate computers on which to perform the analyses.

3.3.5 Proposed Test Location

URS-Washington Division Denver Office
7800 E Union Ave
Denver, CO 80237
Attn: Dave Carroccia
303-843-2038
dave.carroccia@wgint.com

3.4 Hot to Cold Leak Detection Method

Leak detection requirements depend on the safety class. This plan assumes the most stringent safety class standard so a method of leak detection is included in this test plan. The method chosen will be validated by physical testing. Thermal imaging appears to be a viable option. Thermal imaging could be used to observe temperature trends within the cross vessel. Temperatures will vary based on power levels. However, the duct temperature profile should be relatively consistent at the various power levels.

Any hot to cold leg leak will likely be a trend based event, meaning that the leak will start small and grow in size over time. The thermal imaging software could be calibrated to look for thermal trends within the hot and cold ducts that are independent of power fluctuations. Gross variations in the ducts' thermal profiles could also be detected in the case of a sudden or catastrophic failure.

3.5 Insulation Connection Method

A method of connecting the insulation to the duct material must be determined. This method will be validated during TRL-5 development testing.

4.0 TASKS TO ACHIEVE TRL-5 (COMPONENT AND COUPON TESTING)

TRL-5 is defined as component verification at experimental scale. This level is meant to provide the necessary design data for complete component demonstration, but the test article does not necessarily need to be a model of the final component design. For the hot duct and insulation this involves three physical tests that are designed to provide final validation of the selected materials and initial validation of the insulation connection method. Once the tests are completed a life cycle cost analysis will be performed.

Coupon tests are an important precursor to full-scale physical testing. Testing coupons, or small sections of material, confirms that the material can withstand the rigorous requirements before investing in the expense of building a scale mock-up.

At the time of this plan's implementation all available pertinent material test data from other ongoing NGNP projects and studies will be gathered. If similar tests have been completed using the material(s) selected for the hot duct, the data collected can be used instead of re-creating the same tests. The sub-sections below explain each coupon test.

4.1 Required Test Facility Capabilities

The following test facility capabilities are required. These requirements apply to all tests completed as part of this test plan.

- High pressure helium storage capacity. The insulation and hot duct will both be subject to constant helium exposure. A source of pressurized, high quality helium must be available for a variety of testing.
- Helium heating capability. Testing facility must have capability of heating high purity, pressurized helium mentioned above for testing at elevated temperatures.
- Materials heating capability. Testing facility must have high temperature heat source, autoclave or similar for material testing at high temperatures. Facility must also be capable of producing and maintaining plant peak operational temperatures for operational testing, including temperature cycling. Raw material testing for the insulation and hot duct must be capable of testing at maximum plant operation temperatures as part of environmental qualification of materials.
- High accuracy Flow, Temperature, and Pressure Instrumentation. Testing facility will have all applicable flow, temperature, and pressure measurement devices available. These devices will be calibrated according to the applicable standards, and be subject to frequent inspection.

Proposed testing configuration will consist of a bank of pressurized, high purity helium canisters stored at room temperature. In-line filtration, resistance heaters, recirculation, and pressure boosting compressors will be available to produce a supply of clean, dry helium at elevated pressures and temperatures to the applicable testing rig.

Testing rigs will consist of flow verification equipment where they can be subject to scaled flows of primary coolant quality helium flows. Test rigs will be fitted with high accuracy, calibrated flow instrumentation to precisely meter and record observed flow, and flow characteristics.

Other test rigs will include high temperature "ovens" where selected materials can be subjected to high temperatures for short duration, and prolonged periods. High accuracy temperature, calibrated measurement and recording equipment will be available for use.

4.2 Hot Duct and Insulation Material Property (Coupon) Tests

Coupon tests are required where gaps exist between valid and traceable manufacturers data and the anticipated operating environment. Data acquired during testing must be of the suitable quality level and contain traceability information as specified in the quality requirements below. CFD and FEA results will be used to determine at what temperature and stress values the tests should be completed.

Unless otherwise specified in the sub-sections below, the following location is recommended for performing the coupon tests:

IMT Intermountain Testing
2965 S. Shoshone
Englewood CO 80110
1-800-742-5621
joe@intermountaintesting.com

4.2.1 Environmental Exposure/Embrittlement

This test will involve exposing the coupon to all chemicals, atmospheric impurities and environmental factors (such as temperature and pressure) it will experience during operation. This test will be the first coupon test conducted to allow for the maximum exposure to environmental conditions. Hydrogen embrittlement testing and halogen (Iodine) exposure data will be obtained.

These test values will be used to validate vendor data and provide information to fill in any gaps between the available data and expected operating temperature. Material performance data collected will be used for material down-selection. An estimate of the range of exposures to environmental risks will be necessary to perform this test.

4.2.2 Room Temperature and High Temperature Properties

Tests of material properties at normal and elevated temperatures that are conducted in compliance with ASTM A370 requirements will verify that the vendor data used in the simulations is accurate. All material properties used for the simulations must be verified. The CFD analysis results will be evaluated to determine the expected operating temperature of the components to determine the relevant range of temperatures the candidate material coupons should be subjected to. CMTR's (Certified Material Test Reports) will be provided by the testing organization. All applicable material property tests will be performed at the expected operating temperature.

- Yield tensile strength
- Ultimate tensile strength
- Impact test/fracture toughness
- Thermal expansion

4.2.3 High and Low Cycle Fatigue, Creep Rupture and Relaxation

A number of material coupons will be subjected to simulated operating environments and analyzed for both the high and low cycle fatigue properties and the creep and relaxation properties exhibited by the candidate materials. Properties must be obtained at elevated temperatures using ASTM E-139. These values will be examined against the expected values to be encountered over plant life. An estimate of thermal and mechanical cycles must be made to determine the expected level of service. High cycle fatigue specimens are usually cycled until failure, and the stress level and accumulated cycles at failure will be compared with the expected service conditions.

4.2.4 Weld Strength

The weld strength coupon test will involve producing weld samples for destructive testing to confirm the strength characteristics of the weld and the heat affected zone. The following tests, which may involve subjecting the specimens to high temperatures, will be conducted on the weld coupons:

- U-bend at the weld joint, with dynamically applied load
- Heat affected zone material properties and microstructure
- Creep and evidence of creep crack initiation or void formation

If joints to adjacent piping involve welds to dissimilar alloys, then dissimilar weld specimens will be tested. Weld procedure methods may have to be generated and utilized for this test.

An additional area of interest in this category is identifying applicable NDE methods. Methods that can be performed on-site will be useful for field welded joints, and shop applicable methods will be useful for factory welds. Obtaining data on the minimum flaw size detection level using these methods and comparing this with the critical crack size for dynamically stressed material at high operating temperatures will be useful for qualifying NDE methods.

4.2.5 Stress Corrosion Cracking

Several coupons will be subjected to accelerated corrosion conditions and then stressed to projected operating stress levels to investigate whether corrosion accelerates the propagation of cracks. Test Method: ASTM STP 1210 (Slow Strain Rate Testing for the Evaluation of Environmentally Induced Cracking).

4.2.6 Irradiation

Several coupons representing the hot duct will be subjected to the equivalent amount of radiation flux the actual hot duct is expected to endure throughout the plant's operating life. Metallurgical examination, microstructure evaluation, morphology and destructive strength testing will be completed and compared against the un-irradiated room temperature coupon performance. This testing will be accomplished in partnership with the US DOE National Lab efforts to qualify NGNP materials. Facilities for irradiation simulating the high flux fields found in a reactor environment exist only at INL (ATR) and ORNL (HFIR). Facilities for post irradiation metallurgical examination exist at Argonne, (Environmentally assisted cracking of reactor materials), Idaho (Hot Fuels Examination Facility or HFEF), and Oak Ridge (Irradiated Materials Examination and Testing or IMET).

4.3 Component Level Test

This component verification test will involve the insulation, duct and graphite connected using the method determined during TRL-4 development. The test apparatus will be subjected to the operating temperature and pressures and examined for the factors listed in the sections below. Figure 3 is a schematic of the component level test and Table 2 summarizes the test.

Table 2: Component Level Testing Summary	
Test Objectives	1. Verify insulation and duct material compatibility 2. Verify insulation connection method 3. Verify environmental qualification of insulation 4. Determine erosion and corrosion characteristics of materials (ablation rate) 5. FEA/CFD upfront simulations validation
Test Description	The insulation, duct and graphite will be assembled per the design. The duct will be supported and heated helium will be forced through the duct.
Conditions	950°C operating gas, Helium impurities, heated hot duct to reflect cold leg flow induced temperature
Configuration	See Figure 3
Duration	See Table 1
Test Location	**Hazen Research, Inc.** 4601 Indiana Street Golden, Colorado 80403 http://www.hazenusa.com/ Phone: (303) 279-4501 Fax: (303) 278-1528
Measured Parameters	1. Temperature of duct outer surface at several locations 2. Temperature at outer surface of insulation at several locations 3. Corrosion between insulation and duct 4. Strain in the duct at critical locations
Data Requirements	Quantification of impurity particulate dispersal, temperatures at outer surfaces of insulation and duct at several locations, flow measurements
Test Evaluation Criteria	1. Ability of insulation to protect hot duct from temperatures that exceed its code allowable temperature 2. Durability of insulation to duct connection. 3. Acceptability of differential corrosion between insulation and hot duct.

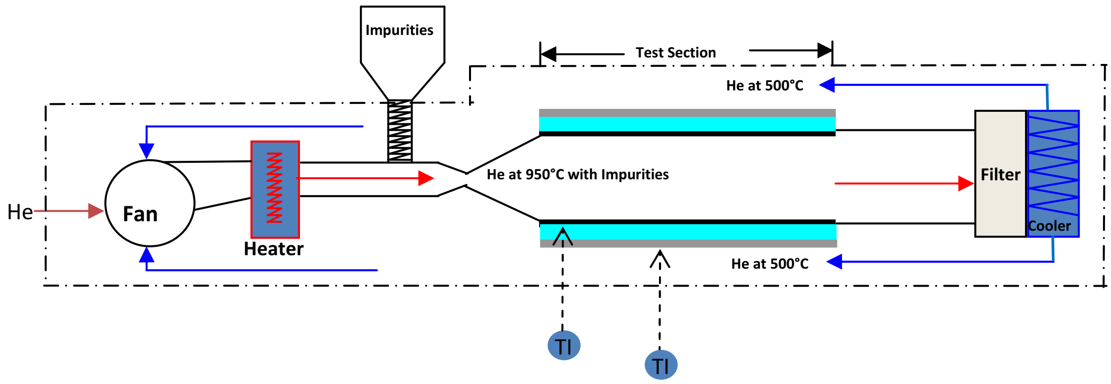


Figure 3: Component Level Test Schematic

4.3.1 Insulation Connection Method

The chosen method of connecting the insulation to the hot duct will be verified in this TRL step. The test will confirm the connection through replication of process temperatures, gas impurities and flow magnitude. Further validation of the insulation connection method will be provided during TRL-6 development when differential pressures are applied at the sub-system level testing.

4.3.2 FEA/CFD Simulations Validation

The test apparatus will correspond to the CFD and FEA models. The measured strain, temperatures and flow characteristics will be compared to the FEA and CFD results. The difference between predicted and observed results will be quantified and any discrepancies will be used to adjust the model. The determined discrepancies will also be used for the risk assessment done later in this plan.

4.3.3 Insulation Performance

The insulation performance will be gauged by the heat transfer coefficient calculated from measured data. Several temperature indicating devices will measure duct temperature at several points to gauge insulation effectiveness.

4.4 Acoustic and Flow Induced Vibrations Test

Acoustic interactions were considered both in the design of the hot duct and as part of the CFD analysis. The physical test outlined below in Table 3 will verify the design performs as intended.

Table 3: Acoustic and Flow Induced Vibrations Test	
Test Objective	Determine frequency spectra and sound pressure levels generated by hot duct assembly as a function of flow velocities and geometry. Determine that the helium flowing through the hot duct will not cause dynamic instabilities.
Test Description	The flow-induced vibration test will represent all relevant design details. Initial CFD results will be confirmed through wind tunnel testing.
Conditions	Operating flow velocities and gas temperatures
Configuration	A ¼ model of the hot duct assembly will be used for wind tunnel testing. Speed of sound in air is about ¼ the speed of sound in helium, so a wind tunnel with ¼ scale air flow will be used to match sound wave velocity. The test configuration details will be determined by the test facility.
Duration	See Table 1
Test Location	Possibly ANL or commercial facility set up for wind tunnel testing
Measured Parameters	Temperature, pressure, test frequency spectra and measured resonant frequencies
Data Requirements	Representative of measurement parameters
Test Evaluation Criteria	Verify that the hot duct's operational frequencies do not match resonant frequencies predicted by wind tunnel testing

4.5 Life Cycle Cost Analysis

This analysis will be completed once the material properties and selection have been confirmed. This analysis considers material cost, fabrication cost, and all associated operating inspection and maintenance costs to produce an expected life cycle cost for the hot duct and insulation.

4.6 RAMI Analysis

As part of an integrated plant program, Reliability, Availability, Maintainability and Inspectability (RAMI) analysis will be performed to ensure that hot duct will meet mission needs safely and reliably with minimum life cycle cost. The RAMI analysis task involves a process of identifying top-level (major) system availability requirements, decomposing these requirements into meaningful downtime statements for subsystems and/or components, and formally summing these downtimes to estimate the availability of the entire interactive system.

Standard engineering reliability methods are utilized to determine the mean time of service up to failure for the component(s) in question. The reliability analyses for each sub-component of the system being analyzed are combined to calculate the mean time to failure (MTTF) for the analyzed system. Industrial data on existing equipment of a similar nature within the nuclear power plant environment will be utilized to determine MTTF. Previous HTGR data will also be researched.

The next step in establishing a RAMI program is to develop a requirements statement to define the following parameters.

- Operational needs for the design life of the component
- Expected normal and worst-case operating conditions
- Expected downtime for inspection and either corrective or preventive maintenance actions.

The requirements statement is used to create an availability statement for the plant. Stating the total uptime needed for the system or subsystem establishes the allowable downtime. The total downtime is then allocated to all the lower tier (component level) systems in the form of design requirements. After the component downtime is allocated to each of the involved subsystems, analytical techniques are used to estimate the actual downtime expected to be experienced by the various subsystems during operation. These estimates include failure frequency (FF) and the mean time required to return the failed system to operational status, or mean time to restore or repair (MTTR). The estimates are then summed to estimate the availability of the system as designed and compared with the availability requirement (A) as a measure of design success.

$$A = MTBF/(MTBF + MTTR)$$

The ease of maintainability of the component contributes to the mean time to restore. Components designed to facilitate maintenance will, in turn, contribute to the system's overall availability. Inspectability, built into the hot duct design, allows for operational parameters and performance to be closely monitored allowing preventive maintenance to be scheduled with greater efficiency. As a part of a larger RAMI program, this allows for coordinated and more precisely scheduled maintenance that helps eliminate maintenance when it's not needed and encourage maintenance that positively impacts availability. Improved inspectability and performance monitoring also helps to prevent unanticipated outages due to in-service faults.

The hot duct and insulation achieve the inspectability requirement because they will be verified initially then inspected periodically during service to ensure proper functionality. They will either be removed from service for testing during a re-fueling or other planned outage, or tested in-place.

Consideration of performance of maintenance will be a priority in the design of the hot duct and insulation.

4.7 Endurance Limit Analyses

The endurance limit of a component is determined through an analysis that considers all factors that contribute to the expected component life including static and cyclic loads, temperature, creep, fatigue, erosion, corrosion and other factors. Localized stresses from FEA analysis combined with CFD results for local and component temperatures will be utilized in the Endurance Limit Analyses.

The recommended analysis expert is:

Becht Nuclear Services
http://www.bechtns.com/
2415 Campus Drive, Suite 275
Irvine, CA 92612
949-660-1480

4.8 Creep Analysis

The purpose of the creep analysis is to ensure that the materials (hot duct and supports) do not permanently deform under the influence of high temperatures and stresses (below acceptable code values) over an extended period of time. Both analytical modeling (FEA) and coupon tests will be utilized in the Creep Analysis. Creep analysis makes use of non-linear modeling techniques to be performed by subject matter experts.

The recommended analysis expert is:

Becht Nuclear Services
http://www.bechtns.com/
2415 Campus Drive, Suite 275
Irvine, CA 92612
949-660-1480

4.9 ALARA Analysis

The purpose of the ALARA analysis is to ensure that radiation doses to workers are as low as reasonably achievable under the anticipated operating conditions and modes including inspection and maintenance. The hot duct assembly will be examined for potential contamination traps, which could lead to increased exposure during maintenance activities. The radiation dose to the exposed workers will be estimated by using 3d modeling techniques that incorporate materials of construction of the components and other nearby radiation sources as well as a portion of the physical environment the hot duct and components resides within. ALARA trained personnel will consider personnel protection requirements, and if temporary shielding is needed, then this too will be incorporated in the model. The recommended organization to perform this analysis is as follows:

URS-Washington Division
Paul Reichert
510 Carnegie Center
Princeton, NJ 08540, United States
(609) 720-3210
www.wgint.com

4.10 LIMIT Analysis

Welding Research Council (WRC) Bulletin 464 describes sizing of vessels using modern limit analysis. LIMIT analysis will be used to validate the results of the FEA analysis; it will serve as an independent check on the allowable wall thickness of proposed hot duct.

Achieve vessel sizing by closed-form formulas, equilibrium relations of free bodies, and finite element lower bound analyses. When coupled with a finite element analysis, a lower bound analysis is an effective tool for the sizing of any vessel or its components.

The recommended analysis expert is:

Becht Nuclear Services
http://www.bechtns.com/
2415 Campus Drive, Suite 275
Irvine, CA 92612
949-660-1480

5.0 TASKS TO ACHIEVE TRL-6 (SUB-SYSTEM TESTING)

This development level involves verifying the design at the sub-system level. This sub-system is comprised of the hot duct, insulation, graphite, bellows and cross vessel. The FEA and CFD simulations completed for component level advancement will be built upon to include all system components. Simulations will be completed and performance predictions documented prior to the physical test. The physical test will be comprised of only the sub-system components but be constructed such that this test apparatus can be used for the expanded system test completed during TRL-7 development.

5.1 FEA Simulation System Optimization

The FEA model developed for Component level testing will be expanded to include all system components. This model will include the differential operating pressures between the hot and cold legs. The entire system is being modeled so all interactions between the hot duct, cross vessel, reactor vessel, and PCU can be fully accounted for to ensure an integrated design. Several FEA analyses will be performed to investigate stress, strain and deflection to determine the following:

- Duct stress
- Adequacy of end joint connections (bellows, cross vessel and vessels)
- Response to temperature loads
- Differential thermal expansion
- Stress imposed on vessel nozzles versus allowable values
- Response to external loads and design basis hazards

The knowledge gained from this system simulations will be applied to the sub-system level test described below in Section 5.3.

5.2 CFD Simulation System Optimization

The CFD analysis from the Component level simulation, like the FEA analysis, will be expanded to include all Sub-System components. This analysis will also include differential pressures between the hot and cold legs. The analysis results will be verified by the Sub-System level test outlined in Section 5.3. The goal of this simulation is to determine and optimize the inlet and exit flow conditions.

5.3 Sub-System Level Test

This level's sub-system test will include all differential pressures and operating temperatures over the range of operating conditions. The test will be analyzed for the items outlined in the sections below. See Section 4.1 for required testing facility capabilities. Figure 4 shows the sub-system level test apparatus and Table 3 summarizes the test.

Table 4: Sub-System Level Testing Summary	
Test Objective	Verify sub-system level design 1. Leak detection method verification 2. In-service inspection verification 3. Verification of sub-system under operational pressures
Test Description	The insulation, duct and graphite will be assembled per the design. The duct will be supported on one end and heated gas will be forced through the duct.
Conditions	950°C operating gas, differential cold and hot duct operating pressures
Configuration	See Figure 4
Duration	See Table 1
Test Location	**Hazen Research, Inc.** 4601 Indiana Street Golden, Colorado 80403 http://www.hazenusa.com/ Phone: (303) 279-4501 Fax: (303) 278-1528
Measured Parameters	1. Temperature of duct at several locations (including initial vs. end of life temperatures to measure decrease in insulation effectiveness) 2. Temperature at outer surface of insulation at several locations 3. Strain in duct
Data Requirements	Representative of measured parameters
Test Evaluation Criteria	The hot duct must be shown to have adequate strength based on applicable code acceptable stress values for the measured maximum hot duct temperature. Hot duct temperatures must remain below acceptable code limits. Leak detection and in-service inspection techniques must be shown to be valid.

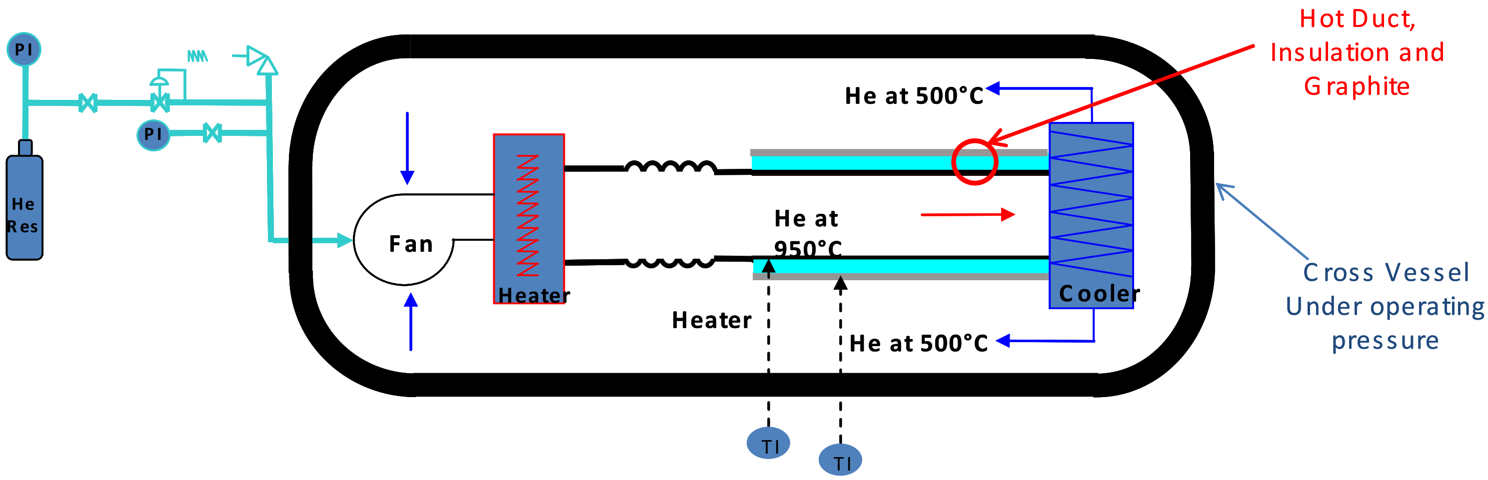


Figure 4: Sub-System Level Test Schematic

5.3.1 Leak Detection Method Initial Concept Verification

The leak detection concept outlined in the previous TRL step will be tested for feasibility. The amount of testing possible or relevant to validate the concept will depend on the method decided upon. Thermal imaging, if chosen, could be initially tested but would also require verification at the system level later in this development plan.

