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

Technology Development Road Mapping Report for NGNP with 750°C Reactor Outlet Helium Temperature

Prepared by General Atomics
For the Battelle Energy Alliance, LLC

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EXECUTIVE SUMMARY

A decision was made by the NGNP Project in October 2008 to reduce the nominal reactor outlet helium temperature for the NGNP from 950°C into the range of 750°C to 800°C. This decision to reduce the reactor outlet helium temperature has a significant impact on the technology development effort required to support the NGNP. Specifically, 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 report presents the work that the General Atomics (GA) NGNP team has performed to revise the NGNP Technology Readiness Level (TRL) assessment presented in GA Report PC-000580 to reflect a reduction in the nominal reactor outlet helium temperature from 950°C to 750°C. The Technology Development Road Maps (TDRMs) in PC-000580 and the corresponding Test Plans were developed in 2008 for the NGNP configuration shown in Figure E-1. Based on the decision by the NGNP Project to lower the reactor outlet helium temperature into the range 750°C - 800°C, GA has selected the configuration shown in Figure E-2 as its reference NGNP configuration, which will be the point of departure for NGNP conceptual design.

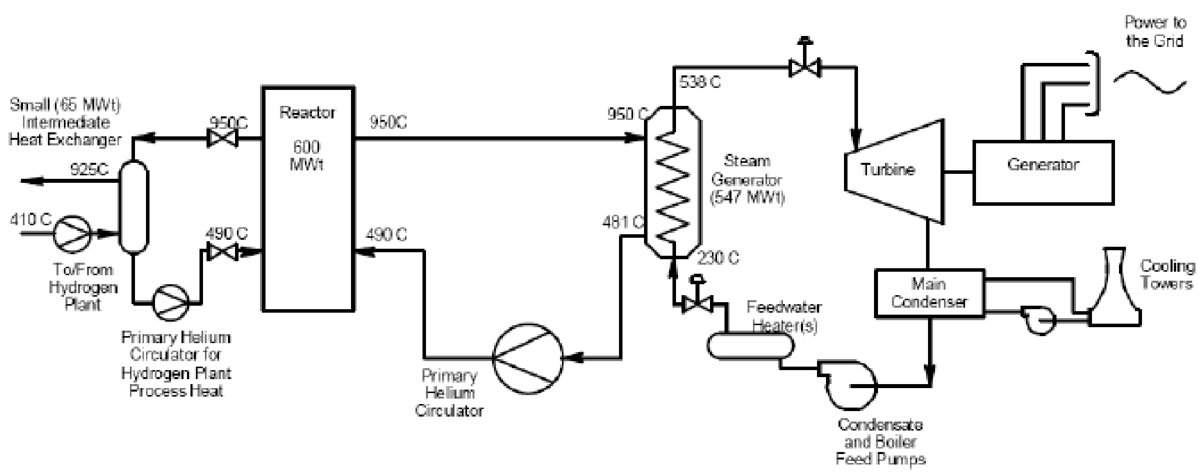

Figure E-1. NGNP Configuration for Technology Development Roadmapping in 2008

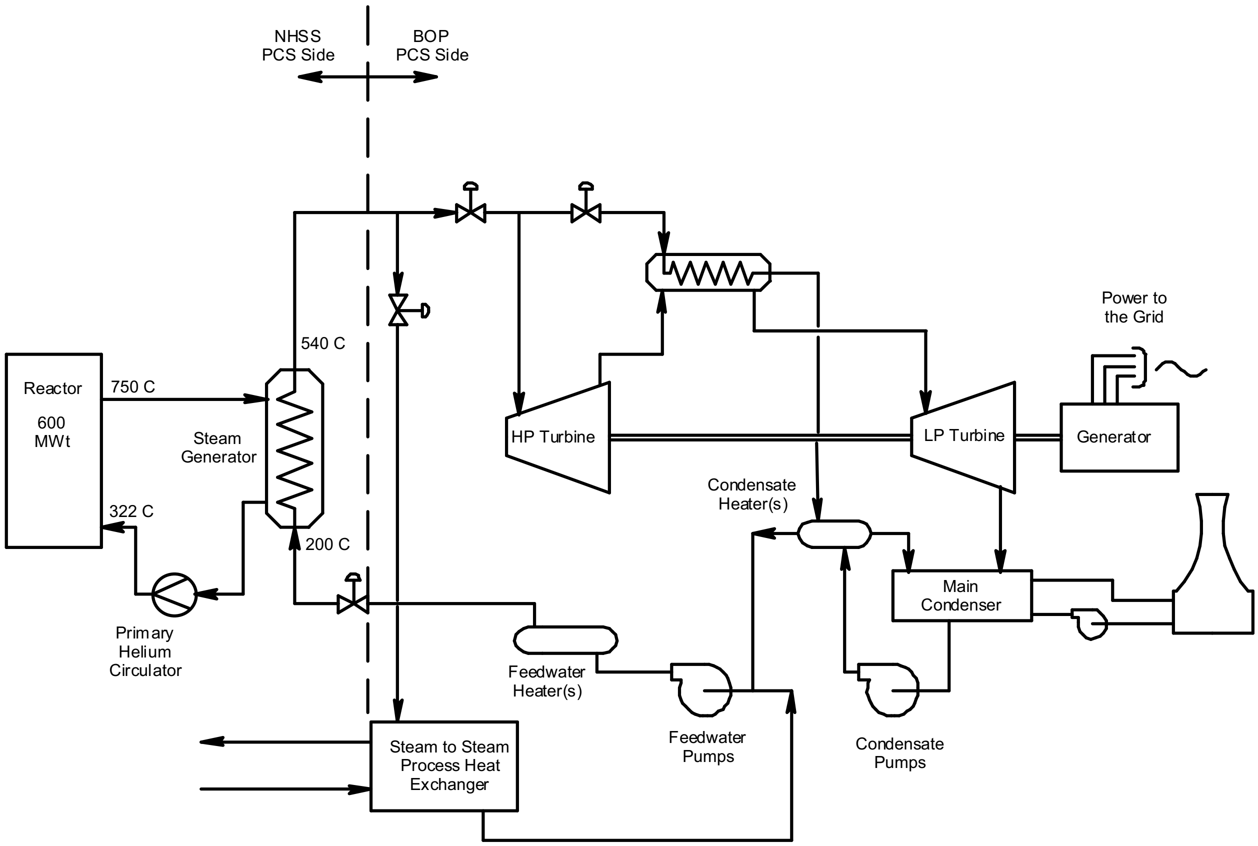

Figure E-2. Current NGNP Configuration for Technology Development Roadmapping