5.3.2 In-Service Inspection Techniques Validation

The Inspection techniques laid out in the RAMI analysis will be validated during this sub-system test. The cross vessel is the pressure boundary for the NGNP; the cross vessel represented in the sub-system tests will recreate actual operating challenges with respect to leak detection. The techniques must be demonstrated to be efficient and with plant personnel exposure.

5.4 Final Thermal Expansion Analysis

The final thermal expansion analysis will include all system components and the interaction between the hot duct, cross vessel, PCU vessel, reactor vessel and vessel support frame. Existing data on these interactions may substitute for this analysis. The thermal expansion information will be applied to design of the Sub-System and System test apparatus.

6.0 TASKS TO ACHIEVE TRL-7 (INITIAL SYSTEM TESTING)

This development level tests all components of the system at Engineering scale and prepares the system for integrated CTF testing if deemed necessary.

6.1 Testing of Integrated System

As previously stated, the cross vessel, hot duct, reactor vessel and PCU vessel all interact as a system. It is necessary to represent all aspects of this system so the stress and deflection from the system's interaction can be captured. See

Section 4.1 for required testing facility capabilities. Table 5 summarizes the test information.

Table 5: 1:10 Scale System Level Testing Summary	
Test Objective	Verify system level design under operating temperatures and pressures 1. Stress caused on or as a result of vessel nozzle connections 2. Installation techniques verification 3. Validation of optimized FEA analysis
Test Description	The insulation, duct and graphite will be assembled per the design within the cross vessel. Heated gas and differential operating pressures will be applied to the test apparatus.
Conditions	950°C operating gas, operating pressures, differential cold and hot duct operating pressures, reactor vessel operating temperature
Configuration	See Figures 6 and 7
Duration	See Table 1
Test Location	**Hazen Research, Inc.** 4601 Indiana Street Golden, Colorado 80403 http://www.hazenusa.com/ Phone: (303) 279-4501 Fax: (303) 278-1528
Measured Parameters	1. Temperature profile of duct at several locations 2. Temperature at outer surfaces of insulation and hot duct at several locations 3. Strain in hot duct 4. Gas flow
Data Requirements	Representative of measured parameters
Test Evaluation Criteria	The hot duct must be shown to have adequate strength based on applicable code acceptable stress values for the measured maximum hot duct temperature. Hot duct temperatures and stress levels must remain below acceptable code limits. Installation techniques must be shown to be valid. Thermal expansion must not cause any stresses that were not originally accounted for. The inspection techniques must also be validated.

6.1.1 Interface with Adjoining Structures

Interface details of the hot duct sub-assembly with adjoining structures are needed and will be detailed in the FEA model. These include the interfaces to the power conversion unit and the reactor pressure vessel internals. Connections, nozzles or penetrations and required field welds shall also be shown and analyzed in the FEA model.

The reactor vessel core support structure provides support for the lower plenum and a path for the primary coolant through the hot duct. It also must maintain structural integrity of the reactor vessel and a coolable core geometry during postulated licensing basis events. See Figure 5.

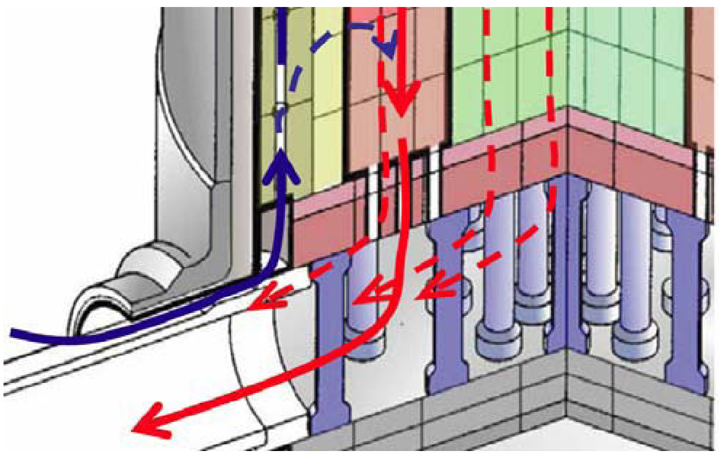


Figure 5: Reactor Pressure Vessel Interface

The hot duct is enclosed in the cross vessel between the power conversion system (PCS) interface (e.g., steam generator) or the IHX for process heat. The cross vessel is an ASME III Class 1 component that comprise part of the RCPB.

6.1.2 Installation Techniques

Installation techniques shall be established based on existing design information, along with results generated by analytical modeling. This is to include any required field welds and the controls necessary to ensure their acceptability. Any applicable installation drawings will also be developed.

6.1.3 Test Apparatus

The test apparatus for the hot duct shown in Figure 6 is fabricated to be 1:10 scale of the MHTGR system. An FEA can be correlated to the test article and used to predict the stresses in the actual article. WRC-107 [Wichman, 2002] may be effectively used to get preliminary ranges of loads in the test article that correlate with the expected stresses in the actual article. The WRC-107 is not part of the code, but is generally accepted within the code analysis community and is considered an empirically based approximation suitable for defining model scaling factors.

The approach of building a 1:10 scale test apparatus results in a reduction of cost and lead time compared to building a full scale test apparatus while providing results suitable for simulation validation and demonstration of the hot duct effectiveness. The full scale reactor vessel is approximately 75 feet tall and 24 feet in diameter with a 10-inch wall thickness. This is an extremely large vessel that limits the number of qualified fabrication shops, creates transportation challenges, substantially increases material cost, uses enormous resources, and limits the test facilities capable of supporting such large equipment.

The pressure vessels expand and contract due to pressure and temperature loads causing deformation of the vessel and nozzle. This deformation results in stresses on the nozzles that must be depicted and understood to accurately represent the localized reaction loads at the connections.

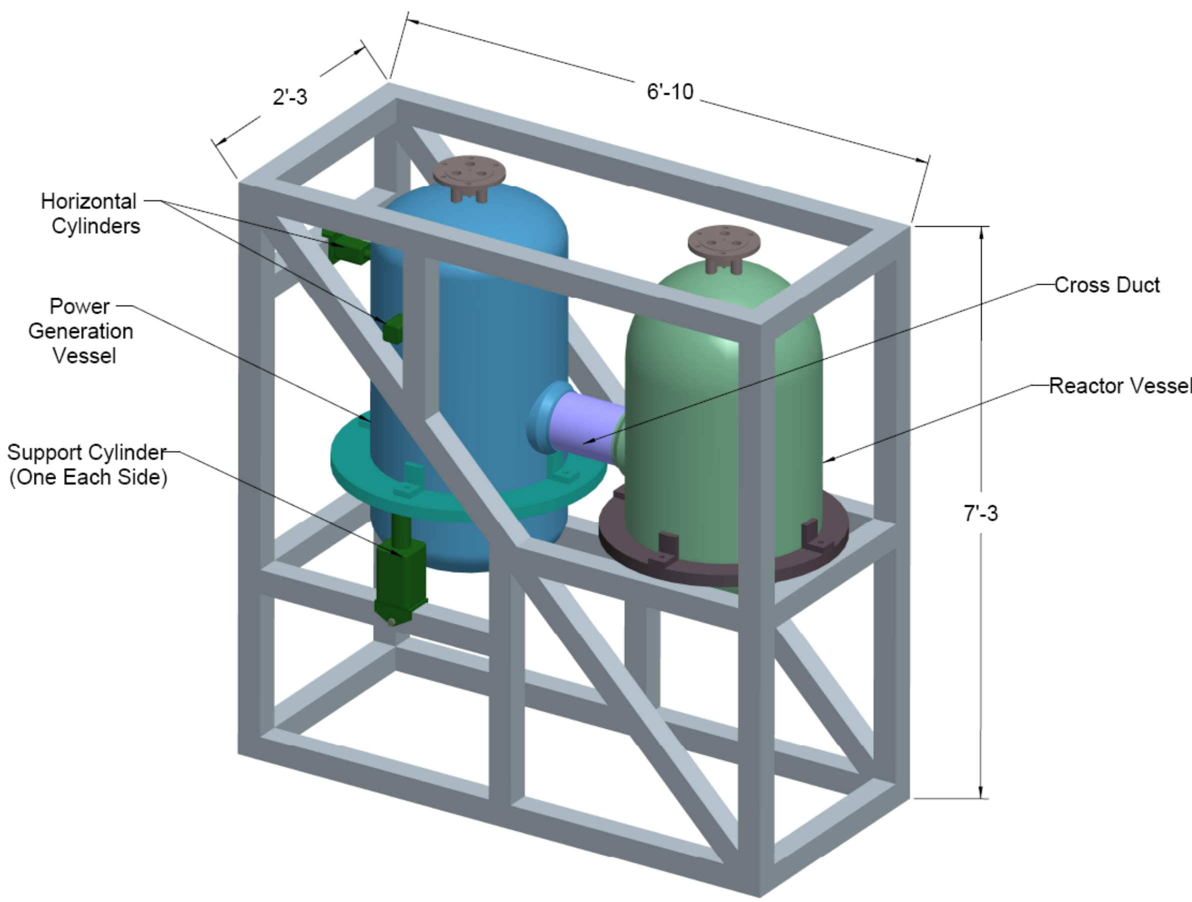


Figure 6: System Level 1:10 Scale Test Apparatus

It is not necessary to represent the full height of the pressure vessels, although adequate vessel height above and below the vessel nozzles must be represented to adequately capture local stress effects from the nozzles. Section III of the ASME Pressure Vessel Code states that local stress effects from a pressure vessel penetration extends in the meridianol direction not more than the square root of Rt, where R is the mean vessel radius and t is the vessel thickness.

The reactor vessel (right side of Figure 3) is fixed to the frame near the bottom of the vessel and supported by guide pins at the top (guide pins not shown). The frame is sized large enough to accommodate the vessels' thermal growth and the reactor vessel is mounted on slots to allow for thermal expansion. Various vessel constraint methods are possible and will be considered depending on test goals.

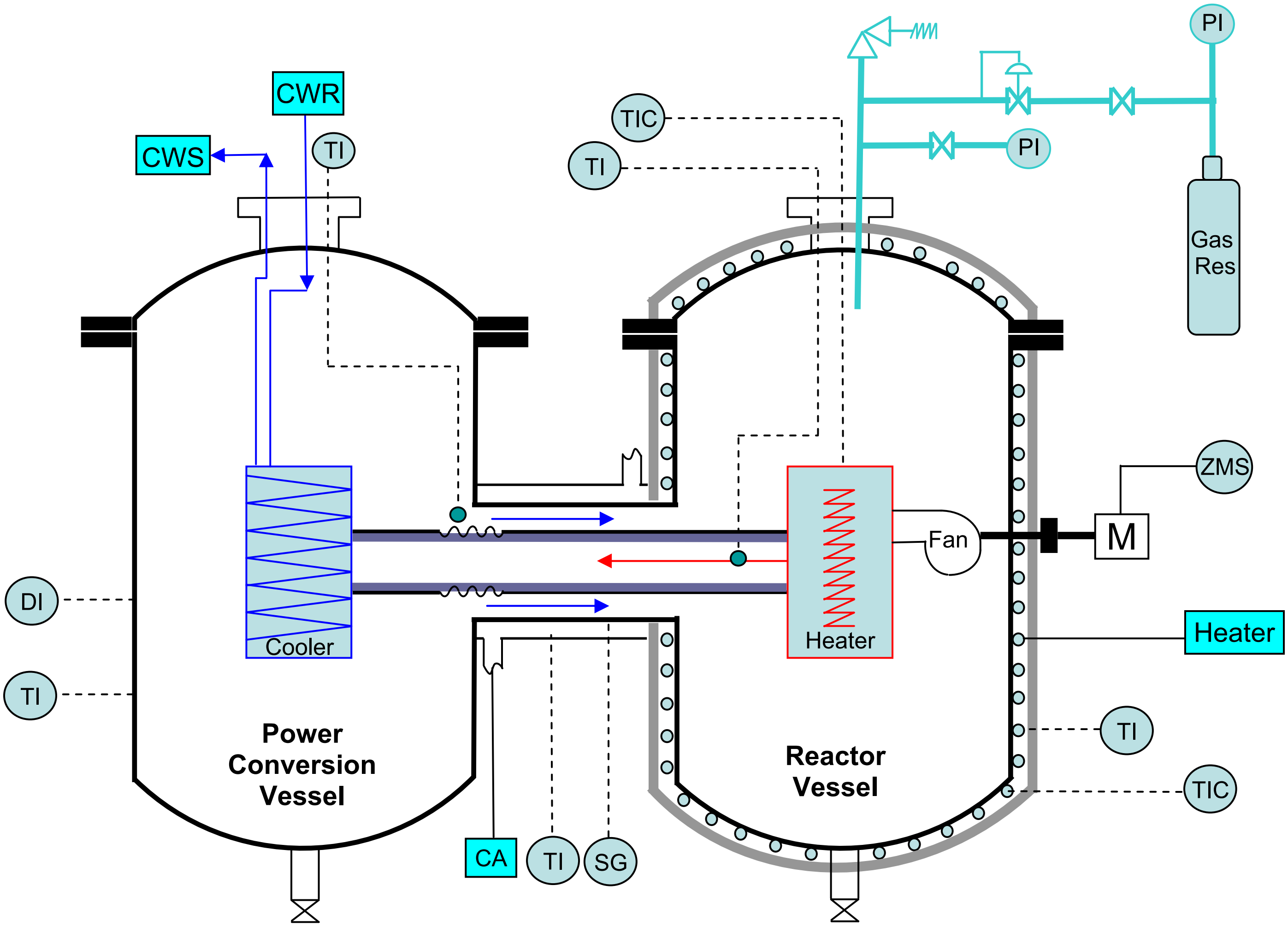
The support cylinders shown in Figure 3 allow for DBA and other external loads to be applied to the vessels in the test apparatus. The operating conditions (temperature, pressure) will be imposed to represent the resulting thermal growth and subsequent axial nozzle loads. The large cylinders shown for the power conversion vessel both support the vessel and allow for the external loads to be applied. These support cylinders are pinned to the frame to allow horizontal thermal growth of the hot duct and power conversion vessel.

Figure 4 shows a schematic of the test apparatus with the required instrumentation. A bead heater and insulation encloses the reactor vessel in order to achieve the reactor vessel operating temperature. This is necessary to create the vessel's thermal expansion and the resultant nozzle displacement and reaction loads. The hot duct bellows is represented in the test apparatus to accurately exhibit thermal stresses.

The hot duct and Power Conversion Unit (PCU) will achieve their operating temperatures the same way they are achieved during actual operation; by conduction through the hot duct and convection from the re-circulating gas. Figure 7 below shows the piping and valve schematic for introducing and safely controlling the gas pressure. The working gas will be supplied from a pressurized reservoir with sufficient pressure (accounting for heat induced pressure increase) to achieve the desired system operating test pressure.

The gas is introduced into the reactor vessel through the vessel's top nozzle. The gas is then propelled with the reactor vessel's internal fan through a gas heater. The gas from the heater is sent through the hot duct to the gas cooler in the PCU. The cooled gas is expelled into the PCU, differential pressure moves the cool gas through the cold duct back into the reactor vessel where the fan again moves the gas through the heater and into the PCU via the hot duct. The fan is powered via an external motor whose driveshaft penetrates through a mechanical seal in the vessel's side.

Figure 7 also features the basic instrumentation necessary to achieve the test objectives. Several temperature indicating devices are shown representing the components and gas streams whose temperatures must be measured. Multiple temperature indicating devices will be necessary for each vessel and to measure the gas streams although only one is represented for each in the schematic. Other instrumentation includes displacement indicators, multiple strain gages and the appropriate data acquisition and motor control systems. Required utilities include a 240 volt 3-phase power source for the heaters, cooling water, working gas and hot duct shroud cooling gas.



LEGEND

SG – Strain Gages
PI – Pressure Indicator
CWS – Cooling Water Supply
CWR – Cooling Water Return
DI – Displacement Indicator
CA – Cooling Shroud
TI – Temperature Indicator
TIC – Temperature Indicating Controller
ZMS – Motor Control
Air Duct Insulation
Vessel Insulation

Figure 7: System Level 1:10 Scale Test Apparatus Schematic

6.1.4 CFD/FEA Optimization System Simulation Validation

The data gained from the System test will be compared against the final FEA analysis performed earlier in this development step. This comparison will be a further validation of the FEA simulation results from Component testing.

6.2 Risk Assessment for CTF Testing

A risk assessment will be performed to determine the tradeoff of full scale integrated CTF testing versus the 1:10 scale test apparatus outlined above. This assessment will in part be based on the CFD and FEA results and the level of confidence that those results accurately represent the expected experimental results. Several coupon and small scale tests have been outlined in previous sections; the results from these tests will also help determine whether integrated CTF testing is desired.

7.0 TASKS TO ACHIEVE TRL-8 (FINAL SYSTEM TESTING)

This TRL step involves full-scale integrated CTF testing. Integrated CTF testing, if found to be necessary by the risk assessment performed earlier in this plan, will be completed in conjunction with other reactor component tests at the Component Test Facility at INL. This testing would be coordinated with testing of other NGNP components.

8.0 QUALITY ASSURANCE REQUIREMENTS

8.1 Quality Assurance Program

All aspects of the QA plan shall be compliant with the Quality Assurance Program Plan (QAPP) of General Atomics. A recommended outline is provided below for the proposed QA program.

8.1.1 Program and Organization

8.1.2 Training

8.1.3 Personnel Requirements

8.1.4 Limiting Conditions

8.2 Design, Engineering and Data Control

8.2.1 Inputs

8.2.2 Drawings

8.2.3 Specifications

8.2.4 Criteria Documents

8.2.5 Revisions

8.2.6 Change and Configuration Control

8.2.7 Design Analysis

8.2.8 Design Review

8.3 Verification

8.3.1 Alternate Calculations

8.3.2 Design Review

8.3.3 Testing Under Most Adverse Conditions

8.4 Procurement

8.4.1 Procurement Document Control

8.4.2 Review

8.4.3 Approval

8.4.4 Handling, Storage and Shipping

8.4.5 Instructions, Procedures and Drawings

8.4.6 Control of Purchased Items and Services

8.4.7 Certification

8.4.8 Source Verification

8.5 Inspection

8.5.1 Shop Inspection

8.5.2 Post Installation Inspection (field)

8.5.3 Control of Special Processes

8.5.4 Test Control

8.5.5 Control of Measurement and Test Equipment

8.5.6 Result Documentation

8.5.7 Inspection, Test and Operating Status

8.6 Identification and Control of Items

8.6.1 Control and Disposition of Supplier Nonconformance

8.6.2 Corrective Action

8.6.3 Commercial Grade items

8.6.4 QA Records

8.7 Audits

8.8 Approval

9.0 PROPOSED TEST LOCATION

Test location will be determined based on the physical testing needed. Analytical simulations may be performed at the URS – Washington Division. Scale testing, if required, may also be managed by the URS - Washington Division.

10.0 SCHEDULE

An outline schedule of the Hot Duct and Insulation Test Plan is provided below. .

<table>
<tr><th rowspan="2">Readiness Level</th><th colspan="12">Year (FY 20xx)</th></tr>
<tr><th>09</th><th>10</th><th>11</th><th>12</th><th>13</th><th>14</th><th>15</th><th>16</th><th>17</th><th>18</th><th>19</th><th>20</th></tr>
<tr><td rowspan="3">NGNP Schedule</td><td colspan="4">Conceptual Design
Prelim Design</td><td colspan="8">Final Design for NGNP</td></tr>
<tr><td colspan="8">Site Work</td><td colspan="4">Construction</td></tr>
<tr><td colspan="10"></td><td colspan="2">Startup / Testing</td></tr>
<tr><td>CTF</td><td colspan="12">================================</td></tr>
</table>

TRL-3[(1)]	= >
TRL-4	< = = >
TRL-5	< = = >
TRL-6	< = = >
TRL-7	< = = = = >
TRL-8	< = = = = >

[(1)] COS and other design bases provided in a timely fashion to determine test parameters

11.0 REFERENCES

[PSER, 1986] “Draft Pre-application Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor (MHTGR), March 1989.

[PSER, 1996] “Draft Copy of Pre-application Safety Evaluation Report (PSER) on the Modular High-Temperature Gas-Cooled Reactor (MHTGR), February, 1996.

[PSID, 1992] DOE-HTGR-86-024, Preliminary Safety Information Document for the 350 MW Standard MHTGR, Amendment 13 issued August 7, 1992.

[Richards, 2008] “Reactor Containment, Embedment Depth, and Building Functions Study” RGE 911124, Rev. 0, General Atomics, San Diego, CA, September 2008.

[Vollman, 2008] “NGNP Composite R8D Technical Issues Study”, RGE 911125 Rev 0. General Atomics, San Diego, CA

[Wichman, 2002] “Local Stresses in Spherical And Cylindrical Shells Due to External Loading”, Welding Research Council Bulletin 107, New York, NY

Appendix A

Statement of Work Example

URS
Washington Division

Statement of Work

To

Name of Organization

For

State: Technologies or Services

Hot Duct Testing Program

Rev. X

Date

Project Number

Appendix A - Table of Contents

Tables

Appendix A – Statement of Work Example

1. SCOPE

A Statement of Work (SOW) will be provided to each sub-contractor responsible for fulfilling an element of the test plan. This SOW will state the services to be provided by the sub-contractor to satisfactory accomplish the test plan element. These services shall consist of:

- Appropriate indoor/outdoor facilities with necessary infrastructure and utilities to house the test process.
- Certain pieces of process equipment currently owned by the Consultant.
- An appropriately trained and qualified workforce to assemble, modify, calibrate, operate, and disassemble the test equipment.
- Analytical services to support testing operations.
- Appropriately permitted facilities and services for the disposition of gaseous, liquid, and solid wastes and effluents resulting from testing operations (if applicable).
- Procurement services for process consumables, leased equipment, and certain services, as needed.
- Procurement services for certain new equipment items designed or specified by the Buyer.
- Shipping of product samples to XXX to conduct additional testing or inspection.

2. BACKGROUND

3. SCHEDULE AND DELIVERABLE MILESTONES

Table 1 Schedule Milestones

Task	Milestone Date After Subcontract Award*	Responsible Party
Subcontract Award		
Complete New Equipment Procurement		
Complete Procurement of Consumables		
Complete Modification of Existing Equipment		
Complete Operating Procedures		
Complete Equipment Checkout, Calibration and Functional Tests		
Complete Readiness Review		
Begin preliminary qualification Tests		
Complete preliminary Tests		
Begin Production Tests		
Complete Production Test		
Complete Analytical Testing	XX days after completion of test	

Submit Test Report and Project Records DVD.	XX days after completion of Production Run	

*Milestone dates may change at the discretion of the Buyer.

Buyer will provide the following personnel to help accomplish the work scope. These people will work as an integrated team with the Consultant. (Example)

- One (1) to three (3) engineers to support equipment modification and assembly.
- Engineers and technical specialists to facilitate checkout and the functional testing.
- Two (2) technical engineers per shift plus other technical support personnel to facilitate the preliminary and production test runs.
- QA and environmental compliance / waste management personnel, as needed.

The technical engineers will provide technical process direction. The Consultant shall provide the operations supervisor, who will have overall authority for the operation of the pilot plant, and the process operators and technicians to operate and maintain the process equipment.

4. OVERVIEW OF THE HOT DUCT TESTING PROGRAM.

5. PROCESS SYSTEMS AND EQUIPMENT

6. CONSUMABLES AND UTILITY SERVICES

6.1 Utility Services:

Table 2 Utility Requirements

Utility	Expected Requirements
Steam	
Nitrogen	
Oxygen	
Instrument Air	

6.2 Electrical Supply

The process equipment requires the following estimated electrical services as given in Table 3. Consultant shall provide the required MCCs and cabling.

Table 3: Electrical Supply Requirements (Example)

Electrical Equipment	Expected Requirements
	TBD kW, 480 VAC, 30-Phase, 60 Hz
	TBD kW, 480 VAC, 3-Phase, 60 Hz
	60 kW, 480 VAC, 3-Phase, 60 Hz
	TBD kW, 480 VAC, 3-Phase, 60 Hz
	As required
	As required
	As required
	As required
	As required
	As required
	2 Hp, 480 VAC, 3-phase, 60 Hz

7. MATERIALS AND SERVICES PROVIDED BY BUYER (EXAMPLE)

- Test manager
- Technical engineers on each shift to provide technical direction of all test activities including: operating conditions, frequency and quantity, sampling frequency and methods, operating temperatures and parameters, and other process operational functions, in consultation with Consultant;
- Test plan and matrix;
- Operating procedures with assistance of Consultant;
- Design Basis Document (DBD), including equipment lists, instrument lists, valve lists, equipment descriptions, mass and energy balance, and equipment modification sketches;
- Quality Assurance Project Plan (QAPP);
- Sampling and analyses specification;
- Others as required

As noted above, please refer to the DBD for the major equipment lists.

8. MATERIALS AND SERVICES PROVIDED BY CONSULTANT (EXAMPLE)

Consultant shall provide the following equipment, supplies, and services:

- Flow meter suitable for measuring density, mass, and volumetric flow rate – manufacturer and model to be approved by Buyer.
- Data Acquisition and Control System (DACS): Use XXX system. The DACS is for monitoring, recording, and trending all instruments shown on the flow diagram and controlling all automated valves and heaters. DACS shall include HMI with at least three computers and monitors, one for use by Consultant operations personnel, one for Buyer operations/technical personnel, and one for reviewing trends and data. The DACS shall include a process diagram that shows all remote instrument readings overlaid on corresponding process equipment graphics. Scope includes programming HMI, emergency shutdowns, and alarm and trend windows;
- Operational Analytical Services: Consultant shall perform analytical testing of samples as specified in Table 4. In addition, Consultant shall provide a "mini-lab" containing a microscope for product inspection, analytical scale with accuracy to 0.001g for use during operation.

- Labor to pull process samples for the above analyses... The specific controls will be specified in the final test plan or QAPP.
- Instrument calibration services by qualified persons using traceable standards. Copies of all instrument calibrations shall be retained and an instrument control database shall be maintained that completely describes the process instrumentation and its calibration.
- Labor to assemble all equipment and operate all process equipment and utility services for the checkout and start-up work and on a 24/7 basis for functional tests, preliminary tests, and production test runs.
- Disposal of all waste materials, including hazardous wastes.
- Clean-up and disassembly of the equipment following the test program.
- Good safety practices, to include facility safety training for all personnel; consistent and proper use of required personal protective clothing; and consistent use of good industrial safety practices.
- High caliber conduct of operations, including the generation and compliance with appropriately detailed and approved operating procedures, repeat-back of operating instructions, and the maintenance of accurate and fully legible data sheets and a sequential operating logbook.
- In conjunction with Buyer, operating procedures for all process, calibration, and analytical activities.
- A final hardcopy data report and Project Records DVD as detailed in 14.

9. THE TESTING PROGRAM

9.1 Testing Program Overview

The testing program will be carefully planned and implemented to ensure all consumables and hardware are available and ready to meet the required test schedule. There will be functional and preliminary tests, followed by production test runs. The preliminary sequence and estimated duration of the test operation is given below. The specific tests and operations will be specified in the test plan. Actual timeframes may vary.

- Checkout and Functional Tests (~XX days)

  Following all equipment modifications and calibrations, functional tests will be performed. The functional tests will verify that all systems and equipment function as designed. These tests will be performed and documented in accordance with the existing procedure for functional tests

- Scoping Tests (~XX days)

  Preliminary tests will be performed in accordance with the test plan and approved operating procedures. They will establish baseline operating parameters for the production runs, such as ...

- Production Runs (~XX days)

  Production runs will be conducted around the clock for approximately XX. These runs will be conducted in accordance with the test plan and approved operating procedures.

10. ANALYTICAL SPECIFICATION

Throughout the test program the process inputs and outputs will be periodically sampled. Certain samples will be analyzed at the Consultant's lab. Others will be shipped off-site for analysis by other labs under contract to the Buyer.

Consultant shall generally pull the samples listed in Table 4 and perform the indicated analyses. Final sampling and quality assurance requirements will be provided in the test plan and QAPP.

Table 4 Process Liquids and Solids Sampling and Analysis Requirements

Sample Location	Sample Frequency	Number of Samples per Sampling Event	Analytical Laboratory	Analyses/Data Required

11. QUALITY ASSURANCE

The Buyer may periodically review Consultant's QA compliance via quality surveillances and/or management assessments. Consultant shall cooperate with these surveillances and/or assessments and shall be responsive in correcting any observed deficiencies.

Quality assurance requirements will be applied as appropriate for each item and application in accordance with the graded approach. The items procured under this contract will be classified and handled as commercial grade items in accordance with PQP XX, unless otherwise specified.

Detailed QA requirements for Consultant's conduct of the above-described work scope will be provided in the formal test plan documents. These will include Data Quality Objectives, which will establish the degree of QA/ QC (quality control) necessary to meet the data quality needs of the test objectives. Important QA/QC parameters are comparability and consistency. The ways these are achieved include:

- Using traceable standards and standard procedures for instrument calibrations
- Using standard sampling and analytical procedures/methods where possible
- Documenting necessary deviations from standard procedures/methods
- Using approved procedures for process operations and ensuring changes are rigorously approved and documented.

As a minimum, specific QA/QC requirements will be applied to:

- Fabrication and procurement of new equipment items designed by the Buyer.
- Process sampling and analysis, including packaging, shipping, and chain-of-custody for those samples to be analyzed
- Traceable standards for calibration of XXX
- Calibration of process instrumentation

12. ENVIRONMENT, SAFETY, AND HEALTH

Consultant shall conduct all activities in compliance with applicable federal, state, and local laws and regulations and in such a way as to protect the safety of workers, the public, and the environment. Federal, state, and local laws and regulations in effect at the Consultant's location shall be controlling for such functions as industrial safety, industrial hygiene, hazardous waste handling and disposal, and environmental emissions.

Safety shall be a core value in all activities conducted and shall take precedence over cost and schedule considerations. Consultant shall participate in safety meetings, discussions, and other activities conducted by the Buyer's environmental lead.

Consultant shall submit a copy of its safety plan and/or manual and a copy of its environmental documentation (e.g., hazardous waste permit, air permit, environmental program, as applicable etc.), if requested or not already submitted.

Consultant shall make available the safety training records for all personnel who will be assigned to this project. Additionally, the Buyer will conduct safety inspection/walk downs of the Consultant's facilities at times chosen by the Buyer. Consultant shall be responsive in correcting any observed deficiencies.

13. CONDUCT OF OPERATIONS

To serve the interests of safety and test integrity, Consultant shall conduct test operations in a controlled, disciplined manner. This shall include:

- Analysis and documentation of process and job safety hazards
- Preparation and approval of accurate and complete operating procedures, including the incorporation of controls to mitigate hazards
- Training of operating personnel on the process, equipment, operating parameters, procedures, and process safety
- The availability and use of approved operating procedures for process and equipment operations

- Chronological and complete documentation of operational activities in an appropriate logbook, especially events related to process upsets or safety related conditions
- Clear, concise communication among individuals involved in operational activities, especially between supervisors and operators/technicians.
- Distracting activities shall not be conducted in the operating area. This includes horseplay, television, loud music, and literature not related to process operations.

14. REPORT DELIVERABLES

14.1 Hardcopy Final Report

Consultant shall prepare a final report package covering the test activities. Consultant's final report shall contain manually generated hardcopy records, including data sheets, calibration sheets, functional test records, loop check sheets, V&V reports, and operating logs.

14.2 Project Records DVD

Consultant shall prepare a project records DVD with the table of contents shown in Table 5. Each section will have an "X" in the ART column as it is added to the compilation. Additions / modifications to this table of contents shall be made as necessary to adequately capture all of the project records.

Table 5 Example Table of Contents for Project Records DVD

DOCUMENT TITLE	ART	File
CONTRACTS		
Proposals		
Contracts		
Scopes of Work		
Work Authorizations and/or Modifications		
ACCOUNTING		
Latest Budget Updates		
Billing Rates by Person		
Summary of Invoices		
PROCUREMENT		
List of PO's, Receipt Inspections		
PO Log		
PO's, Receipts, Invoices		
SAFETY		
Safety statistics (FA, recordables, LTA, etc)		
QUALITY		
Reading Documents – Signature Pages		

Training Documents – Signature Pages		
Non-conformance files		
Equipment Calibrations		
Chemical Certificates of Analysis		
Leak Checks		
Functional Checks		
Loop Checks		
Logic Checks		
Welder qualification records - Consultant		
Vendor qualification records - Supplier, etc.		

OI 1.1 - Verified P&ID		
DATA/SHEETS		
Communication Sheets		
Digital Data from Daily Operations		
Daily Reports / Graphs		
Logbook, Data Sheets, and Calibration Sheets		
Operational Data Sheets		
Product Data Sheets		
Sample Logs		
Measurements, other data on Metal Coupons		
PHOTOS		
Photos - Equipment, Samples, etc		
ANALYTICAL		
Analyses Sheets		
Summary Analytical Results		
EQUIPMENT		
Index of Manuals		
INVENTORY		
Latest Inventory		
Records of Return		
Disposal Documents (P.O.C.)		
SAMPLES		
Inventories		
COC for Delivery		
Disposal Documents (P.O..C.)		
SYSTEM DESCRIPTION		
Instrument Tag List		
LabView VI (or code used) Code		

LOGS		
Summary of Operations		
Operator Logbook Entries		
I&C Operator Logbook Entries		
Shift Change Brief Logbook		
Client (specify) Logbook Entries		
LISTS		
Master Instrumentation List		
TESTS		
Nozzle Test Data and Videos		
Auger/Grinder Test Sheets and graphs		
PROCEDURES		
Operator Aids		
OI 1.1 - Preparation for Startup - Completed		
OI 1.1 - Checked Instrument List		
OI 1.1 - Verified ESS List		

Appendix B

Test Report Format

Appendix B - Test Report Format

The following common report format will be adapted as warranted to each element of the Hot Duct Test Plan in order to present the requirements for each required physical test. This draft version of the test plan only contains part of the information required to perform each required physical test. This format and all information described herein will be rolled into each specific test element section in this report to adequately describe all requirements for each physical test. A description of the contents of each section is provided following the list. Note that some of this information has been included in the Test Element Section for some of the outlined tests.

- Test Identifier
- Purpose and Scope
  - Features or aspects to be tested
  - Features or issues not to be tested (excluded elements)
  - System Description
- Test Approach, Assumptions & Input Data
  - Test Procedure
- Suspension criteria, resumption requirements and contingencies
- Resource needs and rationale
- Schedule
- Acceptance/Rejection criteria
- Approval of Certifications and Assumptions
  - Properties/Criteria, References
- Roles and Responsibilities
- Limiting Conditions of Operation
- Test Results and Result Summary
- Conclusions
  - Path Forward
- Data (Appendix)

1. TEST IDENTIFIER

Each test element will have a unique name and number, and all related documentation will be so marked.

1.1 Purpose and Scope

The purpose and scope of the test or simulation will be provided that describes the reasons, intentions, objectives and functions to be tested. The application of loads and the range of variables to which the test item shall be subjected shall be indicated. The particular feature, property or characteristic which is the focus of the test will be identified. All necessary features and aspects of the test or simulation shall be designated. Features, components or influences that are to be excluded or bypassed (if any) shall be stated.

2. TEST APPROACH, ASSUMPTIONS AND INPUT DATA

A description of the test approach that outlines the strategies involved in the test or simulation will be provided that includes everything that will be part of the test, and how the objectives are to be realized. This section of the test report describes the overall approach to the test plan element, the goals, activities, how it will be organized and outlines the tester's needs that must be met in order to properly carry out the test. In analytical simulations the methods used to perform the analysis and specifics of the modeling program used will be clearly stated along with boundary conditions, physical properties under anticipated conditions, applied loads, sources, and references. During physical testing, the instrument accuracy, instrument deadload and data quality used to indicate test conditions shall be specified. Such inputs and readings shall be of a suitable quality level for the performance of the particular role intended by the test objectives. Approval of the test approach by the test director is required

Assumptions used shall be stated and unverified assumptions shall be listed that must be closed or resolved at a later point in the development task. Calculations will be accompanied by a standard Calculation Disclosure Statement (sample included). When a physical simulation or prototype test is involved, all aspects of the test article and the expected outcome shall be described. The approach plan shall also include parameters and details of the external factors that must be present, data to be acquisitioned, necessary instruments, monitors and calibrations, control systems, limiting devices, safety systems, and quality assurance provisions. Certifications that are necessary prior to performance of any physical tests shall be stated. Presence of compliant, pre-accepted, manufacturing certifications shall be confirmed prior to initiation of any physical tests.

2.1 Suspension Criteria, Resumption Requirements and Contingencies

In physical test cases, prior to test initiation, conditions that constitute cause for the test to be halted, aborted or suspended shall be noted. Safeguards shall be provided and described that ensure personnel are not at risk prior to, during, or following the test, and that test facilities, equipment or the test specimen is not damaged as a result of the test (If the particular element involves destructive testing, the expected outcome shall be accounted for). Anomalies or events that occur during the test that have not been anticipated prior to test initiation can also occur. Plans to confront any contingencies shall be prepared for in advance and described. This aspect is especially important where there is a potential for risk to personnel or test equipment. Resumption requirements shall also be stated that describe the conditions that are required to restart a suspended test. Aspects of this plan shall be reviewed by test personnel during test preparation and prior to test initiation.

2.2 Resource Needs

A detailed description of the necessary resources on the part of personnel, equipment, instruments, facilities, consumables and provisions shall be provided. The qualifications or level of training of personnel involved in the test or provision of test equipment must be stated, and how they will take part in the test must be described. Where quantifiable measurements are involved, it shall be specified in detail how the testing will be accomplished, who will perform the tests, where the test will be conducted, what will be tested and what facilities and testing instruments will be required. Additionally, the utilization of resources and the duration will be estimated and provided. Who will be obtaining the measurement and under what conditions, how the measurement will be obtained, and the quality level of the data will also be specified. Furthermore, how the test will be controlled, the range over which the test is expected to occur, the data needed to be obtained and the necessary accuracy will be specified. Where pertinent, safety aspects of the activity will be described. Typically, for simulations, resources will be limited to the software and computer hardware used. Rationale for the selections made in the test plan will be presented.

3. SCHEDULE

An estimated schedule will be presented in outline form that indicates when and where the test will be performed, what external factors, personnel or entities must be present, and provides milestones and a framework suitable for making logistical arrangements that must be prepared for in advance. Resource needs must be identified in such a way that ensures their provision at the test location in a timely manner.

4. ACCEPTANCE/REJECTION CRITERIA

Acceptance/Rejection criteria for the test shall be provided in advance of the test or analytical simulation. The criteria that signifies acceptance of the article shall be inclusive of all aspects that must simultaneously be achieved under the conditions stated. Rejections occur when one or more particular aspect/s do not meet pass-fail criteria under the test conditions. Criteria include the quality standards that must be met by the data acquired during the test, or by the software utilized.

5. APPROVAL OF CERTIFICATIONS AND ASSUMPTIONS

Testing shall take place only with approved test apparatus and test articles. Necessary certifications shall accompany the acceptance of material used during the test. Certification must be performed by qualified personnel, and quality assurance and/or inspection data shall be provided using certified equipment operated by certified inspectors. Approval of Limiting Conditions of Operation (LCO) by the test director including certification data shall take place prior to test performance.

6. ROLES AND RESPONSIBILITIES

A list of the specific roles and responsibilities that will be required on the part of the test participants, material or technology providers will be supplied for each test element. Participants shall have completed necessary training, have familiarity with test procedures, safety precautions and/or quality provisions, and shall be suitably qualified in advance of participation.

7. LIMITING CONDITIONS OF OPERATION

Limiting conditions of operation (LCO) of test equipment shall include personnel that must be present during the test, including their roles prior, during and following the test, and shall include certified operators, control operators, safety and engineering personnel, data gatherers, observers, representatives and/or witnesses.

8. TEST RESULTS

Test results shall be acquired and documented during the performance of the test, and/or immediately following the test prior to influence from external factors outside the conditions of the test environment. Use of 'lab notebooks' or temporary data is acceptable, however in short order, while test conditions are still 'fresh' in the minds of the participants, that raw data will be translated into permanent format suitable for incorporation in the test results of the element test report. All relevant test data, environment and load conditions as well as the dated signature of the data taker is necessary to ensure data quality. Computer printouts and digital analytical data from measurements made from instruments likewise shall be simplified and reduced to contain information pertinent to the test and/or calibration procedure. How the data is used to formulate and describe the actual test results shall be clearly shown in a manner that other individuals, familiar with the technical subject, can decipher and easily follow. Approval of the test results by the test director is required.

Conversions and data reduction calculations shall be checked and the engineering units of all numerical quantities shall be shown. Once test data is acquired it cannot be changed, although test results can change over several iterations of the test (i.e. a preliminary test does not necessarily indicate the final result). Follow-on testing shall be indicated by a unique test identifier (i.e. –dash number). Data from suspended tests may or may not be useful. Best practice would be not to discard such data until such time that its need is overcome by events that provide useful data along the lines of the intended test goals. A spreadsheet format workbook file shall be provided for each test plan element containing test data and data reductions. Comments and labels contained in the test result data describing how the data is consolidated shall accompany the data tables. A summarizing statement shall be supplied describing the test record, the quality of the test and data gathered. Any unexpected results or external influences that may alter the quality of the data shall also be included.

When the test element is completed, the result summary provides a brief description of the test or simulation and the results. The result summary is intended to be of use toward making conclusions about the test, the results, the outcome and the path forward.

8.1 Conclusions

An element test report will be issued comparing the apparent result with the intended result, and the performance of the test article with respect to the design goals of the component or system. Conclusions may indicate acceptability unacceptability or undetermined acceptance of the test article, component or assembly. In all cases successful execution of the intended test procedure must take place in order to provide real and authentic conclusions. Review of the test conclusions by the test director and other responsible individuals is necessary. The degree to which test objectives are met should be stated and quantified to make clear the path to proceed. The test director shall indicate that the test execution was determined to be successful.

Depending on the test results and conclusions, outcomes indicating the path forward will become apparent as the test plan is filled out (i.e. as individual test plan elements are completed). Important goals, for example, are go/no-go material selections, or what worked and what did not. Such information should be included in path forward recommendations. The path forward section of the test report should include recommendations based on the success or failure of the system or component to meet the intended objectives. A successful outcome to a successful test should clear the way to proceed to the next test plan element, however it that is not the case, and other aspects need to be made clear before proceeding or making a decision, then that too should be indicated.

Peer review of the test findings and recommendations is required. The entire test plan element data package should be made available for use by reviewers. It is important that the report be complete, correct, and consistent with the goals of the overall test plan.

9. DATA APPENDIX

This is the repository of all important test information that is not contained within the body of the element test report. Should it become necessary either as a part of organizational review, review by an external or regulatory body, or as a part of some future review process, a complete file of all test data relevant to the test plan element will be provided with the report in an appendix. The appendix is to be organized with a table of contents and page count. All forms of references may be included in the appendix including drawings, sketches, pictures, interim results, preliminary revisions, hand calculations, vendor data, calibration records, raw data from tests and lab notebooks and dimensional or NDE shop inspection results. All data should be labeled for later understanding by persons that did not witness or take part in the test. Each sheet of all data records will likewise be labeled with the test plan identifier.

Appendix C

Example Test Report Form

Test Plan Element

Hot Duct Test Report

Release Version xx.xx.xx (Draft)

Subject to comment and approval

Document

Test Identifier		**Version:**	0
Filename			

	Project Role	**Name**	**Department**	**Signature**	**Date**
Creation	Test Director Test Designer				
Review & Approval	QA				

Changes

Version	**Name**	**Reason for Change**	**Date**
00		First Version	

Test Report Table of Contents

Form Ownership: (P.O.C.) dave.carroccia@wgint.com

Appendix C – Example Test Report Form

1.0 GENERAL

- An instruction guide accompanies this form.
- Bulleted Guidance information of a general nature accompanies most sections (below).
- In some cases (for this sample) example entries are provided. These will generally be deleted as the report form is filled out.
- Sections may be added or deleted as necessary.
- Contact Form Owner (listed on table of contents page) in case of omissions / inadequacies.

1.1 Purpose

This document describes the reasons, intentions objectives and functions of the test plan element. It includes all the information necessary to plan and control the test effort for the subject system

1.2 Scope

This Test Plan Report relates to the following system (i.e. object under test):

System Name	Hot duct
System to be Tested:	
Test Plan Version	xx.xx.xx
Test Run	3

The tested version is labeled **xxxx**.

- Applicable to all tests and simulations
- Provide Calculation Disclosure for all calculations and simulations

1.3 System Description

- Describe completely the system to be tested

2.0 TEST APPROACH

- Certifications that are necessary prior to performance of any physical tests shall be stated.
- Presence of compliant, pre-accepted, manufacturing certifications shall be confirmed prior to initiation of any physical tests.
- Approval by the test director of necessary compliance requirements and assumptions is required.