It is currently anticipated that the NGNP will be used to co-generate electricity and process heat in the form of steam, and that the steam will be provided to the end-user's facility via a tertiary loop. Heat will be transferred from an appropriate location (or locations) in the power conversion loop to the tertiary loop via a steam-to-steam heat exchanger. A steam-to-steam heat exchanger may also be needed in the power conversion loop to reheat the steam exiting the high pressure turbine as a means of improving power conversion cycle efficiency and reducing steam wetness at the back end of the cycle.

The purpose of the technology development roadmapping effort is to systematically define the current technology readiness level (TRL) for the critical systems, structures, and components (SSCs) and to define the activities necessary to advance the TRLs to the level required for installation and operation of the critical SSCs in the NGNP. The scope of the work to revise the TRL assessment presented in PC-000580 included the following subtasks:

- Revise the list of critical SSCs to reflect the lower reactor outlet helium temperature and revised reference NGNP configuration

- Revise the technology development road map (TDRM) and the supporting TRL rating sheets for each critical SSC for which the technology development requirements have changed due to the reduction in the reactor outlet helium temperature
- Assign an initial TRL rating for any newly identified critical SSCs and prepare TDRMs and the supporting TRL rating sheets
- For any SSCs for which the technology development requirements have changed due to the reduction in the reactor outlet helium temperature, modify the previously prepared Test Plan for the 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
- Prepare a final report that includes the TDRMs, the supporting TRL ratings sheets, and the integrated SSC test schedule