2.1 Assumptions

Assumptions	Description
	-
	-

Unverified Assumptions	Description
	-
	-

- All unverified assumptions shall be verified at a later date.

2.2 Input Data and Boundary Conditions

2.3 Test Procedure

2.4 Test Strategy

- At least 100% of failed test cases will be retested.
- In addition to the system validation, localized GUI masks are validated in an individual process.

2.5 Test Environment

- Provisions and preconditions have to be established in order to successfully and reproducibly conduct technical validation of the test. Such information shall be provided along with calibration data, instrument accuracy and traceability requirements.
- A schematic drawing or piping and instrument diagram shall be attached as required to indicate instruments, order of connections, operating conditions and tag numbers. Similarly, in a control and/or data acquisition system a wiring diagram, network plan and instrument I/O list shall be provided.

2.6 Metrology, Data Acquisition, Control and Instrumentation Systems

Test systems	**Operating System and instrument data**	**Control Variable or Output**	
Operation and Control System	Windows, Lab view, Platform, Accessories...		
Data Acquisition system			

Record of System Test Computer Equipment

Instrument	Instrument Address	Characteristic or output)
Strain gage, resistance temperature device, displacement probe	Tag Number	Ohms per Volt or mV per mm

Record of System Test Instrumentation

I/O	IP-Address	Type, Version/Configuration

Record of test data I/O

2.7 Interfaces

Support systems and Utility requirements	**Specification**	**Connection / Interface**	
		Test Article	Test System
Cooling water	45°F 20 gpm	Per drawing	
Signal Generator	4-20 mA, Thermocouple simulator	Provide test signal to instrument prior to test and indicate reading	

Record of System Support Requirements and interfacing equipment

- Interfaces with other systems or test equipment (if applicable)

3.0 TEST CRITERIA

- Applies to Physical Tests only.
- The criticality of any test case shall be rated and described. Critical tests shall be documented and approved by test director.
- Indicate requirements for instrument coverage and accuracy needed. It is generally necessary for there to be a minimum of one test case per critical specification.

3.1 Test Suspension Criteria

Item	Cause for Suspension	Remarks
Test Article pressure gage	- Leakage of Fitting	Isolate gage and inspect pressure decay during pressure testing
Volt Meter	- Loss of Voltage	Contuniuity fault or power outage
	-	

3.2 Test Resumption Criteria

Item	Cause for Resumption	Remarks
Test Article pressure gage	- With test gage isolated, pressure decay meets criteria	Inspect during pressure testing
Volt Meter	- Normal voltage indication	Voltage indicator display normal during application of test signal
	-	

3.3 Contingencies and Safety Issues

Item	Cause for Resumption	Remarks
Test Article pressure gage	- No Leakage at Fitting	Inspect during pressure testing

Item	Cause for Resumption	Remarks
Volt Meter	- Normal voltage indication	Voltage indicator display normal during application of test signal
	-	

4.0 RESOURCE NEEDS

- Applicable to physical tests and simulations

Test Parameter to be measured	Requirements to achieve
	- Acceptable Error band
	- Simulation software and Version

4.1 Resource Requirements

4.2 Rationale

4.3 Measuring equipment

- Applies to physical tests only
- Pressure Instrument (list, range, location, working fluid, copatibilities, calibration)
- Temperature Instrument, volt meter, dimensional measuring device...

5.0 TEST SCHEDULE

- Provide an estimated schedule in outline form that indicates when and where the test will be performed, what external factors, personnel or entities must be present, and provides milestones and a framework suitable for making logistical arrangements that must be prepared for in advance.

6.0 ACCEPTION / REJECTION CRITERIA

- Applicable to physical tests and simulations
- No critical defects are tolerated.
- At least 100% of failed cases will be retested
- Go/No-go tests

Test Item	Acceptance Criteria	Remark

Test Item	Rejection Criteria	Remark

7.0 ROLES AND RESPONSIBILITIES

Role	Responsibility	Persons (and P.O.C)
Test Director	- Coordination of design verification & validation - Management of resources for validation - Approval of Test Approach and Conclusions including Assumptions, Acceptability of results, Limiting Conditions of Operation and Certifications,	
Test Designer	- Definition of test cases - Supervision of the testing and validation activities	
Group of Testers	- Execution of test cases - Reporting of test results	

- Indicate, in advance, necessary personnel resource requirements and points of contact.
- Include permissions and approvals or arrangements that must be gained or made in advance including personnel, equipment, instruments, facilities, consumables and provisions

7.1 Limiting Conditions of operation

Limiting Conditions of Operation	Indication	Action required)

Record of Limiting Conditions of Operation

- LCO includes personnel that must be present during test
- Approval of Limiting Conditions of Operation by the Test Director is necessary

8.0 TEST RESULTS

- Approval of acceptability of results by the Test Director is necessary

8.1 Results Summary

- Applicable to physical tests and simulations
- The result summary provides a brief description of the test or simulation and the results.
- The result summary is intended to be of use toward making conclusions about the test

8.2 Test Results

- Applicable to physical tests and simulations
- Provide Test Data

9.0 CONCLUSIONS

- Applicable to physical tests and simulations
- Review by Test Director and associated SME's required

10.0 PATH FORWARD

- Applicable to physical tests and simulations
- See Instruction document

11.0 DATA APPENDIX

- Provide Page Count for each data set
- All test data shall be marked with Test Plan Identifier
- All data should be labeled for later understanding by persons that did not witness or take part in the test

Data Appendix Table of Contents			
Ref.	**Title**	**Test Identifier**	**Version**
1	Other related Test Plan Elements		0
2	Other References		

Appendix D

Sole Source Justification Form

Appendix D – Sole Source Justification Form

REFERENCE NUMBER (RFQ/RFP) XXXXX PO-00Y		ESTIMATED VALUE $XXXXX.00	
MASTER DESCRIPTION OBTAIN SERVICES FROM XXXX FOR YYYYYYY			
THIS IS TO REQUEST THAT THREE COMPETITIVE BIDS NOT BE SOLICITED FOR THE ABOVE REFERENCED GOODS OR SERVICES ACQUISITION FOR THE FOLLOWING REASON: [ ] Sole source item or service as designated by the requisitioner [ ] Purchasing strategy is other than competition (single source use of key Supplier Agreements per Procurement Plan) [ ] Lack of three acceptable sources of supply [ ] Other (explain):		[ ] Sole Source item or service as directed by the client [ ] Emergency requirement, time not permitting three bids	
REQUESTER (Signature)		TITLE TESTING ENGINEER	DATE
☐ REQUEST APPROVED COMMENTS:		☐ REQUEST DENIED	
PROCUREMENT		TITLE	DATE
PROJECT MANAGEMENT		TITLE	DATE
CLIENT		TITLE	DATE

APPENDIX B
TEST PLAN FOR HIGH-TEMPERATURE VALVES (SSC-12)

Revision: Final

ENGINEERING SERVICES FOR THE NEXT GENERATION NUCLEAR PLANT (NGNP) WITH HYDROGEN PRODUCTION

TEST PLAN FOR

High Temperature Valves

Prepared by URS – Washington Division
For General Atomics

PROJECT EXECUTION
Efficiency *with* Certainty

URS
Washington Division

URS Washington Division

Study Report:

Test Plan for

High Temperature Valves for NGNP

and

Technical Readiness Level Ratings Sheets

Date: 12/03/08

Preparer: .... David T. Carroccia

Signature of Preparer .... Dave Carroccia

Reviewer: .... **William McTigue**

Title: .... **Consulting Engineer**

Signature of Reviewer ....

Approved .... **HanKwon Choi**

Title .... **Project Director**

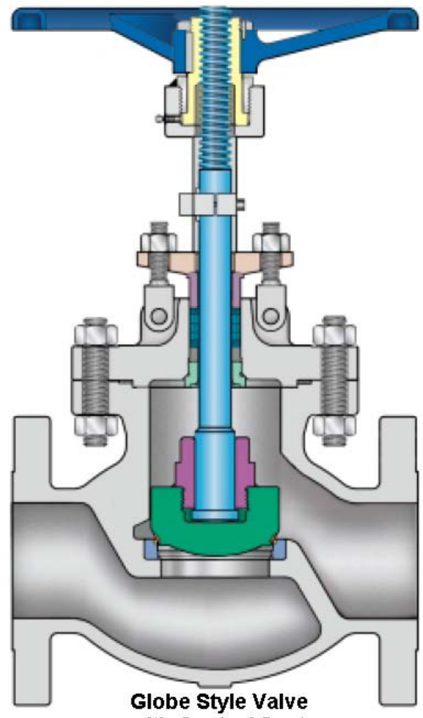
Signature of Approver ....

Project Number: 29105-3000

HIGH TEMPERATURE VALVE TEST PLAN TABLE OF CONTENTS

LIST OF ACRONYMS

DDN Design Data Need
FSV Fort St. Vrain
HTIV High-Temperature Isolation Valve
HTGR High Temperature Gas-Cooled Reactor
HTTR [Japanese] High Temperature Engineering Test Reactor
HTS Heat Transport system
IHX Intermediate Heat Exchanger
JAEA Japan Atomic Energy Agency
LWR Light-Water Reactor
MHTGR Modular High Temperature Gas-Cooled Reactor
PCS Power Conversion System
PHTS Primary Heat Transport System
SSC Structures Systems and Components
SCS Shutdown Cooling System
SHTS Secondary Heat Transport System
TDP Technical Data Package
TRL Technical Readiness Level



Globe Style Valve
with Conical Seat

1.0 HIGH TEMPERATURE VALVE TEST PLAN

1.1 Introduction

The high temperature valve test plan specifies the scope, approach, methodology, goals, resources, and schedule of each step of the Technology Development Plan to drive the Technological Readiness Level (TRL) of the high temperature valves to be used at NGNP from a TRL of three (TRL-3) to an eight (TRL-8). This last level (TRL-8) is assumed to involve dynamic testing of full size prototype valves at the Component Test Facility (CTF). The individual activities that are to be performed to fully complete the tasks for TRL advancement are referred to as test plan elements. The individual activities to fully complete the technology development and testing tasks are referred to as test plan elements. The following aspects of each test plan element are addressed in the test plan.

- scope of each test or simulation
- associated accept/reject criteria
- risks and contingencies
- test deliverables (including results)
- responsibility for accomplishment of the element goals
- quality requirements

A guide to preparation of the test report is provided in Section 9, and a blank test report format document is included as an example.

This test plan represents the overall approach to demonstrating the capability of high temperature valves, including helium isolation valves and vessel system relief valves, to meet specified performance requirements over the design plant lifetime. The test plan can be revised as execution of the technology development plan progresses. An outline of all the test plan elements, referenced to the technology readiness level, is provided following the introduction.

Although isolation valves and relief valves have been used in nuclear power plant environments for half a century, the application environment for the NGNP MHTGR is special because of the 800°C - 950°C dry helium working fluid the valves will be exposed to during the 60 year working lifetime. The valves must work with an extraordinary degree of reliability, and serve as a part of the primary coolant boundary (including body, stem, bonnet, packings and seals) without fugitive emissions. Additionally, the relief valve provides overpressure protection for the reactor vessel and power conversion unit and hence is classified as a safety related component requiring qualification.

1.2 Test Plan Summary

Table 1 below lists the TRL tasks to achieve a TRL 8. These tasks are cross referenced with the applicable section in this test plan. Expected durations are provided.

Table 1: TRL Task Summary and Estimated Duration

TRL	Task Description	Section	Estimated Duration
4	Safety Class Determination	2.1	1 year
	Relevant Standards and Codes Applicability	2.1	
	Establish Conditions of Service	2.2	
	Material Properties	2.3	
	Coupon Tests	2.4	
	Fasteners	2.5	
	High Temperature Weld Formation	2.6	
	Helium Permeability	2.7	
	Gaskets, packing materials, Seals	2.8	
5	Material Selection	3.1	1 year
	3d Modeling and Analytical Test Simulations	3.2	
	Acoustic and Flow Induced Vibrations	3.3	
	Differential Thermal Expansion Analysis	3.4	
	FEA Stress Analysis	3.5	
	Endurance Limit Analysis	3.6	
	Creep Analysis	3.7	
	ALARA Analysis	3.8	
	Routine and Non Routine Maintenance requirements	3.9	
	RAMI Analysis	3.10	
6	Physical Test Preparation	4.1	1 year
	Test Apparatus	4.2	
	Determine Methods of conducting Valve Inspections	4.3	
7	Integrated Experimental Scale Model Test	5.1	1 year
	FEA Simulation Optimization	5.2	
	CFD Simulation Optimization	5.3	
	Final Leak Detection Validation	5.4	
	In-Service Maintenance and Inspection Techniques Validation	5.5	
8	Integrated CTF Testing	6.1	2 years
	Maintenance, In Service Test and Inspection Techniques Validation	6.2	
	Stress Analysis Validation	6.3	
	Temperature and Flow Analysis Validation	6.4	

1.3 Overview

Purpose and scope of the Test Plan: The Test Plan task begins with identifying the high temperature valves, their location within the plant, the intended service, the design bases, and determining the Conditions of Service under normal, upset, emergency and faulted conditions. The properties of candidate materials for the valve body, seating materials, packing and other parts exposed to the working fluid and high temperatures are compiled over the range of operating conditions to determine compatibility with their environment as part of advancement to TRL - 4. Although preliminary models will be constructed in level 4 to determine operating stress levels and component temperatures, high fidelity 3d models are then constructed in Level 5 representing the design, configuration and geometry of the proposed valve. A number of analyses will be performed which simulate the various aspects of operation, allowing the behavior of the design in response to loads to be visualized and quantified. Detailed analytical simulations will be performed in at least seven areas, and then results will be combined to evaluate the response of the system to all loads applied simultaneously under all the various operating cases. Areas of detailed study represented by FEA models and the corresponding results obtained are shown in Table 2 (in section 3).

This test plan represents the overall approach to demonstrating the capability of selected valves to meet specified performance requirements over its design lifetime. The test plan can be revised or terminated as execution of the technology development plan progresses. QA requirements, which apply to all test plan elements, are listed separately.

Basic models of candidate valve configurations and materials should be generated initially, 'up front' (i.e. during advancement to TRL - 4) in the execution of the test plan; allowing the behavior of a number of alternative material combinations to be examined under various load cases. These models will also be utilized to determine the necessary load levels that materials used in tests have to be exposed to. Down-selection of valve types and materials based on merit can then take place and then optimization of the design can occur. Detailed 3d FEA models will be generated in TRL level 5 and refined for use as the final model providing analytical justification of the final design. The final models will feature all the aspects of the final designs of each of the valves proposed to be installed in the system. This analytical model will then be validated based on test results obtained from prototype testing of physical scale models (in TRL level 6) and full scale optimized prototypes tested in an integrated manner at the CTF (if it is determined that they are necessary) in TRL Level 8. Between TRL - 6 and - 8, final design optimization takes place together with design verification of the fully integrated valve consisting of all its components. Through these models and associated tests, a number of necessary assessments of the design can be performed including ASME Code compliance, certification of compliance to regulatory requirements, Creep and Endurance Limits, leak tightness, ALARA, and response to Design Basis Accidents (DBAs) and hazards.

1.4 Background

This test plan applies to the following high temperature valve applications for NGNP:

<u>High Temperature Isolation Valve (HTIV)</u>
This test plan assumes the selected NGNP design uses a large Intermediate Heat Exchanger (IHX) with helium as the secondary fluid for the Power Conversion System (PCS). This assumption requires the development of a large, high temperature helium isolation valve. As stated in [Labar 2008], there are no currently available large-size He isolation valves suitable for this application. There are, however, suitable isolation valves available for steam-water secondary systems.

The secondary heat transport loop between the IHX and hydrogen production plants will likely have three isolation valves on each hot and cold leg. Two of the valves would be located near the IHX and one or more valves would be located hear the process heat exchangers. HTIV design requirements are influenced by the IHX design, e.g., isolation valves may be required to equilibrate IHX pressures during design basis events to limit

IHX creep fatigue damage caused by occasional high pressure differentials at temperature [Labar 2008].

For the Secondary Heat Transport System, the significant Design Data Needs (DDNs) are associated with the HTIVs. Additional DDNS are associated with internal insulation. Testing of a HTIV at the Japan Atomic Energy Association (JAEA) High Temperature Engineering Test Reactor (HTTR), was performed to address technical issues involving materials, summarized in [Hanson 2007]. (See Figure 1) An angle valve with an inner thermal insulator was selected. A new valve seat material, with sufficient hardness and wear resistance over 900^{o}C, was developed based on the Stellite alloy that is used for valves at around 500^{o}C. A component test of the valve seat indicates that a flat type valve seat can maintain the face roughness of the valve seat within allowable limits during operation. A 1/2 scale model of HTIV was fabricated to confirm seal performance and structural integrity. The He leak rate was confirmed to be less than the target value. HTTR operating experience and test data are considerations for NGNP HTIV test plan development.

<u>Vessel System Relief Valve</u>
Technology development needs for the primary vessel relief valve design is affected by the PCS design. Relief valve lifts would result from large water ingress events [Hanson 2007]. Selection of a PCS with a steam generator on the PHTS increases the potential for relief valve lifts due to water ingress events. The discharged fluid characteristics are likewise affected by the PCS selection. The potential need for relief valve filtration to meet radiological dose limits is identified as a DDN in [HTGR 86025]. Relief valve design and ASME Code considerations are further described in Section 2.1.

<u>Circulator Shutoff Valves</u>
The Primary Heat Transport System (PHTS) and Shutdown Cooling System (SCS) shutoff valves are addressed as part of the helium circulator test plan.

2.0 TASKS TO ACHIEVE TRL-4

TRL-3 is in essence a proof of concept. Volumes of related industrial experience exists for high temperature valves in similar applications. Therefore TRL-3 (proof of concept testing) for the high temperature valves is not required. However, bench scale <u>material</u> tests are likely to be required. For the valves, laboratory scale tests will consist of various accelerated environmental exposure tests and material tests where a gap exists between available material manufacturers' data and design basis conditions relevant to the valve's application. Additionally, during this phase of technology development, conditions of service must be documented, safety class determinations must be established, code and standard applicability issues resolved, and initial material selections must be made so the valve(s) design can be advanced to TRL-4. Critical design characteristics must be established at this level, to determine the acceptable rate of valve leakage, required response times, pressure drop at rated flow, accident basis pressures and temperatures, and to adopt the valve configuration and actuator type. During this level of the TDP, relevant data from other facilities will be researched and made available to the engineering design files across the spectrum of valve applications. Relevant applications include the FSV HTGR, HTTR and NGNP.

2.1 Establish Safety Class and Codes and Standards Applicability

Safety classification of MHTGR structures, systems and components (SSCs) is identified as a licensability issue in [PSER, 1996], in part because MHTGR SSCs were considered safety related only if they were required for accident dose consequence mitigation. Current NRC licensing policy for advanced reactors, e.g., as given in NUREG-1860, suggests that safety classification criteria for the NGNP will be similar to that of current light-water reactors (LWRs). Manufacturing standards for valves described in ASME QME-1-2007: "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants" will be utilized. The in-service standard for nuclear plant valves is ASME OM Code-2004 (soon to be 2008): "Code for Operation and Maintenance of Nuclear Power Plants". This code has a section on the in-service testing of nuclear valves. In-service inspection requirements will utilized this standard, see also sections 3.9 and 6.2. The determination of performance characteristics for HTIVs and Relief valves shall be made, and the applicability of performance models such as EPRI PPM (Performance Prediction Methodology) shall also be determined. The effect of all documented failure modes and how they relate to the particular valve designs and applications shall be determined.

Relief valves providing overpressure protection for the vessel system are safety-related and subject to ASME Section III Class 1 requirements. ASME code considerations particular to the NGNP relief valve design include:

- Characterization of discharge flow with respect to potential moisture and particulate content, which is affected by Power Conversion Unit (PCS) selection (i.e., steam cycle or gas turbine).
- Effect of filtration on backpressure should be minimized such that it does not adversely affect pressure relief capability.
- If a rupture disk is used in conjunction with a relief valve, then the rupture disk may only be installed downstream of the valve.
- Provisions for effective removal of moisture and particulates from the valve seating surfaces must be included in the discharge line.

High Temperature Isolation Valves: The HTIV function of secondary loop isolation is identified as an NGNP protection function. HTIVs may also be required to perform a reactor building isolation function e.g, on high radiation signal. [Labar 2008]. These functions imply a nuclear safety classification is applicable to the HTIVs.

The specific HTIV nuclear safety classification is affected by the overall NGNP HTS design approach. Designing the secondary system to satisfy the requirements of a Class 1 primary pressure boundary is expected to cause excessive plant costs. HTIVs will be required to create a boundary between the primary and secondary systems to avoid the need to design the secondary system to function as a Class 1 pressure boundary [Labar 2008]. Secondary system isolation valves in LWRs (i.e., feedwater isolation valves and main steam isolation valves in pressurized water reactors) are typically designed to nuclear class 2 standards, invoking ASME III subsection NC criteria. A nuclear class 1 designation (ASME III, Subsection NB) for HTIVs would impose more stringent design criteria than nuclear class 2. In either case, ASME III limitations on maximum temperatures will require ASME code changes to support HTIV design and NGNP licensing [Bolin 2008].

Determination of Safety Class: By GA

2.2 Conditions of Service

Normal, Upset, Emergency and Faulted Conditions will be examined for all the high temperature valve applications. Conditions of service, which reflect the environment/s to which the valve(s) are exposed and expected operating conditions, are established.

COS: Provided by GA

2.3 Material Properties

Obtain candidate material properties over the operating range. Evaluate Tensile strength, creep and relaxation data, high and low cycle fatigue criteria, fracture toughness, high temperature endurance limit, fabrication limitations and tolerances, thermal expansion, welds and heat affected zone material properties, coatings and surface finishes, sliding surface friction values and how they change over time following exposure to the high temperature and Helium or other working fluid in the valves' operating environment, lubrication, effects of exposure due with aging, material and fabrication cost, along with chemical and radiation resistance over the range of operating conditions including Normal, Upset, Emergency and Faulted conditions.

Thermal deformation of valve body and seat are examined together with leak tightness at temperature. Wire drawing and welding due to high temperature, and dry gas flow conditions are studied with regard to material selection. Recent studies of material behavior under exposure to high temperature, dry helium flow have shown tendency to wire draw valve seats, as well as cause "dry" welding of seat to body. Available study data will be examined here and the need for additional testing will be assessed.

The US Department of Energy entered a cooperative agreement with ASME Standards Technology, LLC (ASME ST-LLC) to update and expand appropriate materials, construction and design codes for application in future Generation IV nuclear reactor systems that operate at elevated temperatures. These studies will be referenced and utilized as applicable during material selection and evaluation.

Studies shall be performed that determine the requirements for periodic verification of valve operability under design basis conditions. These requirements are developed with consideration of the program elements described in NRC Generic Letter 96-05, Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves, dated September 18, 1996), and conform to ASME Section XI and OM Code criteria as endorsed by NRC via 10CFR50.55a Codes and Standards. Valve design, installation inspection and testing provisions shall accommodate the ability to periodically verify operability, and are influenced by material selection..

Provided by: Testing Organization (See section 8.3)

2.4 Coupon Tests

Coupon tests are an important precursor to full-scale physical testing. Testing coupons, or small sections of material, confirms that the material can withstand the rigorous requirements before investing in the expense of building a full-scale mock-up. The sub-sections below explain each coupon test.

Materials of particular concern are the valve body and seat, as they are exposed to the highest temperatures. Where body, bonnet and/or seat coupons are necessary, materials shall be cast, forged, rolled and subjected to the same manufacturing processes (i.e. including secondary operations) as will the genuine article. Coupon tests are required where gaps exist between valid and traceable manufacturers data and the anticipated operating environment. Data acquired during testing must be of the suitable quality level and contain traceability information as specified in the quality requirements below.

Provided by: Testing Organization

2.4.1 Environmental Exposure/Embrittlement

This test will involve exposing the coupon to all chemicals, atmospheric impurities and environmental factors (such as temperature and pressure) it will experience during operation. This test will be the first coupon test conducted to allow for the maximum exposure to environmental conditions. Hydrogen embrittlement

testing and halogen (Iodine) exposure data will be obtained. Material performance data collected will be used for material down-selection. An estimate of the range of exposures to environmental risks will be necessary to perform this test.

Exposure Basis: Provided by GA

2.4.2 Room Temperature Properties and Chemistry

Tests of material properties at normal and elevated temperatures that are conducted in compliance with ASTM A370 requirements will verify that the vendor data used in the simulations is accurate. All material properties used for the simulations must be verified. CMTR's (Certified Material Test Reports) will be provided by the testing organization.

A source for such tests is IMT Intermountain Testing located at 2965 S. Shoshone in Englewood CO 80110 Tel. 1-800-742-5621, P.O.C. joe@intermountaintesting.com

2.4.3 High Temperature Properties

Several coupons will be tested to confirm all the pertinent elevated temperature material properties. All material properties used for the simulations will be verified. A CFD analysis will be performed to evaluate the expected operating temperature of the valves and components to determine the relevant range of temperatures the candidate material coupons should be subjected to.

2.4.4 Elevated Temperature Tensile Strength

Several coupons will be tensile tested to determine the actual yield and ultimate tensile strength of the material, particularly the valve body material. The test values will be used to validate vendor data and provide information to fill in any gaps between the available data and expected operating temperature. This test will involve exposure to hot dry helium necessitating specialized testing apparatus. Pre and post exposure data points will be collected. A FEA analysis will be performed to evaluate the expected level of stress within the valves and components to determine the relevant range of loads the candidate material specimens should be subjected to.

2.4.5 Fracture Toughness

Several coupons will be fracture tested to determine the actual fracture toughness of the material. Both the valve and seat material fracture toughness values are useful in determining their behavior during the plant lifetime. Notch sensitivity is important to determining the resistance to strain in areas where stress concentrations exist in a cyclical loaded application. The test values will be used to validate vendor data and will be made available for detailed analysis purposes.

2.4.6 Thermal expansion

Several coupons will be heated to temperatures that match normal, upset and fault temperatures so actual thermal expansion can be measured and compared against vendor data for validation if necessary. A thermal differential expansion calculation will be generated, based on the configuration of the valve(s) and materials of construction, and the operating temperatures obtained from CFD and thermal FEA models. Areas will be identified where sensitivity exists to differential thermal expansion and this test will be targeted to verify thermal expansion characteristics in the areas of interest.

2.4.7 High and Low Cycle Fatigue, Creep Rupture and Relaxation

A number of material coupons will be subjected to simulated operating environments and analyzed for both the high and low cycle fatigue properties and the creep and relaxation properties exhibited by the candidate materials. Properties must be obtained at elevated temperatures using ASTM E-139. These values will be examined against the expected values to be encountered over plant life. An estimate of thermal and mechanical cycles must be made to determine the expected level of service. High cycle fatigue specimens are usually cycled until failure, and the stress level and accumulated cycles at failure will be compared with the expected service conditions.

2.4.8 Weld Strength

The weld strength coupon test will involve producing weld samples for destructive testing to confirm the strength characteristics of the weld and the heat affected zone. The following tests, which will involve metallurgical inspection after subjecting the specimens to high temperatures, will be conducted on the weld coupons:

- U-bend at the weld joint, with dynamically applied load
- Heat affected zone material properties and microstructure
- Creep and evidence of creep crack initiation or void formation

If joints to adjacent piping involve welds to dissimilar alloys, then dissimilar weld specimens will be tested. Weld procedure methods may have to be generated and utilized for this test.

An additional area of interest in this category is identifying applicable NDE methods. Methods that can be performed on-site will be useful for field welded valves, and shop applicable methods will be useful for factory welds. Obtaining data on the minimum flaw size detection level using these methods and comparing this with the critical crack size for dynamically stressed material at high operating temperatures will be useful for qualifying NDE methods. If deemed warranted, then additional coupons will be necessary. Specimens with 0.032" FBH (Flat Bottomed Holes) may be of use to ascertain flaw detection levels.

2.4.9 Accelerated Erosion and Corrosion

Accelerated wear and corrosion tests will be completed to verify that the selected material can indeed withstand the environmental conditions to which it will be subjected. Accelerated flow and solids loading in the gas stream will be used to complete this test. Test parameters must be specifically designed to accomplish this test due to the unique environmental requirements placed on valve seals exposed to erosive flows.

Erosion and Exposure Basis: Provided by GA

2.4.10 Stress Corrosion Cracking

Several coupons will be subjected to accelerated corrosion conditions and then stressed to projected operating stress levels to investigate whether corrosion accelerates the propagation of cracks.

Test Method: ASTM STP 1210 (Slow Strain Rate Testing for the Evaluation of Environmentally Induced Cracking) test article qualification, performance test, and validation.

2.4.11 Irradiation

Several coupons representing different components of the valve(s) will be subjected to the equivalent amount of radiation flux the actual valve materials, seat seal and packing materials are expected to endure throughout the plant's operating life. Metallurgical examination, microstructure evaluation, morphology and destructive strength testing will be completed and compared against the un-irradiated room temperature coupon performance.

Test Method: Test article qualification, performance test, and validation. Note this testing will be accomplished in partnership with the US DOE National Lab efforts to qualify NGNP materials. Facilities for irradiation simulating the high flux fields found in a reactor environment exist only at INL (ATR) and ORNL (HFIR). Facilities for post irradiation metallurgical examination exist at Argonne, (Environmentally assisted cracking of reactor materials), Idaho (Hot Fuels Examination Facility or HFEF), and Oak Ridge (Irradiated Materials Examination and Testing or IMET).

2.5 Fasteners

Any fasteners used on the valve(s) will be tested and compared against vendor data and design requirements. The measured yield strength from destructive bolt testing will be compared against vendor data for validation. If torque-tension relationships are used for field or assembly activities, those parameters will have to be determined by controlled and instrumented tests.

Test Method: Test article qualification, performance test, and validation. Charpy V-Notch Testing (for high strain rate qualification) will also be performed on fastener materials. There may be peculiar aspects of fastener geometry due to remote maintenance considerations that necessitate special fastener fabrication. See Section 3.9 and Figure 5 hereinafter.

Provided by: Testing Organization

2.6 High Temperature Weld Formation

Hot helium, with an extremely high quality (low moisture content) has been shown to cause valve body to seat “welding”. Testing will be conducted with seat, seat seal, packing and body materials exposed to flows off hot, dry helium to determine properties and potential for welding.

This testing requires longer term exposure of the coupon to the test environment. Samples of valve seat and body metal, (or ball and seat in the case of a ball valve) will be placed together under load in a configuration that approximates a valve body and plug, subjected to extended periods of exposure to flowing NGNP quality high temperature hydrogen, and inspected to see if the coupon exhibits any potential for material welding. Additional valve material selection studies may need to be conducted at this point, depending on results. If necessary the test plan will be adjusted accordingly, or alternative designs will be considered.

Test Method: Coupon material qualification, long term exposure to high temperature He flow / material (combination) validation. A special apparatus will be required to perform this test.

Provided by: Testing Organization in conjunction with Valve Suppliers

2.7 Helium Permeability

The possibility of helium escape via valve body permeability will be examined during this test. The use of cladding of valve body and other methods to reduce coolant external leakage will be studied. Previous experience with hot helium shows cast valve bodies exhibit helium permeability at high temperature and pressure. The degree of helium permeation has been significant enough in some applications to require cladding of the valve body.

Test Method: Coupon material validation; Assuming valve has a cast or forged body, fabricate duplicate castings or forgings for testing using a.) identical-alloy and casting process, or b.) identical alloy and forging process. Evaluate helium permeability by helium leak detection.

Provided by: Testing Organization in conjunction with Valve Suppliers

2.8 Gaskets, packing materials, seals, moving parts

Any gaskets, packing materials and seals used on the valve(s) will be tested and compared against vendor data and design requirements. The measured data from testing will be compared against vendor data for validation. If adjustments are involved in field servicing or assembly activities, those parameters will have to be determined by controlled and instrumented tests. Qualification and stability of moving parts involving friction surfaces, sealing materials, actuator force/torque characteristics, lubricants (including position sensors), torque transmission devices in the load path, deterioration of surfaces or lubricants due to aging and exposure to the operating environment, and other potential contributors that may have the effect of increasing the total load required to operate the valve, or reducing the power available to actuate the valve must be determined. These inter-relationships are dependent on the specific valve design, and although final designs are not available at this point in the test plan, material tests required to obtain an understanding of these contributing effects are to be gathered to the extent possible during this portion of the test plan.

Test Method: test article qualification, performance test, and validation. This aspect of the valve design will be realized by discussions with qualified potential valve suppliers such as those mentioned below in this report. Stems may require exterior cooling systems or may incorporate special features (i.e. stem extension) to isolate and insulate packing from service temperatures. See Figure 8 (Section 8). This will result in a service temperature for the packing that is different from the service temperature of the seat seal.

Provided by: Testing Organization in conjunction with Valve Suppliers

3.0 TASKS TO ACHIEVE TRL-5

Advancement to TRL-4 (previous section) involves bench scale testing to verify material properties, and will make use of <u>preliminary</u> 3d models and FEA analysis to determine stress levels and operating temperatures. As noted above, a level of component verification will also be performed in advancement to TRL-4 to determine allowable load ranges and compatibility of the materials with the operating environment.

Since the high temperature valves are components based on technology that to a large extent has been demonstrated in similar situations, analytical modeling is an acceptable method of determining a components qualification for the intended service. Advancement to TRL-5 will consist of constructing <u>detailed</u> models and performance of the simulations and analysis studies outlined in Table 2 as well as development of maintenance and periodic verification methods. Final material selections will be made by down-selecting among candidate materials. In TRL-5, the conceptual design of the MHTGR will be completed allowing establishment of a complete valve list, the locations the valves will occupy will be known, and the services that particular valves will have to perform. This will allow the configuration of the valves to be established.

Table 2: Analysis Studies for High Temperature Valves

Analysis	Analysis Method	Results	By:
Flow	CFD	Pressure Drop and Velocity profile throughout cross section of valve internals; Back Pressure, Relief Capacity, Blowdown and Solids Accumulation (Relief Valve Only)	
Heat Transfer	CFD	Conduction, Convection, Thermal radiation, Heat Transfer Coefficients, Contact resistance to Heat Transfer, effects due to internal and/or external insulation, and insulation effectiveness	
Body Structure	FEA	Stress, Strain, Deflection, Body, Seat area, Bonnet, pressure boundary	
Seat, disk Seat Seal, contact pressure	FEA	Stress, Strain, Deflection, creep threshold, Seal load, Seal on backside of renewable seat	
Response to Disturbance	FEA, CFD and non-linear analysis	Transient Response	
Supports and Connections	FEA, Hand Calc.	Leak tightness, Localized stress, Joints at ends of body, Flanged joints, actuator coupling, How Position Detection is implemented	
Thermal Expansion	FEA, Hand Calc	Response to temperature loads, elongation, differential thermal expansion of both internal and external valve parts, effectiveness of insulation and cooling or jacketing	
Acoustic Vibration	Acoustic Vibration Specialist	Response to harmonic and acoustic induced vibrations	SME
Creep	Creep Analysis Specialist	Creep effects (Non-Linear)	Becht NS
LIMIT	LIMIT Analysis Specialist	Independent check on the allowable wall thickness of body	Becht NS
Endurance	Endurance Analysis Expert	Endurance limit of body and actuator connection, reliability of moving parts	Becht NS
ALARA	Computer Code (dose calcs.), ALARA SME committee	Design optimization to reduce radiation dose to workers (including maintenance, repair service, and periodic verification of performance (relief valves)	URS-WD (Princeton) + IP Team
RAMI	Hand Calc, software	Quantification of Reliability, Availability, Maintainability and Inspectability parameters	Integrated Proj. Team

Note: in structural and thermal FEA models the results from the several models will be combined, and loads from various cases will be applied simultaneously

Responsibility for completion of TRL-5: Testing Organization unless specified otherwise

3.1 Material Selection Rationale and Valve Configuration

Valve Body and Bonnet Candidate Materials

- Alloy 800H (AT/HT) (N08810/N08811)
- Haynes 230 (N06230)
- Hastelloy X (N06002)
- ASTM A297 Grade HK

Except for the iron-based Alloy 800H, the above HT (high temperature) alloys are more costly nickel-based alloys. These alloys were identified by many investigators in the technical literature as having high creep-rupture strength and other properties to resist He gas conditions to 950C. Alloy 800H is the most industrially mature and best established of these HT alloys, used widely in the power, refinery and process industries. Alloy 800H(AT/HT) is approved by the ASME Code Sections I and VIII Div. 1 up to 900C and good creep-rupture strength up to 927-980°C. It also has useful oxidation resistance up to 1038C and is readily weldable.

A comparison of the ASME Code approvals of Alloy 800H with two other HT alloys is shown in Table SEG 1, Attachment 1. Haynes 230 is considered as a replacement of Inconel 617 so the table is limited to three HT alloys. Even for Alloy 800H, ASME Sect. III limits its use to about 430C, so that ASME code cases must be submitted and approved by the Code for 800H and Haynes 230 up to 900/950°C.

ASTM A297 HK is an austenitic iron chromium nickel and is one of the strongest heat resisting casting alloys at temperatures above 1900°F (1038°C). This material would be considered for the body and bonnet castings.

Minimum wall thicknesses must be selected for the various alloys considered, but in no case would they be less than those specified by ASME 16.34 Class 600 regardless of the size of the valve.

3.1.1 Valve Seat Candidate Materials

- Stellite cobalt-based alloys
- Stellite/Chromium carbide composites
- Ceramics
- Tungsten or Silicon Carbide

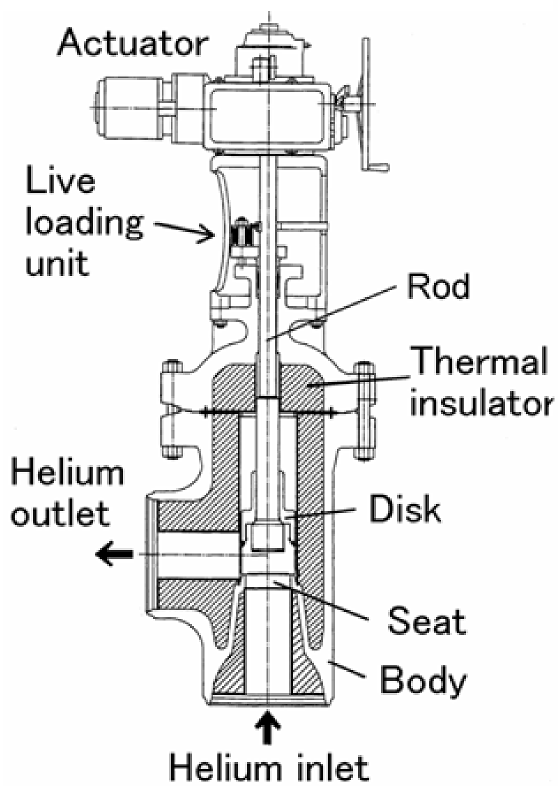
Since cobalt can form long half-life isotopes in a nuclear reactor environment, the emphasis on valve seats, plugs, or balls for ball valves should be on suitable non-Co materials such as very hard ceramic materials. Stellites have good corrosion/erosion/wear characteristics and have been widely used as valve seats, but Co is a radioactively detrimental wear product. Thus, potential ceramics such as alumina and various metal carbides will be assessed and evaluated for this high wear application.

For Ball Valve applications, a type 316SS ball and seat will be considered, which would be sprayed with chrome carbide and match-lapped to each other.

3.1.2 Design of Valve Body and Seat

Results of material testing discussed above will be input to the development of the valve overall configuration for testing. Material and operability concerns will dictate the necessity for valve internal insulation, and/or internal or external active cooling (isolation valve only).

Figures 1 and 1A below provide cut away illustrations showing design characteristics typical of high temperature valves. On the left of Figure 1is a cross section of a manually actuated high temperature globe type isolation valve with internal insulation which served as a test article for seat seal testing [Nishihara 2004 for JAEA]. Figure 1A (right) represents a globe valve with an external active cooling jacket, courtesy of Target Rock Flow Control.



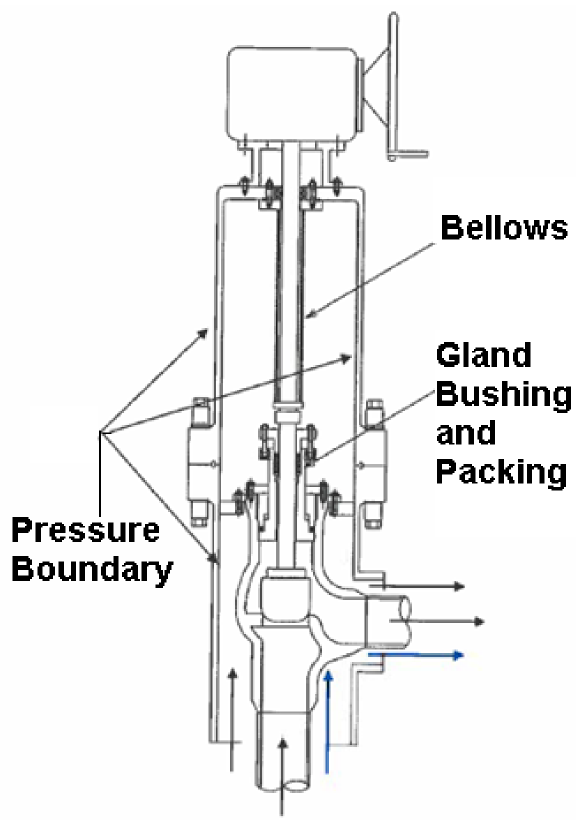
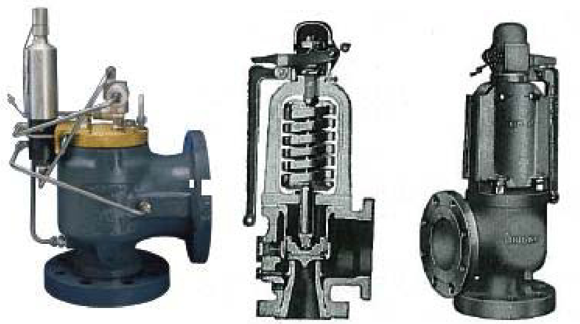


Figure 1 and 1A: High Temperature globe valves with passive and active cooling features

Other valve body designs may also be considered for isolation valve applications. See also section 3.9 below



Typically, a spring loaded globe is used for the relief valve. Indirect acting pilot operated designs are also available.

Although designs used in previous applications will be considered for the MHTGR, a spring actuated ball valve is a viable candidate for this application. Upon signal to the actuator, the normally closed valve would open to relieve vessel overpressure. In this design a signal indicating the overpressure condition would notify the operator, and automated supervisory control methods could be employed. Unlike the globe, when open, the ball valve's seat would not be subjected to the flow of discharging coolant with entrained particulate, ensuring that when closed the valve will seal completely. Opening of the reactor vessel relief valve is an extremely abnormal occurrence, and having the capability of interjecting supervisory control on such an occurrence is felt to be highly desirable. Additionally, the straight through port arrangement on a ball valve flows much more efficiently and with lower back pressure than a globe, which has, comparatively, a rather more torturous flow path. Further, controlling the speed of the actuation during opening of a ball valve to relieve pressure will eliminate a sudden opening characteristic and decrease design loads on downstream discharge piping. Finally, by exerting the ability to control the valve, the operator or control system can be completely effective in controlling vessel pressure.

The seat and seal in a ball valve would be 316 SSt which bonds well with chromium carbide. The sealing surfaces between the ball and the seat would be lapped to provide a class VI shutoff or better (depending on specified leakage requirements) and would be acceptable for service at 1800°F. The seal, packing, stem and bonnet and how such features are integrated within the overall valve design are all of importance during this portion of the study.

The above discussion is centered around two-way valves (with a single inlet and outlet). Three Way High Temperature Valves are also within the scope of this test plan (if needed).

The requirements for Equipment Qualification found in ASME QME, specifically Section QV (qualification of valves) should be considered early in the design process and performed as part of the prototype testing. Required qualification tests include:

Seismic

End load

Functional

Environmental

Sealing Capability

Also, Section QV-G Determination of Performance Characteristics, should be factored into the design and testing.

3.1.3 Dissimilar Material Considerations

Some material candidates may be eliminated based on their interaction with the other assembly materials (piping material, seat material, insulation). All potential candidate materials will be analyzed as a system for corrosion potential. This comparison will be based on available vendor and consensus code requirements. Materials combinations that show corrosion potential will be eliminated.

The valve assembly will have to be field welded to the adjoining pipes at installation. Therefore, the welding compatibility of the adjoining piping or ductwork is of importance. With the vessel or attaching piping material known, a welding compatibility analysis can be performed for each candidate material. Materials that can not be welded to the adjoining components or materials that result in deficient or inadequate weld properties will be eliminated.

3.1.4 Differential Thermal Expansion

Once the candidate materials are screened out based on material properties and dissimilar material corrosion potential, a differential thermal expansion analysis will be completed. All materials expand and contract under thermal loading to varying degrees; several factors must be investigated including:

- Stresses caused on valve connection points by interaction of the adjacent piping and insulation
- Stresses imposed on the valve internals from expansion or contraction
- Stresses from external forces encountered from operation and design basis hazards
- The effects of differential thermal expansion that could lead to thermal binding

It is not expected that differential thermal expansion will be a factor in material selection but rather an additional design challenge that will be identified early on in the design. Manufacturing tolerances can typically be tailored to allow for the desired clearances at normal operating temperatures based on the thermal expansion and performance requirements of the utilized materials, however, necessary tolerances and clearance requirements must be determined.