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. For the plant configuration shown in Figure E-2, GA has identified the following critical SSCs:

- Reactor control equipment
- Control rods
- Upper core restraint
- High temperature ducting (hot duct)
- Reactor core assembly
- Reactor graphite elements
- Reactor pressure vessel
- Helium circulators (PHTS and SCS)
- Shutdown cooling heat exchanger (SCHE)
- Reactor cavity cooling system (RCCS)
- Steam generator (SG)
- High temperature valves
- Fuel handling and storage system
- Primary circuit and balance of plant instrumentation
- RPS, IPS, and PCDIS
- Fuel

Table E-1 lists the initial TRL rating[1] that GA has assigned to each critical SSC and shows how the critical SSCs and TRL ratings have changed relative to those in PC-000580.

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

		Initial TRL Rating	
SSC #	**Critical SSC**	**Figure E-1**	**Figure E-2**
1	Reactor control equipment	4	4
2	Reactor internals		
	a1. Control rods – composite	2	2
	a2. Control rods – metallic	NA*	4
	b. Control rod and RSM guide tubes	2	4
	c. Metallic core support structure (insulation)	3	3
	d1. Upper core restraint - composite	2	2
	d2. Upper core restraint - metallic	NA	4
	e. Upper plenum shroud (thermal barrier)	2	4
3	Hot duct	2	2
4	Reactor core and graphite		
	a. Reactor core	5	5
	b. Graphite	6	6
5	Reactor pressure vessel	5	5
6	Helium circulator	6	6
7	Intermediate heat exchanger	2	NA
8	Shutdown cooling system heat exchanger	4	4
9	Reactor cavity cooling system	4	4
10	Steam Generator	3	4
11	Turbomachinery (for combined cycle PCS)	4	NA
12	High temp. isolation valves and pressure relief valves	3	4
13	S-I hydrogen production system	3	NA
14	Fuel handling and storage system	4	4
15	Primary circuit and BOP protection inst.	3	3
16	RPS, IPS, PCDIS	4	4
N/A	Fuel	4	4
*NA = Not applicable			

As indicated in Table E-1, three of the critical SSCs for the previous NGNP configuration shown in Figure E-1 are not critical SSCs for the new reference configuration shown in Figure E-2. These include the intermediate heat exchanger, the S-I hydrogen production system, and the turbomachine, none of which are needed for the new plant configuration. On the other hand, variations of two of the critical SSCs have been added. These include metallic control rods and

[1] See Table 2-1 for TRL definitions

metallic upper core restraint elements. These SSC variations were added because GA considers it likely that these components will have to be made initially from high-temperature metals while ceramic composite components are being developed and qualified as a future design improvement. This approach is considered necessary because it is considered unlikely that the ceramic composite components can be developed and qualified on a schedule that would make them available for an NGNP startup in 2021. However, It is anticipated that the parallel effort to develop ceramic composite control rods and upper core restraint elements will eventually lead to replacement of their metallic counterparts. Reflecting the need for less technology development for the metallic components, the initial TRL assigned to the metallic control rods and upper core restraint elements is 4 compared to 2 for the corresponding ceramic composite components.

With respect to changes in the initial TRL ratings, the initial TRL rating of the steam generator (3 to 4), the control rod and RSM guide tubes (2 to 4), and the upper plenum shroud (2 to 4) have been increased due to elimination of the need to develop and qualify high-temperature materials for these components. This is a direct benefit of the reduction in the reactor outlet helium temperature. Another component that is greatly impacted by the reduction in the reactor outlet helium temperature is the reactor pressure vessel because the lower reactor outlet temperature eliminates the need for a direct vessel cooling of the vessel. Although the initial TRL level assigned to the reactor pressure vessel was 5 for both the old and new reference NGNP configurations, the level of effort and cost associated with increasing the TRL from 5 to 8 is substantially reduced as a result of the reduction in the reactor outlet helium temperature.