3.1.5 Environmental Qualification of Candidate Materials

A detailed study of all candidate materials will be conducted to determine the environmental qualifications of the candidate materials. Down selection may be possible if a material stands out as being inadequate. Environmental considerations include tolerance to elevated temperature, contaminants/impurities and radiation. This study is specific to valve application and performance requirements. Steam exposure and mixed flow concerns will also be addressed.

The study will be based on available industry testing data; if it is determined at this step that additional raw material testing data is needed, relevant testing will be conducted. Allowable properties of materials will be compared against design environmental conditions and down-selection to several target materials can be performed.

3.1.6 Erosion and Corrosion Allowances

Any material selected will need an associated erosion and corrosion allowance. This initial quantification of necessary erosion and corrosion allowances will be based on the specific material properties. It is possible, although unlikely, that any down-selection of materials will be performed knowing the required erosion and corrosion allowances. However, having this information early in the design will allow for proper initial determination of the required wall thickness for use in the FEA and CFD analyses.

The seat seal material's vulnerability to particulate entrained in the process fluid will be determined by physical testing. Some valve designs are more vulnerable in this area than others. The sealing surface on globe valves, as shown in Figure 1 and in the in-line globe valve shown in the frontispiece (adjacent to the acronym list), is directly in line with the high velocity particle streamline, and is subject to wear which may erode its ability to completely seal when closed. On the other hand, the sealing surfaces in ball valves, as shown below (See Figure 5) are completely protected in both the open and closed positions. Furthermore, both the seat and seal are easily replaced in the ball valve design shown. The back sides of the seals achieve leak tightness using a metal o-ring. In a globe valve the seal can be easily replaced, however, the seal striking surface must (usually) be refurbished by in-place refinishing operations. These are important considerations in valve type selection by application.

3.1.7 Valve Body Helium Tightness

The possibility of helium escape through the valve body by permeability will be examined during this investigation. Note a coupon test has been performed previously involving a forged or cast specimen, but this test will make use of an actual valve body. The use of cladding of the valve body and other methods to reduce coolant loss via external leakage will be studied if necessary. Previous experience with hot helium shows cast valve bodies exhibit helium permeability at high temperature and pressure. The degree of helium permeation has been significant enough in some applications to require cladding of the valve body. Available information from material suppliers may not be complete in this area, and testing at the appropriate environmental conditions may be required. This investigation may also include tritium confinement.

3.1.8 Interfaces with Adjoining Structures

3.1.8.1 Attachment Methods to Piping

Welding methods to attach valves to adjacent piping, for both relief valves and isolation valves, will be examined analytically through material studies and FEA and CFD analyses. Concerns include weld cross-section, strength, loads from external sources and flow induced vibrations, heat transfer effects and stress concentrations due to non-uniformity throughout the joint. Depending on application, external piping loads may be significant. Although the conceptual design may be complete at this point, detailed piping reactions may not be available, therefore upper limits using ASME allowable nozzle loads may be applied to the model until detail loading data becomes available.

3.1.8.2 Factory and Field Joints

Joining the valves to the attached system will involve both field and factory joints. Welding methods and procedures must be identified that are appropriate to the application. Requirements for field and factory joints will be identified at this point, along with NDE and other quality assurance requirements to determine the geometry and material properties to be used for analysis.

3.1.8.3 Internal and External Insulation Attachment Methods

Insulation pins, clips, and other attachment methods will be modeled analytically to determine effectiveness within appropriate environmental conditions.

3.1.9 Installation

Installation techniques will be examined during this step, with regard to the above mentioned items. Welding feasibility, in conjunction with the attachment methods for insulation and cooling systems (if necessary) will be investigated from an integrated standpoint.

3.1.10 Known Valve Failures

The intent of this section is to ensure that during execution of the valve test plan, steps have been taken to safeguard against known failure modes (such as those shown below) by a test program specifically tailored to check against these possible pitfalls. Documentation of the approaches used during valve qualification and testing will help ensure that the necessary precautions are taken during plant operation. Such records should be kept in a special file within the test report. A high level of confidence is necessary to ensure that valve test plans provide the necessary defense-in-depth throughout the plant life.

There will be additional issues due to aging of components, especially with respect to elastomers, packing, lubricants, wiring insulation, seals and highly stressed materials exposed to the working fluid that must be addressed. Many of these aspects are highly specific with regard to valve type (isolation or relief valve, globe, ball, gate, angle, 2-way or 3-way), application (environment, mounting location and orientation, COS, etc). Taken together with the potential valve manufacturers' knowledge base, with regard to valve dependability and safety, the test plan should address all degradations and life issues that may be encompassed, and an inspection plan and design basis verification plan adopted that accounts for prevention of such failures.

3.1.10.1 Mechanical Degradations that have occurred in Power Plant Valves include:

- Handwheel to Motor Clutch mechanical connection, Loose Stem Nut Locknut
- Limit Switch Lubricant Degradation
- Valve Shaft to Actuator Key or Motor-to-Shaft Key Failure
- Loose Anti-Rotation Device Setscrew, Loose Worm Bearing Locknut
- Valve Spline Adapter failure, Coupling Failure, MOV Key Failure

3.1.10.2 Switch Settings causing field failures include:

- Incorrect Torque Switch Bypass Settings or Incorrect Torque Overload and Torque Switch Settings
- Incorrect Torque and Bypass Switch Settings or Low Torque Switch Settings
- Valve Damage due to Backseating
- Incorrect or Improper Switch or Bypass Settings

3.1.10.3 Field Failures due to Valve Sizing Calculations include:

- Failure to Close or Failure to Open Against Differential Pressure
- Undersized Valve Actuators
- Incorrect Valve Sizing Practice
- Underestimated Valve Seat Friction
- Improper Justification and Validation of Analytical Assumptions
- Issues related to Stem Rejection Load
- Actuator Stall Thrust Issues, Thrust Limits and Potential Overstressing
- Valve Torque Requirements over/under estimated

3.1.10.4 Design Issues contributing to field valve failures include:

- Torque Switch Bypass Circuit
- Isolation Valve Position Indicator Signals, Effects of Changing MOV Switch Settings
- Misapplication of Throttle Valves
- Environmental Qualification
- Control Circuit Deficiencies
- MOV Failures due to Hammering
- MOV Motor Burnout Events
- Motor Wiring Environmental Qualification (EQ) Deficiencies
- Stop Check Failures due to Low Flow
- Valve Actuator Qualification
- Valve Damage due to Improper Backseating
- DC Motor Design Issues , DC Motor Cable Sizing
- Horizontally Installed Gate Valves
- Valve Stem Failure from Materials Incompatibility including corrosion and embrittlement
- Improper Installed Position of Plant Valves

3.1.10.5 Maintenance Issues that affect valve failures include:

- Training of Plant Personnel on maintenance issues and Coordination of Plant Personnel During Testing
- Marine Growth, sediment buildup or Corrosion of Valve Internals
- Incorrect Pinion Gear Installation, Failures due to Stem Protector Interference
- Valve Stem Corrosion Failures, Gate Valve Corrosion
- MOV Installation Procedures
- Motor Termination Issues
- Housing Cover Bolting and Component Material Properties
- Failure of Torque Switch Roll Pins
- Binding Valve Stems, Packing and Lubrication

3.1.10.6 Pressure Locking and/or Thermal Binding has lead to failures from:

- Pressure locking and Thermal Binding of Flex Wedge Gate Valves
- Other Valves Susceptible to Pressure Locking
- Thermally Induced Pressurization
- Welding due to Exposure of Hot, Dry Helium

3.1.10.7 Actuator Efficiency and Actuator Rating related failures

- Actuator Performance Issues
- Torque Deficiency throughout operating range

3.1.10.8 Diagnostic Systems

- Results of Industry Validation Testing
- Inaccuracy due to Directional Effects
- Accuracy of Diagnostic Equipment

3.1.10.9 Personnel Safety Issues that can lead to safety deficiencies include:

- Lockout / Tag out Procedures
- Maintenance Procedures
- Tagging Procedures

3.1.10.10 Approaches to ensure reliable, proper and safe valve operation include various tests, inspections, verifications and procedures including: :

- Preventive Maintenance, Periodic Valve Testing and Operational Verification
- Stroke Time and Travel Measurement
- Periodic servicing, replacement of lubrication, lubricant sampling, visual inspection of lubricant condition, actuator lubricant level
- In-Service Leak testing (Valve bypass and stem leakage), packing integrity inspection
- Actuator qualification and periodic testing
- Cleaning procedures, stem and packing inspection, adjustment and refurbishment
- Inspection Procedures to ensure mechanical components are properly configured
- Inspection Procedures to ensure electrical and control circuits are properly adjusted, connected and exhibit correct functionality, wiring condition inspection, strain relief's, wiring seals, flexible wiring
- Diagnostic systems are working properly (by physical test)
- Installation Verification, Maintenance Trends (maintained by software as part of a wider effort focused on plant component reliability)
- Operator Training, Maintenance Personnel Training, Safety Training
- Valve Tagging and proper documentation records
- Other requirements as required by ASME OM Code (Operation and Maintenance in Nuclear Power Plants) and other applicable regulatory documents

3.2 3D Modeling and Analytical Test Simulations

CFD - Flow and Temperatures Modeling

High Temperature Isolation and Relief Valves will be computer modeled using Computational Fluid Dynamics (CFD) software.

3.2.1 Purpose, Scope, Desired Outputs

The analysis is intended to determine entrance and exit flow conditions, thrust force vectors, velocity profiles (including high and low velocity regions and extreme velocity gradients), temperature distributions and temperature gradients, pressure differentials and heat transfer coefficients. Also it must be verified that valves can withstand design basis accidents by subjecting the models to accident conditions. Optimize design.

3.2.2 Assumptions and Approach

The assumptions used to perform the analysis, if any, will be listed in the report. The approach to CFD simulation is to apply the known conditions of service as boundary conditions. This will include the ambient environment temperature, flow values for the working fluid, and outlet pressure. Material assignments for the simulation will be based on the initial material selections. The only material properties of concern for this analysis are the thermal properties since they will affect heat transfer within the model. The working fluid is known and can be assigned the appropriate properties. The heat transfer characteristics of the selected materials and the effect they have on the analysis results may help further reduce the number of candidate materials.

3.2.3 Applied Loads, Constraints and Materials

Details of the type and magnitude of boundary conditions and loads will be provided with the analysis. Also the specific physical properties (for the working fluids and construction materials) and thermal properties used will be shown. If multiple analyses are performed using different materials, all material properties for each analysis will be presented. Figure 2 shows an example of a CFD result allowing visualization of the performance of a typical valve. Verified material properties obtained during the previous TRL level will be utilized.

3.2.4 Accept/Reject Criteria

These criteria will be based on the applicable codes and standards, material property limitations, valve operability and reliability.

3.2.5 Results

The results will allow visualization of the inlet, internal and outlet flow patterns and velocity profiles, overall heat transfer coefficients, differential pressure, thrust force vectors, the temperature distribution throughout the valve, insulation (if equipped), heat loss, and individual component temperatures. Results can be prepared that show the valve in different operating modes. Results are quantified on a relative scale.

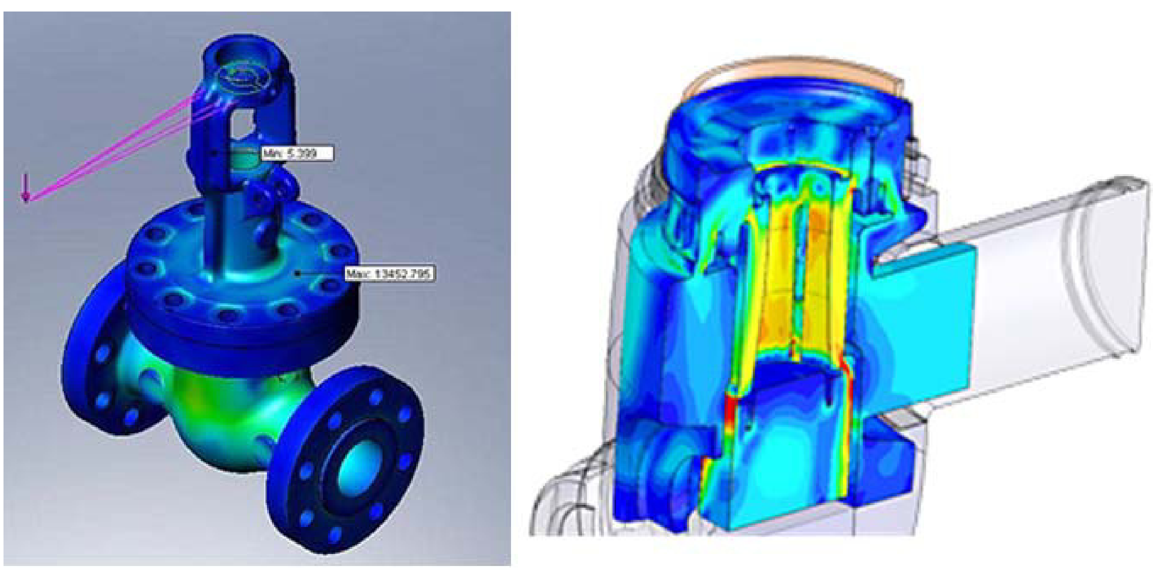


Figure 2: CFD Results from a model of a typical valve

3.3 Acoustic and Flow Induced Vibrations

3.3.1 Purpose, Scope, Desired Outputs

Acoustic and flow induced vibrations will be examined using CFD analysis and acoustic modeling to determine if the selected configuration contributes to any system vibrational issues. Sonic energy from vibration can cause a great deal of harm throughout the system and can lead to other failures. This analysis is a critical step in determining overall system performance. Actual acoustic performance characteristics will be measured using full scale test articles in TRL-8. This study provides tendencies and trends to provide input to inform designers of areas of concern that should be investigated during future studies. Acoustic modeling, conducted by a subject matter expert, can provide insight and guidance into design aspects that should be avoided,

3.4 Differential Thermal Expansion Analysis

3.4.1 Purpose, Scope, Desired Outputs

In a high temperature environment, differential thermal expansion is a major concern, especially during start-up and transients. CFD modeling of thermal gradients within the valves and their associated systems will be performed to assure the materials selected will withstand maximum temperature experienced, and no unacceptable hot spots are present. Distortion of the valve body due to thermal loads must also be demonstrated to ensure binding or loss of seal integrity does not occur during service.

Figure 3 provides an example of thermal analysis of a typical valve body.

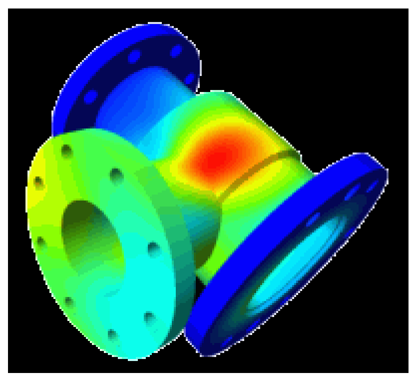


Figure 3 – Analytical Temperature Modeling

3.5 FEA Stress Analysis

Detailed models of the high temperature Isolation and relief valves will be constructed using the latest Finite Element Analysis (FEA) software. Software used for analyses will be validated under the NQA-1 quality assurance program.

3.5.1 Purpose, Scope, Desired Outputs

Several FEA analyses will be performed to in investigate stress, strain and deflection related to determine the following:

- Leak tightness of the seals
- Adequacy pipe to valve connections
- Response to temperature loads
- Differential thermal expansion calculation verification

- Stress imposed on attachment points
- Response to external loads and design basis hazards

Temperature profiles from CFD and thermal models and resultant thermal stresses from FEA analysis will be represented for all aspects of the FEA stress analyses. Elevated temperature material properties will be used and the stress values compared to acceptable code acceptance criteria using validated material properties.

See Figure 4 for a typical FEA analysis visual output.

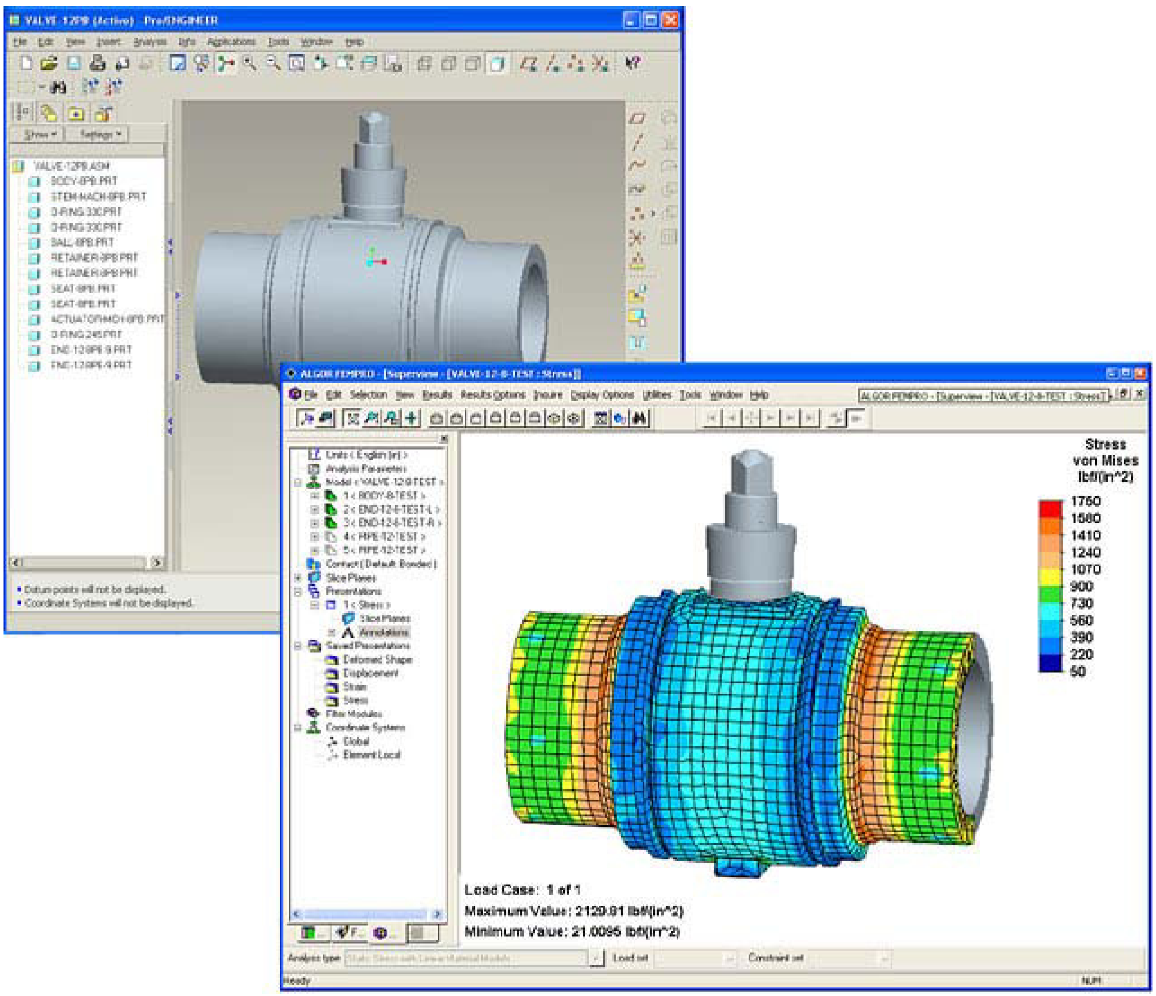


Figure 4 – Typical FEA Analysis

3.5.2 Assumptions and Approach

The valve assembly will first be modeled to investigate the mechanical and structural adequacy of the valve and pipe supports. This initial analysis will also reveal the need for any external valve supports. This initial model will then be expanded to include a larger portion of the system and construction joints to investigate stress in the connections resulting from externally applied loads.

Because this analytical modeling technique is applied to several different valve configurations, a number of 3d FEA models will be constructed.

Several iterations of the FEA analyses must be completed to investigate potential limitations or relative benefits of the initial material selections. A table of results containing all applied loads and constraints for each FEA analysis and the materials used will be provided.

Stress values will be compared against allowable code values. Loads imposed from external sources (piping and actuator) under combined loading scenarios will be compared to allowable material limitations. Deflection at the mating flanges and sealing surfaces must remain within acceptable and vendor approved deflection values.

3.6 Endurance Limit Analyses

The endurance limit of a component is determined through an analysis that considers all factors that contribute to the expected component life including static and cyclic loads, temperature, creep, fatigue, erosion, corrosion and other factors. Localized stresses from FEA analysis combined with CFD results for local and component temperatures will be utilized in the Endurance Limit Analyses. An independent subject matter expert will assist the testing organization in completing this task, which will involve reviews of models, FEA results and test data. This will provide an independent review of the body of knowledge and conclusions made to this point.

3.7 Creep Analysis

The purpose of the creep analysis is to ensure that the materials do not permanently deform under the influence of high temperatures and stresses (below acceptable code values) over an extended period of time. Both analytical modeling (FEA) and coupon tests will be utilized in the Creep Analysis. Creep analysis makes use of non-linear modeling techniques to be performed by subject matter experts.

3.8 ALARA Analysis

The purpose of the ALARA analysis is to ensure that radiation doses to workers are as low as reasonably achievable under the anticipated operating conditions and modes including inspection and maintenance. Valves will be examined for potential contamination traps, which could lead to increased exposure during maintenance activities. The radiation dose to the exposed workers will be estimated by using 3d modeling techniques that incorporate materials of construction of the valve and other nearby radiation sources as well as a portion of the physical environment the valve resides within. ALARA trained personnel will consider personnel protection requirements, and if temporary shielding is needed, then this too will be incorporated in the model. This will be part of a larger effort conducted by an integrated project team (IPT). Qualified URS-WD nuclear engineers, in the Princeton Office, will perform the radiation field modeling.

3.9 Routine and Non-Routine Maintenance Requirements

Remote maintenance requirements are to be defined at this point in the test plan execution. Possibilities exist for personnel exposure during routine maintenance activities, depending on valve physical location within the integrated plant site, scenario under which maintenance activity is being conducted, and service system that the valve is a part of.

In conjunction with determining the safety functions of each of the valves in the system, the method of actuation and control, indication, and maintenance schedule requirements must be determined before advancing to TRL 4. Significant impacts on the overall design of the valve(s) and the testing program to be imposed are dependant on the above mentioned determinations. Periodic relief valve testing, and the method of testing, for example may necessitate installed monitoring devices (if tested in-place), or serviceability studies (if being removed for testing).

A complete valve list is necessary to complete this activity and will be compiled during this TRL level. Table 3 is partially complete and reflects some of the valves to which this test plan currently apples.