As previously noted in PC-000580, 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 proposed in the current 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.

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ACRONYMS

AGR	Advanced Gas Reactor
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing and Materials
BEA	Battelle Energy Alliance
BOP	Balance of Plant
BWXT	Babcock & Wilcox Company
CCD	Conduction Cool Down (event)
CR	Control Rod
CTF	Component Test Facility
DDN	Design Data Need
DOE	U.S. Department of Energy
EHGA	Element Hoist and Grapple Assembly
EPRI	Electric Power Research Institute
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
MHR	Modular Helium Reactor
MHTGR	Modular HTGR
NCA	Neutron Control Assembly

NGNP	Next Generation Nuclear Plant
NP-MHTGR	New Production Modular HTGR
ORNL	Oak Ridge National Laboratory
PASSC	Plant, Areas, Systems, Subsystems, and Components
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
PPM	Performance Prediction Methodology
PRD	Power Range Detector
RC2-SSC	Reference Configuration 2 - System, Structure, and Components
RCE	Reactor Control Equipment
RCCS	Reactor Cavity Cooling System
ROM	Rough Order of Magnitude
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
SOW	Statement of Work
SRD	Source Range Detector
S-I	Sulfur-Iodine
SSC	System, Structure, and Components
TDRM	Technology Development Road Map
TRISO	TRI-material, ISOtropic (with the materials being low-density pyrocarbon, high-density pyrocarbon, and silicon carbide
TRL	Technology Readiness Level
UCO	Uranium Oxycarbide (fuel)
UCR	Upper Core Restraint
VCS	Vessel Cooling System

1 INTRODUCTION

1.1 Scope

A decision was made by the NGNP Project in October 2008 to reduce the nominal reactor outlet helium temperature for the NGNP from 950°C into the range of 750°C to 800°C. This decision to reduce the reactor outlet helium temperature has a significant impact on the technology development effort required to support the NGNP. Specifically, 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 report presents the work that the General Atomics (GA) NGNP team has performed to revise the NGNP Technology Readiness Level (TRL) assessment presented in GA Report PC-000580 [GA TDRM 2008] to reflect a reduction in the nominal reactor outlet helium temperature from 950°C to 750°C. The members of the GA NGNP team that participated in this task included GA and URS Washington Division (URS-WD). The work was performed under Amendment 1 to Release 4 of Subcontract 75309 with the Battelle Energy Alliance (BEA).

The purpose of the technology development roadmapping effort is to systematically define the current technology readiness level (TRL) for the critical NGNP systems, structures, and components (SSCs) and to define the activities necessary to advance the TRLs to the level required for installation and operation of the critical SSCs in the NGNP [TDRM 2009]. The scope of the work to revise the TRL assessment presented in [GA TDRM 2008] included the following subtasks:

- Revise the list of critical SSCs to reflect the lower reactor outlet helium temperature and revised reference NGNP configuration
- Revise the technology development road map (TDRM) and the supporting TRL rating sheets for each critical SSC for which the technology development requirements have changed due to the reduction in the reactor outlet helium temperature
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- Prepare an integrated technology development schedule that supports NGNP startup in 2021
- Prepare a final report that includes the TDRMs, the supporting TRL ratings sheets, and the integrated SSC test schedule

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 (with the exception of two test plans prepared by GA team member URS – Washington division, which are included as appendices). 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 HTGR Component Test Facility (CTF) that is currently planned to be built at the INL [INL 2008a] [INL 2007] to support the NGNP Project.

Because the NGNP design process is at a very early stage, adequate design details to precisely define design data needs (DDN) 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 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.