High T. Valve	Service	Type	Maintenance
Isolation Valve	IHX He Cold Leg	Top Entry Ball, Manual actuation,	Ball Seat, seat seal, packing, actuator, position indication
Isolation Valve	IHX He Hot Leg	Top Entry Ball, Manual actuation	Ball Seat, seat seal, packing, actuator, position indication
Isolation Valve	Secondary heat exchanger	Top Entry Ball, Manual actuation	Ball Seat, seat seal, packing, actuator, position indication
Isolation Valve	Secondary heat exchanger	Top Entry Ball, Manual actuation	BallSeat, seat seal, packing, actuator, position indication
Relief Valve	Vessel Coolant He	Spring loaded Angle Globe with pressure actuated plug	Periodic testing and Calibration, seat and seal maintenance, leak check

Table 3: Valve List

Different plant areas and services require differing degrees of remote maintenance. Test articles (full size and pilot scale) and FEA models must reflect aspects of the valve design that are present to facilitate remote maintenance. Remotely maintained valves are typically accessed through a shield plug, (shown in Figure 5) and specially developed long reach tools are used to perform maintenance.

Actuators and position indicators may be physically separated from the body and extended linkages and drive shafts may be employed to actuate the valve stem. Typical valve configurations feature ‘top-works’ or top entry bodies with bolted removable bonnets that can be accessed completely from one side (or from above) for all service and inspection activities. Custom fasteners with remote maintenance provisions (shown) are employed to allow removal of all serviceable parts for maintenance, replacement, or inspection. Seats, likewise, have custom features to allow and facilitate in-place refurbishment to restore damaged areas. All the ways that remote maintenance features influence the geometry, stress distribution and process fluid flow within the valve must be reflected in test articles and FEA models.

Valves suitable for access using contact maintenance methods will be serviced normally requiring no special procedure development. Appropriate techniques for controlling the spread of contamination will be documented and employed for all valves.

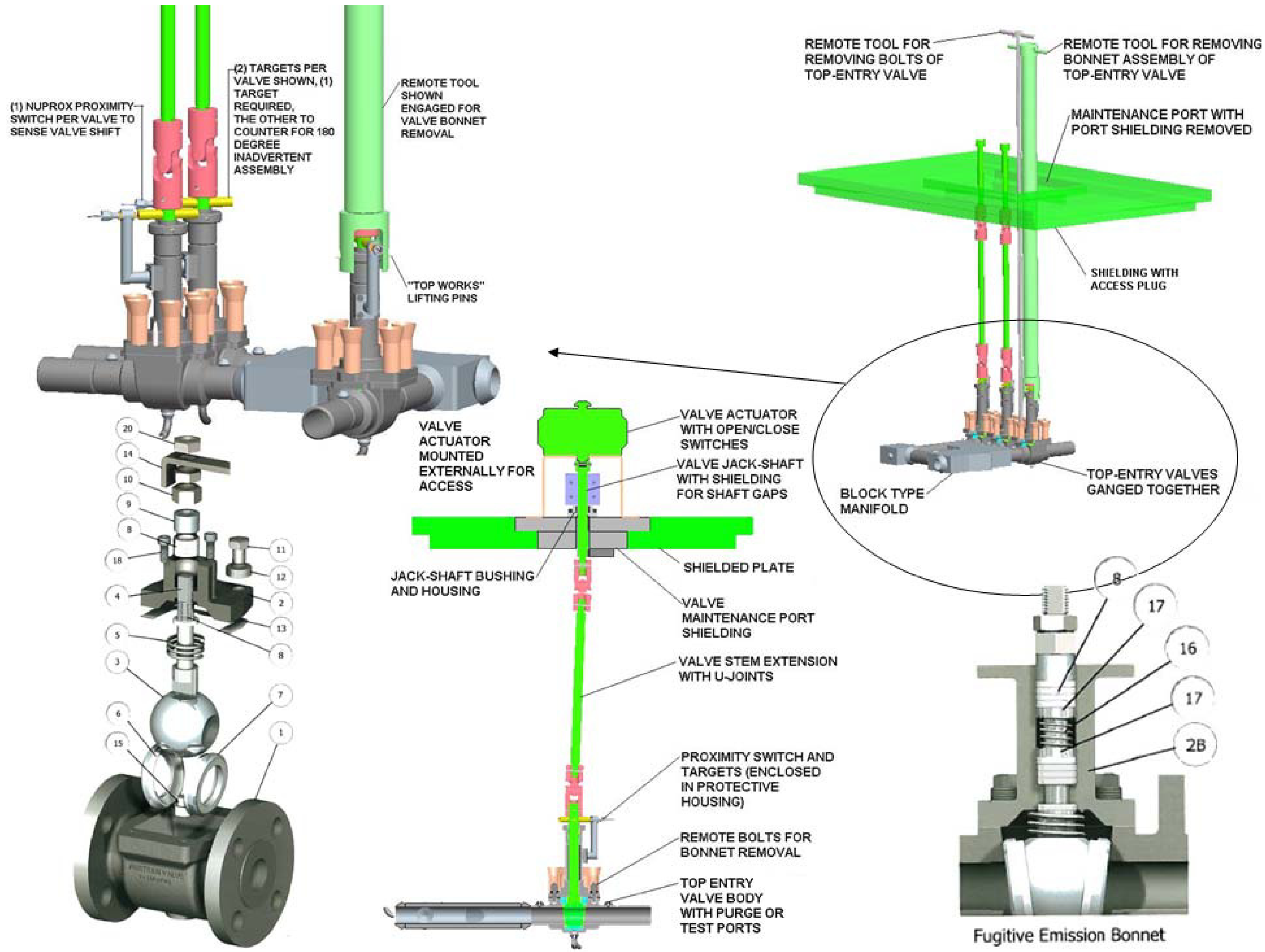


Figure 5: Ball Valve configuration and setup for remote maintenance

3.10 RAMI

As part of an integrated plant program, Reliability, Availability, Maintainability and Inspectability (RAMI) analysis will be performed to ensure that the high temperature valves will meet mission needs safely with minimum life cycle cost. The RAMI analysis task involves a process of identifying top-level (major) system availability requirements, decomposing these requirements into meaningful downtime statements for subsystems and/or components, and formally summing these downtimes to estimate the availability of the entire interactive system.

Standard engineering reliability methods are utilized to determine the mean time of service up to failure for the component (valve) in question. The reliability analyses for each sub-component of the system being analyzed are combined to calculate the mean time to failure (MTTF) for the analyzed system. Industrial data on existing valves within the nuclear power plant environment will be utilized to determine MTTF.

The next step in establishing a RAMI program is to develop a requirements statement to define the following parameters.

- Operational needs for the design life of the component
- Expected normal and worst-case operating conditions
- Expected downtime for either corrective or preventive maintenance actions.

The requirements statement is used to create an availability statement for the plant. Stating the total uptime needed for the system or subsystem establishes the allowable downtime. The total downtime is then allocated to all the lower tier (component level) systems in the form of design requirements. After the component downtime is allocated to each of the involved subsystems, analytical techniques are used to estimate the actual downtime expected to be experienced by the various subsystems during operation. These estimates include failure frequency (FF) and the mean time required to return the failed system to operational status, or mean time to restore or repair (MTTR). The estimates are then summed to estimate the availability of the system as designed and compared with the availability requirement (A) as a measure of design success.

$$A = MTBF/(MTBF + MTTR)$$

The ease of maintainability of the component contributes to the mean time to restore. Components designed to facilitate maintenance will, in turn, contribute to the system's overall availability. Inspectability, built into the design, allows for operational parameters and performance to be closely monitored allowing preventive maintenance to be scheduled with greater efficiency. As a part of a larger RAMI program, this allows for coordinated and more precisely scheduled maintenance that helps eliminate maintenance when it's not needed and encourage maintenance that positively impacts availability. Improved inspectability and performance monitoring also helps to prevent unanticipated outages due to in-service faults.

The relief valves achieve the inspectability function because they will be verified initially then periodically during service to ensure proper functionality to both open when pressure is higher then the set point and close completely when the excess pressure has been relieved. They will either be removed from service and tested during a re-fueling or other planned outage, or tested in-place, assuming a dual relief valve installation. A rupture disk placed downstream of the relief valve outlet will allow placement of a pressure sensor between the valve and the disk to monitor leakage of primary coolant past the seal.

If the reliability of isolation valves is not high enough, often redundant valves are placed in series and locked out (with pressure monitoring in the interstitial space) prior to performance of maintenance on downstream equipment.

Consideration of performance of maintenance will be a priority in the design, arrangement and location for all the high temperature valve applications. (see section 3.9 above)

4.0 TASKS TO ACHIEVE TRL-6 (COMPLETE COMPONENT TESTING)

TRL-6 is defined as component verification and demonstration integrated into a partial subsystem at <u>pilot scale</u>. This level is meant to provide the necessary design data for complete integrated component demonstration. Although the test article may not be an exact model of the final component design, it should be sufficiently representative to serve as a basis for performance demonstration.

4.1 Test Objective

For the high temperature isolation and relief valves the physical testing performed in this TRL is designed to provide validation of valve performance, performance of the selected materials while being subjected to simulated environmental factors, and the installation techniques. In addition, tests involving instrumented test articles will provide verification of the CFD and FEA analyses performed earlier on full scale and scaled down models. Once testing is completed life cycle cost analyses will be performed.

4.2 Physical Test Preparation

Due to the expense involved with CTF testing, physical testing requirements for TRL 5 and higher should be based in part on a risk-based assessment to determine what testing will be done at the CTF. This assessment will be based on confidence levels in the analysis results and uncertainties associated with these analyses, model tests, material tests and performance data attained through simulations. Weighting factors will be developed to objectively determine a feasible required test plan for CTF testing.

Scaled down test articles will be produced for testing. An appropriate scalable model size will be selected, and scaled down valves will be built to examine performance, and validate analytically determined behaviors. Scaled down valve flow tests will incorporate Reynolds Number similarity for scaling parameters.

Valves designs are usually available over a range of sizes. The full size article is likely to be tested at the CTF, and smaller sized units can be used at this level of testing. The test articles used in pilot scale testing during this phase will therefore be sufficiently representative to verify the performance parameters being sought at this level of design development.

4.3 Test Apparatus

Test apparatus will consist of a flow loop in which scaled down valves are placed in service and subjected to simulated environmental conditions. A typical test leg, (from the literature) is supplied as a reference as Figure 6 below.

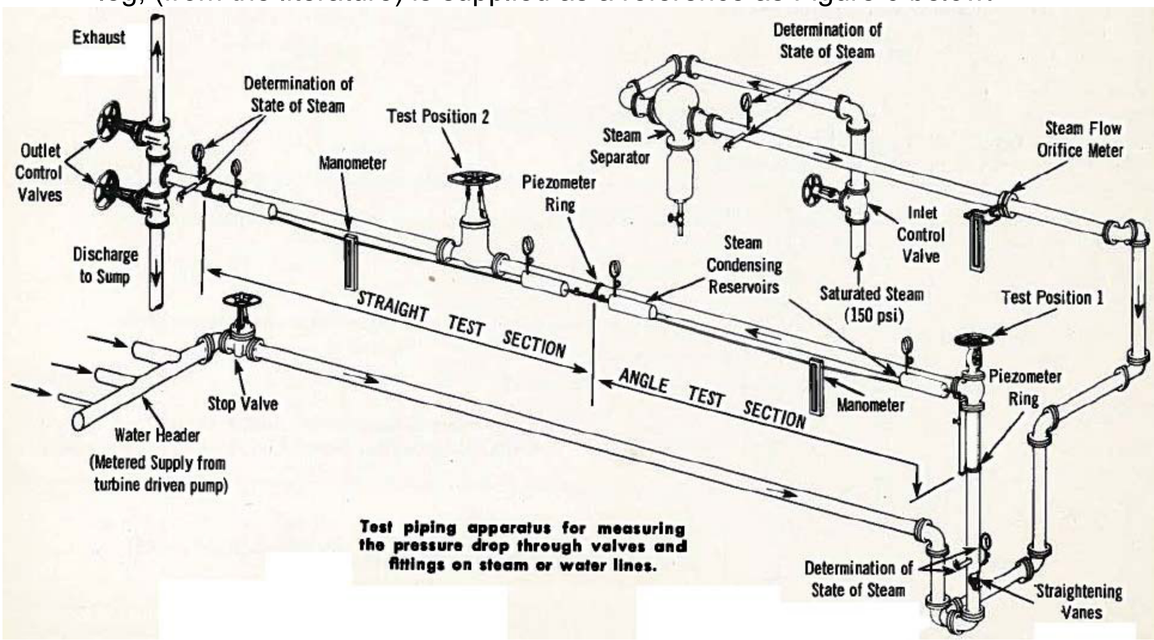


Figure 6: Typical Valve Test Leg

Pressurized, circulating hot helium (electrically heated) will be used as the working fluid. Appropriate flow, temperature, and pressure measuring instruments will be utilized, along with parameter recording equipment. Strain gages, at critical locations on the valve body, will be used to measure localized strain and readings will be used to validate analytical models. In the case of globe style valves, stem force, applied externally from a pneumatic actuator will be measured to determine seating forces. Helium leak detection methods will be used to determine seal and/or packing leakage. If material performance limit testing is performed, NDE methods such as dye penetrant testing will be used to determine pressure boundary integrity. If cyclic testing is involved, then detailed post-service wear inspections will be performed. Actuator performance requirements will be measured and compared to predictions.

This test will be performed at the valve test facility under direction of the Test Plan Director. Valve manufacturers may also be involved in testing using their own capabilities as specified in section 8 of this report. Alternatively, if highly specialized testing is required, the generalized testing organization shown under section 8.4 may be utilized.

4.4 Determine Methods of Conducting Valve Inspections

Pre-determined in-service inspection methods will be fully developed and verified at this step. To accomplish this aspect, valve components must resemble those that will be featured in the full scale design. In conjunction with scale testing and CTF test preparation, methodology for inspecting valve operability, leak rate, external leakage, valve integrity, seat integrity, and weld quality will be developed. Relief Valve operating characteristics inspection method is to be developed. Actuator design aspects, such as heat shields, fins on the extended stem to dissipate heat, and heat transfer barriers must also be present on the test article. See 4.3 for testing location.

5.0 TASKS TO ACHIEVE TRL-7 (FINAL FULLY INTEGRATED VALVE SUB-SYSTEM TESTING)

This development level involves verifying the design for the high temperature valves at the fully integrated sub-system level. This sub-system is comprised of the complete valve subsystem including the valve body, bonnet, plug, seal, packing, insulation, ball and seat, stem, bellows, jacket, drive shaft, couplings, gearbox and actuator (as equipped), together with all lubricants, control equipment position indicators and support systems. The FEA and CFD simulations completed for the component level (previous section) will be built upon to include all the sub-system components. The physical test will be comprised of all sub-system components under all differential pressures and operating temperatures over the full range of operating conditions.

Also at near the end of this TRL level, an assessment will be made by the test director of the test results collected here-to-for during the test plan, and consideration of the risks and benefits of performing full scale valve tests at the CTF will be made. It is anticipated that during the next TRL level the full size relief valve and a full size isolation valve test will be performed, however it is also conceivable that this level of testing will be deemed sufficient to fully qualify the high temperature valves for their intended service.

5.1 Sub-System Integrated Experimental Scale Model Test

Based on the design of the valve(s), driven by CFD and FEA analysis up to this point, scale model testing will be conducted. A test loop will be configured (See Section 8.1 for description of required test facility) to integrate the valves to be tested into a simulated operational environment. Scale model valve testing may be conducted together with other technology development programs for which environmental simulation may be a part, hence an opportunity for coordinated testing may exist.

Scale model testing to achieve TRL-7 will be used to validate the analytical simulations performed up to this point. Testing in a simulated plant environment presents an opportunity to diagnose potential problem areas, investigate areas of concern identified during the analytical process, and may provide pathways to additional design optimization.

Hot, dry helium flow concepts developed will be validated in the scale model test loop, which will consist of, at a minimum, an integration of proposed high temperature isolation valve together with actuator, seals and position indicators, and a pressure relief valve. The loop will be equipped with a helium pressure source, a gas flow producer, heaters, and interconnecting piping with the necessary insulation. The test loop will be arranged in a manner to simulate actual predicted operating conditions for NGNP with regard to duct geometry, valve placement, and flow velocities. Velocity induced vibrations, flow profiles and pressure drops can be examined through the actual valves in scale size. Reynolds number similarity will be used to scale down flow cross sections.

Helium coolant composition, density and temperature will be used as the test parameters for the working fluid

Relief valve operating behavior, including seat to body tightness, lifting forces exerted on the plug and seal, external forces from the adjoining piping, and re-seating characteristics will all be examined. Necessary actuator force parameters will also be determined for isolation or 3-way valves. These are valve behaviors that are difficult to predict accurately with analytical modeling alone.

An integrated scale model test plan will be determined on an "as needed" basis. Many of the test runs to be conducted will be based on problem areas identified during analytical modeling. Additional or continued scale model testing will be dictated by the performance of the valves and comparison with the model. The necessity to re-test, or modify tests conducted to achieve a high level of confidence in proposed valve characteristics and configuration will be determined by the test director before moving to the phase of development that involves full size testing.

Instrumentation on the test loop includes those necessary for parameter measurement in addition to seat closure force measurement, actuator force requirements and measurement of pressure boundary strain at critical locations. Post service NDE and destructive inspection methods may be employed to investigate the effect of exposure to the test environment on components and material coupons.

5.2 FEA Simulation System Optimization

Based on the results of scale model testing, additional FEA simulation runs may be necessary at this point. As described above, many of the steps to achieve TRL-7 will be conducted on a case by case basis, each test being dependant on the outcome of another.

In order to achieve TRL-7, personnel responsible for analytical modeling should work closely with those conducting the physical testing, in a collaborative effort to overcome problem areas, accurately observe and predict performance, and to optimize the overall designs.

5.3 CFD Simulation System Optimization

As mentioned above, additional CFD modeling may be required depending on the results of the scale testing. This process may require several iterations.

5.4 Final Leak Detection Validation

In conjunction with pilot scale testing, the final leak detection method developed earlier will be validated. A proof positive method to determine the magnitude of valve leakage will be stated and tested as part of the scale model simulations. Both isolation valve leak-by, and relief valve leakage (during normal operation and post-actuation re-seat leakage) will be simulated and measured.

5.5 Maintenance, In-Service Test and Inspection Techniques Validation

Proposed concepts for maintenance, testing and inspection will be proven during scale testing. The possibility of valve body welding prohibiting disassembly will be investigated; analytical and theoretical concepts will be validated. Material and component durability will be examined during post service inspection. Field deployed NDE methods and inspection access concepts developed will be examined. Some level of rework may become necessary depending on the results of the validation process. Design constraints to achieve access and special tools needed to perform service will be documented.

6.0 TASKS TO ACHIEVE TRL-8 (FINAL SYSTEM TESTING)

This TRL step involves integrated CTF testing. This step will be completed consistent with risk analyses performed previously. Integrated CTF testing will be completed in conjunction with other reactor component or subsystem tests at the Component Test Facility at INL. This testing would be coordinated with testing of other NGNP components.

6.1 Integrated CTF Testing

Valve testing at the CTF will be conducted based on the risk analysis conducted and the results of the analytical modeling and integrated scale model testing. CTF testing is considered the last step in equipment validation. Note, based on results of integrated scale model testing (previous section), CTF Testing of full size valves may not be necessary for TRL level advancement, instead the test director, in consensus with GA, may deem integrated scale model testing suitable for achievement of TRL-8. The following assumes all or some of the high temperature valves are tested at the CTF.

Valve testing will involve many subsystems at the CTF. In some cases perhaps this will be the first time these full-size systems have been integrated, and testing at this level will validate their overall compatibility.

Valves, both relief and isolation will be manufactured full scale and installed into a test loop and are considered an integral part of the loop itself. Prolonged operation in simulated actual conditions (minus radiation effects) at full scale will validate all concepts tested both analytically and in sub-scale size. Again, re-work potentially involving analytical modeling and/or scale testing may be necessary depending on the findings at this point.

6.2 In-Service Inspection Techniques Validation

Examine test article components for rupture, creep, swell, fatigue cracks, seal degradation and insulation effectiveness: Use developed methodology for inspecting valve operability, leak rate, external leakage, valve integrity, seat integrity, and weld quality to examine valves in test loop. Verify relief valve lifting characteristics and closing reliability. Verify actuator design parameters. Employ appropriate inspection methodology and remote maintenance techniques

6.3 Stress Analysis Validation

The purpose of this test is to verify the FEA stress, strain and deflection results. Behavioral predictions and results from FEA models will be verified by comparison with results from instrumented testing.

6.4 Temperature and Flow Analysis Validation

The purpose of this test element is to confirm the CFD results. Flow velocity, aspects of flow profile, flow magnitude, and temperature predictions will be verified. The acoustical signature of the full size valves will also be tested during this phase.

7.0 QUALITY ASSURANCE REQUIREMENTS

All aspects of the QA plan shall be compliant with the Quality Assurance Program Plan (QAPP) of General Atomics. An outline is provided below of the URS-WD QA Plan for NQA-1 projects that pertains to the test plan elements (described above).

7.1.1 Program and Organization

7.1.2 Training

7.1.3 Personnel Requirements

7.1.4 Limiting Conditions

7.2 Design, Engineering and Data Control

7.2.1 Inputs

7.2.2 Drawings

7.2.3 Specifications

7.2.4 Criteria Documents

7.2.5 Revisions

7.2.6 Change and Configuration Control

7.2.7 Design Analysis

7.2.8 Design Review

7.3 Verification

7.3.1 Alternate Calculations

7.3.2 Design Review

7.3.3 Testing Under Most Adverse Conditions

7.4 Procurement

7.4.1 Procurement Document Control

7.4.2 Review

7.4.3 Approval

7.4.4 Handling, Storage and Shipping

7.4.5 Instructions, Procedures and Drawings

7.4.6 Control of Purchased Items and Services

7.4.7 Certification

7.4.8 Source Verification

7.5 Inspection

7.5.1 Shop Inspection

7.5.2 Post Installation Inspection (field)

7.5.3 Control of Special Processes

7.5.4 Test Control

7.5.5 Control of Measurement and Test Equipment

7.5.6 Result Documentation

7.5.7 Inspection, Test and Operating Status

7.6 Identification and Control of Items

7.6.1 Control and Disposition of Supplier Nonconformance

7.6.2 Corrective Action

7.6.3 Commercial Grade items

7.6.4 QA Records

8.0 TEST LOCATION AND TEST PLAN SCHEDULE

An outline schedule of the Valve Test Plan is provided below.

Readiness Level	Year (FY 20xx)											
	09	10	11	12	13	14	15	16	17	18	19	20
NGNP Schedule	Conceptual Design Prelim Design				Final Design for NGNP							
						Site Work			Construction			
											Startup / Testing	
CTF	================================											

TRL-4[1]	= = >
TRL-5	< = = = = >
TRL-6	< = = = = >
TRL-7	< = >
TRL-8	< = = = =

[1] COS and other design bases provided in a timely fashion to determine test parameters

Using the current NGNP schedule, valve testing tasks are completed well within the final design phase and within the long lead procurement period. Additional time has been allotted for CTF testing of full size prototype valves to enable coordination with other entities. Should schedule priorities demand, individual test element durations can be decreased by as much as 50% with the exception of coupon testing which could be accomplished in approximately eight months if necessary.

8.1 Required Valve Test Facility Capabilities

Testing facility for high temperature valves will require the following capabilities.

8.1.1 High pressure helium storage capacity

Valves (both isolation and pressure relief) will be subject to constant helium exposure. A source of pressurized, high quality helium must be available for a variety of testing activities, including:

- Valve body helium permeability testing
- Valve seat leakage testing
- Material erosion testing utilizing high velocity pressurized helium
- Flow verification testing
- Relief valve actuation testing
- Isolation valve operability testing with dry helium flow

8.1.2 Helium heating capability

Testing facility must have capability of heating high purity, pressurized helium mentioned above for valve testing at elevated temperatures.

8.1.3 Materials heating capability

Testing facility must have high temperature heat source, autoclave or similar for material testing at high temperatures.

Facility must be capable of producing and maintaining plant peak operational temperatures for valve operational testing, including cycling and relief valve popping at elevated temperatures.

Raw material testing, such as valve seat, internal insulation, valve body and actuators must be capable of testing at maximum plant operation temperatures as part of environmental qualification of materials.

8.1.4 High accuracy Flow, Temperature, and Pressure Instrumentation

Testing facility will have all applicable flow, temperature, and pressure measurement devices available. These devices will be calibrated according to the applicable standards, and be subject to frequent inspection.

Proposed testing configuration will consist of a bank of pressurized, high purity helium cylinders stored at room temperature. In-line filtration, resistance heaters, recirculation, and pressure boosting compressors will be available to produce a supply of clean, dry helium at elevated pressures and temperatures to the applicable testing rig.

Testing rigs will consist of flow verification equipment where prototype valves, seat configurations, body designs, etc. can be subject to scaled flows of primary coolant quality helium flows. Test rig will be fitted with high accuracy, calibrated flow instrumentation to precisely meter and record observed flow, and flow characteristics.

Other test rigs will include high temperature "ovens" where selected materials, actuator designs and seat/body configurations can be subject to high temperatures for short and prolonged durations. Remote access methods will be necessary to monitor and control actuation and operation of materials, actuators, valves, etc. High accuracy temperature, calibrated measurement and recording equipment will be available for use.

A safe and previously tested method of relief valve testing will be available at proposed testing location. High accuracy, calibrated flow measurement and recording equipment will be available in addition to pressure and temperature measurement of the same quality for testing of relief valve operation at scale level using high temperature, dry helium. The provision to introduce impurities (such as may be found during NGNP operation) to the flow stream will be available.

8.2 Proposed Valve Test Locations

8.2.1 Proposed Test Location for Custom Relief and Globe type Valves: Target Rock

8.2.2 Qualifications and Capabilities

Curtis Wright Flow Control, particularly Target Rock Division is a qualified manufacturer of high temperature, highly specialized valves. They currently produce most of the valves for the United States Nuclear Navy, as well as valves placed in unconventional commercial applications. They have been a continuous holder of the N-stamp since 1968.

Target Rock is an example of a commercial facility for manufacturing and testing valves in various configurations. One hundred percent of their product line is produced in-house in Long Island, New York. When contacted for NGNP valve input, Target Rock was enthusiastic about the possibility of assisting in the design, engineering and testing of the high temperature valves required.

The company has considerable experience with high temperature, limited leakage valves in critical applications, and has pioneered the design of bellows sealed solenoid actuated products which are highly effective in sealing difficult fluids, such as hot helium. Target Rock currently has several valves installed and operating at the PBMR hot helium test loop in South Africa, and is one of the only valve manufacturers with relevant experience in the industry. See Figure 7 for a photo of Target Rock Facility.

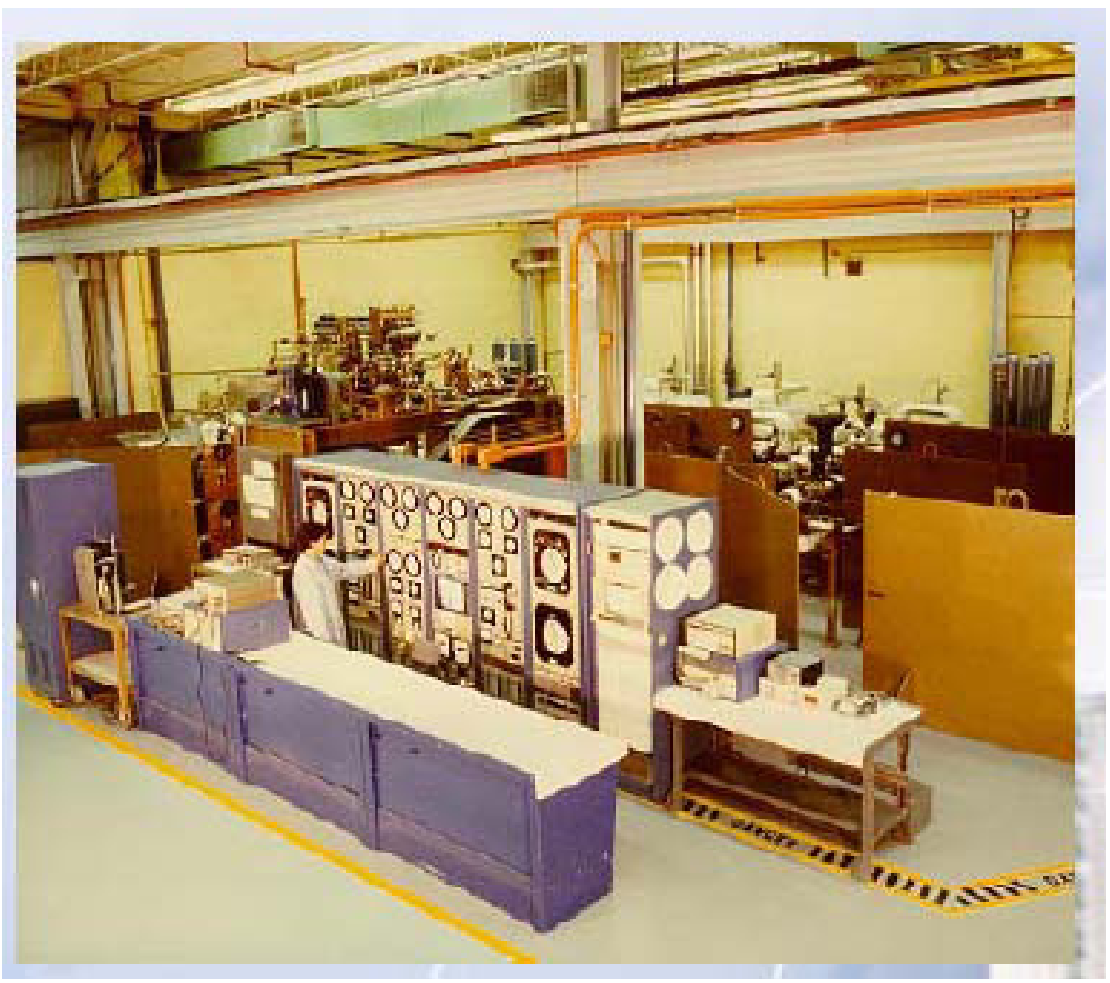


Figure 7: Part of Target Rock's Testing Facilities

Note: See Attached Qualification Summary Sheet from Target Rock

Curtis Wright Flow Control Corp.

TARGET ROCK Division

East Farmingdale, NY

Attn: Steven Pauly – Vice President, Energy Products

8.2.3 Proposed Test Location for Top Entry Ball Valves: Bertrem Valves

8.2.4 Qualifications and Capabilities

Bertrem Valve is a NQA-1 qualified manufacturer of Custom and high temperature Top Entry Ball Valves. Bertrem is an example of a commercial facility for manufacturing and testing valves in various configurations. 100% of their product line is produced in Tulsa Oklahoma USA. They have expressed interest about the possibility of assisting in the design, engineering and testing of the high temperature valves required. The company has considerable experience with high temperature, limited leakage ball valves in critical applications, and has pioneered the design of top access valves suitable for remote access. An example of their top access valve is provided below (see figure 8). Valve designs are available up to 8". 10" size valves may be above the limit of commercially available actuators. [(2)] Besides their design and manufacturing capabilities, their testing capabilities include:

- Fire Testing and Certification
- Data Recording (pressure and temperature)
- Low Temperature Valve Performance Testing
- Actuator Torque Measurement
- High Temperature Testing with Hot Helium

Bertrem Valve Company

6519 East 21st Place

Tulsa OK 74129

Tel. 918-838-3373

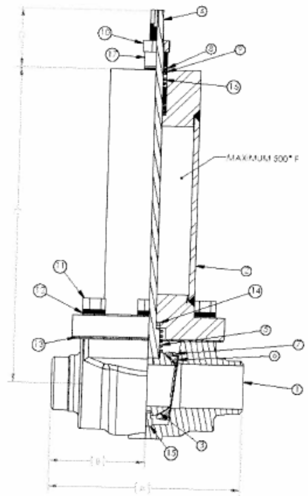
Attn: Brad Bertrem (bbertrem@bertrem.com)

BERTREM VALVE COMPANY, INC.

Top Entry Ball Valves
Buttweld - Zero Emission Bonnet
Working Pressure: 1000 psig
Working Temperature: 1700 degrees F

Item	Description	Material
1	Body	Inconel 625 ASTM A494 GR CW-6WC
2	Bonnet -Zero Emission	Inconel 625 ASTM A494 GR CW-6WC
3	Ball	Inconel 625 Chrome Carbide Coated
4	Stem	Inconel 718
5	Grounding Spring	Inconel 600
6	Seat	Inconel
8	Packing	Graphoil
9	Packing Follower 1	321 SS
10	Packing Nut	18-8
11	Cap Screw	18-8
12	Spring Washers	301 SS
13	Bonnet Seal	Graphoil
14	Packing	Ceramic
15	Trunnion	Ceramic
16	Packing Spring	Inconel 600
17	Packing Follower 2	Inconel 718

Materials and dimensions subject to change without notice



Dimensions					
Size	A	B	C	D	Wall
1/2	6-1/2	3-1/4		2-5/16	Sch. 80S
3/4	6-1/2	3-1/4		2-5/16	Sch. 80S
1	6-1/2	3-1/4		2-5/16	Sch. 80S
1-1/2	7-1/2	3-3/4		1-5/8	Sch. 80S
2	8-1/2	4-1/4		1-5/8	Sch. 80S.

Bertrem Valve Company, Inc.
2205-A NORTH WILLOW AVE.
BROKEN ARROW, OKLAHOMA 74012

Phone: (918) 294-3960
Fax: (918) 294-3969

Figure 8: Bertrem High Temp Extended Stem Top Access Ball Valve

(2) Other manufacturers of top access valves will be surveyed and interviewed. Other manufacturers include Flowserve

Flowserve Corporation
5215 N. O'Connor Blvd.
Suite 2300
Irving, TX 75039
(972) 443-6500

8.3 High Temperature Valve Test Plan Execution

It is proposed that the High Temperature Valve Test Plan and Valve Detail Design activities be completed by the URS Washington Division Test Group in Denver. Qualifications are on file with BEA

7800 E. Union Ave, Suite 100

Denver CO 80237

Attn: dave.carroccia@wgint.com

Phone 303-843-2038

8.4 Erosion and Corrosion Tests

The test director recommends Hazen Research to provide miscellaneous testing services including fabrication and operation of specialized apparatus required for erosion and corrosion testing of valve components. Hazen services include laboratory-scale research on new processes or adaptation of known technology to new situations, followed by pilot plant demonstration, preliminary engineering, and cost analysis. Projects range from beaker-scale experiments, material testing and analyses to multimillion-dollar continuous pilot or demonstration plants. Activities began at the present location in Golden, Colorado, in 1961 and has since grown to a staff of over 120. Sixteen buildings containing an extensive inventory of laboratory and process equipment provide the flexibility for evaluating different unit operations.

Hazen Research Inc

4601 Indiana Street
Golden, Colorado 80403
Phone: (303) 279 4501

www.hazenusa.com

8.5 Non linear analysis and ASME Code Compliance

Becht Nuclear Services

Becht Nuclear Services counts on an outstanding team of industry experts in structural engineering, system design and thermo-hydraulics, mechanical design and integrity, materials and failure analysis, welding and corrosion engineering. The Becht Nuclear Services staff and advisors are here to assist with engineering services, solving nuclear power and nuclear process plant issues with technical excellence, in a responsive and cost effective manner.

Headquarters
22 Church Street, P.O. Box 300
Liberty Corner, New Jersey 07938
Toll Free . . . 800-772-7991
Telephone . . . 908-580-1119
Attn: Greg Hollinger [ghollinger@becht.com]

8.6 Acoustic Testing

TBD

9.0 REPORT FORMAT

A common **report format** will be adapted as warranted to each element of the test plan to present the requirements and results. A draft common report format is introduced here that includes the sections shown below. A description of the contents of each section is provided following the list. A **Test Plan Blank** (common form to be filled out for all test plan elements) is provided in the appendices within the Test Plan for the high temperature Helium Duct and High Temperature Insulation.

- Test Identifier
- Purpose and Scope,
    - Features or Aspects to be tested
    - Features or issues not to be tested (excluded elements)
- Test Approach, Assumptions & Input Data
- Suspension criteria, resumption requirements and contingencies
- Resource needs and rationale
- Schedule
- Acceptance/Rejection criteria
- Approval of Certifications and Assumptions
- Properties/Criteria, References
- Roles and Responsibilities, Limiting Conditions of Operation
- Test Results and Result Summary
- Conclusions
- Path Forward
- Data (Appendix)

Test Identifier: Each test element will have a unique name and number, and all related documentation will be so marked.

The purpose and scope of the test or simulation will be provided that describes the reasons, intentions, objectives and functions to be tested. The application of loads and the range of variables to which the test item shall be subjected shall be indicated. The particular feature, property or characteristic which is the focus of the test will be identified. All necessary features and aspects of the test or simulation shall be designated. Features, components or influences that are to be excluded or bypassed (if any) shall be stated.

A description of the test approach that outlines the strategies involved in the test or simulation will be provided that includes everything that will be part of the test, and how the objectives are to be realized. This section of the test report describes the overall approach to the test plan element, the goals, activities, how it will be organized and outlines the tester's needs that must be met in order to properly carry out the test. In analytical simulations the methods used to perform the analysis and specifics of the modeling program used will be clearly stated along with boundary conditions, physical properties under anticipated conditions, applied loads, sources, and references. During physical testing, the instrument accuracy and data quality used to indicate test conditions shall be specified. Such inputs and readings shall be of a suitable quality level for the performance of the particular role intended by the test objectives. Assumptions used shall be stated and unverified assumptions shall be listed that must be closed or resolved at a later point in the development task. Calculations will be accompanied by a standard Calculation Disclosure Statement (sample included). When a physical simulation or prototype test is involved, all aspects of the test article and the expected outcome shall be described. The approach plan shall also include parameters and details of the external factors that must be present, data to be acquisitioned, necessary instruments, monitors and calibrations, control systems, limiting devices, safety systems, and quality assurance provisions. Certifications that are necessary prior to performance of any physical tests shall be stated. Presence of compliant, pre-accepted, manufacturing certifications shall be confirmed prior to initiation of any physical tests.

In physical test cases, prior to test initiation, conditions that constitute cause for the test to be halted, aborted or suspended shall be noted. Safeguards shall be provided and described that ensure personnel are not at risk prior to, during, or following the test, and that test facilities, equipment or the test specimen is not damaged as a result of the test (If the particular element involves destructive testing, the expected outcome shall be accounted for). Anomalies or events that occur during the test that have not been anticipated prior to test initiation can also occur. Plans to confront any contingencies shall be prepared for in advance and described. This aspect is especially important where there is a potential for risk to personnel or test equipment. Resumption requirements shall also be stated that describe the conditions that are required to restart a suspended test. Aspects of this plan shall be reviewed by test personnel during test preparation and prior to test initiation.

Resource Needs: A detailed description of the necessary resources on the part of personnel, equipment, instruments, facilities, consumables and provisions shall be provided. The qualifications or level of training of personnel involved in the test or provision of test equipment must be stated, and how they will take part in the test must be described. Where quantifiable measurements are involved, it shall be specified in detail how the testing will be accomplished, who will perform the tests, where the test will be conducted, what will be tested and what facilities and testing instruments will be required. Additionally, the utilization of resources and the duration will be estimated and provided. Who will be obtaining the measurement and under what conditions, how the measurement will be obtained, and the quality level of the data will also be specified. Furthermore, how the test will be controlled, the range over which the test is expected to occur, the data needed to be obtained and the necessary accuracy will be specified. Where pertinent, safety aspects of the activity will be described. Typically, for simulations, resources will be limited to the software and computer hardware used. Rationale for the selections made in the test plan will be presented.

An estimated schedule will be presented in outline form that indicates when and where the test will be performed, what external factors, personnel or entities must be present, and provides milestones and a framework suitable for making logistical arrangements that must be prepared for in advance. Resource needs must be identified in such a way that ensures their provision at the test location in a timely manner.

Acceptance/Rejection criteria for the test shall be provided in advance of the test or analytical simulation. The criteria that signifies acceptance of the article shall be inclusive of all aspects that must simultaneously be achieved under the conditions stated. Rejections occur when one or more particular aspect/s do not meet pass-fail criteria under the test conditions. Criteria include the quality standards that must be met by the data acquired during the test, or by the software utilized.

A list of the specific roles and responsibilities that will be required on the part of the test participants, material or technology providers will be supplied for each test element. Participants shall have completed necessary training, have familiarity with test procedures, safety precautions and/or quality provisions, and shall be suitably qualified in advance of participation. Limiting conditions of operation (LCO) of test equipment shall include personnel that must be present during the test, including their roles prior, during and following the test, and shall include certified operators, control operators, safety and engineering personnel, data gatherers, observers, representatives and/or witnesses. Testing shall take place only with approved test apparatus and test articles. Necessary certifications shall accompany the acceptance of material used during the test. Certification must be performed by qualified personnel, and quality assurance and/or inspection data shall be provided using certified equipment operated by certified inspectors. Approval of LCO by the test director including certification data shall take place prior to test performance.

Test Results shall be acquired and documented during the performance of the test, and/or immediately following the test prior to influence from external factors outside the conditions of the test environment. Use of 'lab notebooks' or temporary data is acceptable, however in short order, while test conditions are still 'fresh' in the minds of the participants, that raw data will be translated into permanent format suitable for incorporation in the test results of the element test report. All relevant test data, environment and load conditions as well as the dated signature of the data taker is necessary to ensure data quality. Computer printouts and digital analytical data from measurements made from instruments likewise shall be simplified and reduced to contain information pertinent to the test and/or calibration procedure. How the data is used to formulate and describe the actual test results shall be clearly shown in a manner that other individuals, familiar with the technical subject, can decipher and easily follow. Conversions and data reduction calculations shall be checked and the engineering units of all numerical quantities shall be shown. Once test data is acquired it cannot be changed, although test results can change over several iterations of the test (i.e. a preliminary test does not necessarily indicate the final result). Follow-on testing shall be indicated by a unique test identifier (i.e. –dash number). Data from suspended tests may or may not be useful. Best practice would be not to discard such data until such time that its need is overcome by events that provide useful data along the lines of the intended test goals. A spreadsheet format workbook file shall be provided for each test plan element containing test data and data reductions. Comments and labels contained in the test result data describing how the data is consolidated shall accompany the data tables. A summarizing statement shall be supplied describing the test record, the quality of the test and data gathered. Any unexpected results, or external influences that may alter the quality of the data shall also be included.

When the test element is completed, the result summary provides a brief description of the test or simulation and the results. The result summary is intended to be of use toward making conclusions about the test, the results, the outcome and the path forward.

Conclusions: An element test report will be issued comparing the apparent result with the intended result, and the performance of the test article with respect to the design goals of the component or system. Conclusions may indicate acceptability unacceptability or undetermined acceptance of the test article, component or assembly. In all cases successful execution of the intended test procedure must take place in order to provide real and authentic conclusions. Review of the test conclusions by the test director and other responsible individuals is necessary. The test director shall indicate that the test execution was found successful. The degree to which test objectives are met should be stated and quantified to make clear the path to proceed.

Depending on the test results and conclusions, outcomes indicating the path forward will become apparent as the test plan is filled out (i.e. as individual test plan elements are completed). Important goals, for example, are go/no-go material selections, or what worked and what did not. Such information should be included in path forward recommendations. The path forward section of the test report should include recommendations based on the success or failure of the system or component to meet the intended objectives. A successful outcome to a successful test should clear the way to proceed to the next test plan element, however it that is not the case, and other aspects need to be made clear before proceeding or making a decision, then that too should be indicated.

Peer review of the test findings and recommendations is required. The entire test plan element data package should be made available for use by reviewers. It is important that the report be complete, correct, and consistent with the goals of the overall test plan.

Data Appendix: This is the repository of all important test information that is not contained within the body of the element test report. Should it become necessary either as a part of organizational review, review by an external or regulatory body, or as a part of some future review process, a complete file of all test data relevant to the test plan element will be provided with the report in an appendix. The appendix is to be organized with a table of contents and page count. All forms of references may be included in the appendix including drawings, sketches, pictures, interim results, preliminary revisions, hand calculations, vendor data, calibration records, raw data from tests and lab notebooks and dimensional or NDE shop inspection results. All data should be labeled for later understanding by persons that did not witness or take part in the test. Each sheet of all data records will likewise be labeled with the test plan identifier.

10.0 REFERENCES

[Bolin 2008] "NGNP IHX and Secondary Heat Transport Loop Alternatives Study," General Atomics Report 911119, Issue 0, April 2008

[Hanson 2007] "NGNP Umbrella Technology Development Plan," General Atomics Report PC-000543, Issue 0, July 2007

[HTGR-86025] "Design Data Needs – Modular High-Temperature Gas-Cooled Reactor," DOE-HTGR-86025 Revision 4, Issued by General Atomics of the Department of Energy, October, 1989.

[Labar 2008] "NGNP Steam Generator Alternatives Study," General Atomics Report 911120, Issue 0, April 2008

[PSER, 1986] "Draft Pre-application Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor (MHTGR), March 1989.

[PSER, 1996] "Draft Copy of Pre-application Safety Evaluation Report (PSER) on the Modular High-Temperature Gas-Cooled Reactor (MHTGR), February, 1996.

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