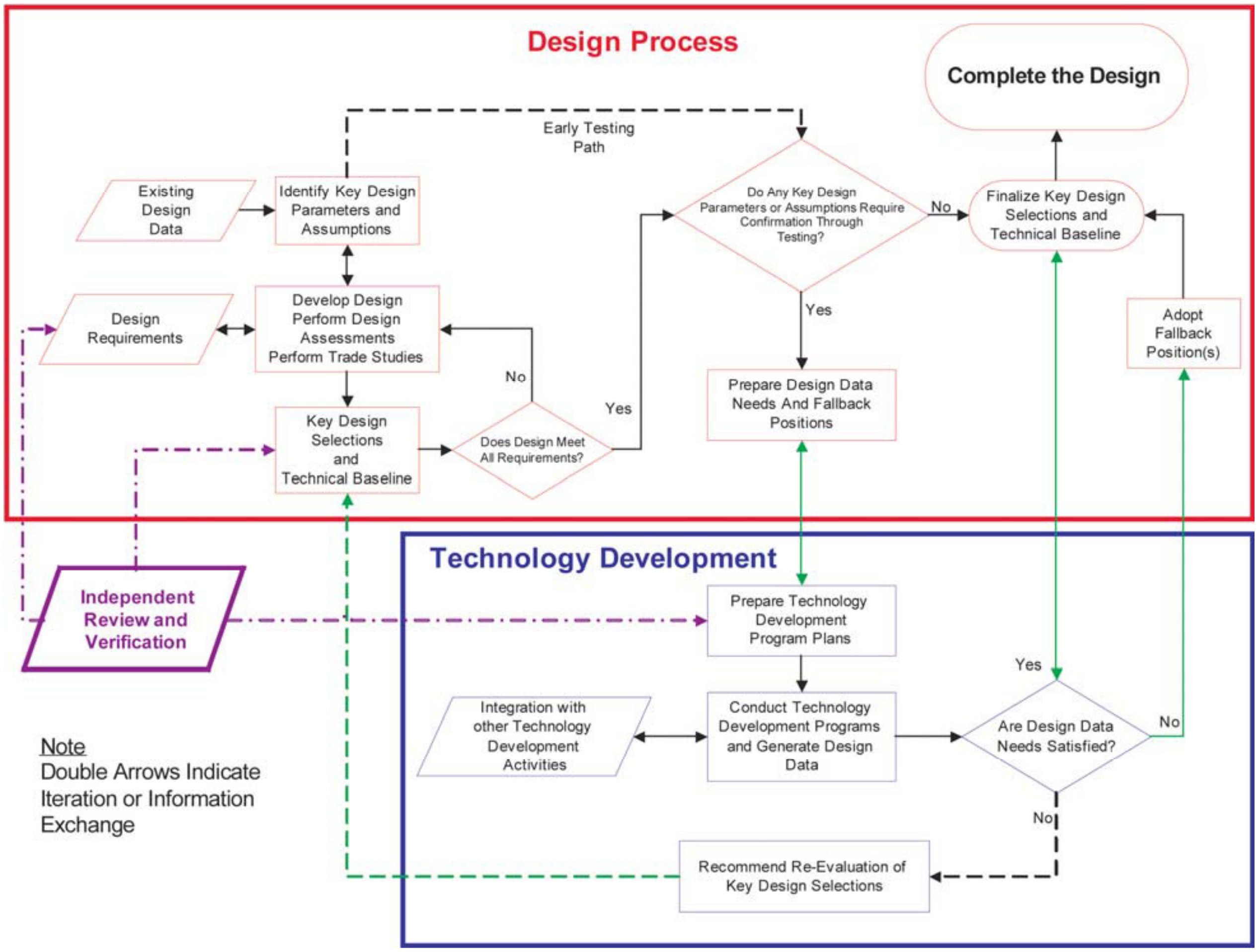


Figure 1-1. Approach for Integration of Design with Technology Development

1.2 Reference NGNP Configuration

The Technology Development Road Maps (TDRMs) in [GA TDRM 2008] and the corresponding Test Plans were developed in 2008 for the NGNP configuration shown in Figure 1-2. Based on the decision by the NGNP Project to lower the reactor outlet helium temperature into the range 750°C - 800°C, GA has selected the configuration shown in Figure 1-3 as its reference NGNP configuration, which will be the point of departure for NGNP conceptual design.

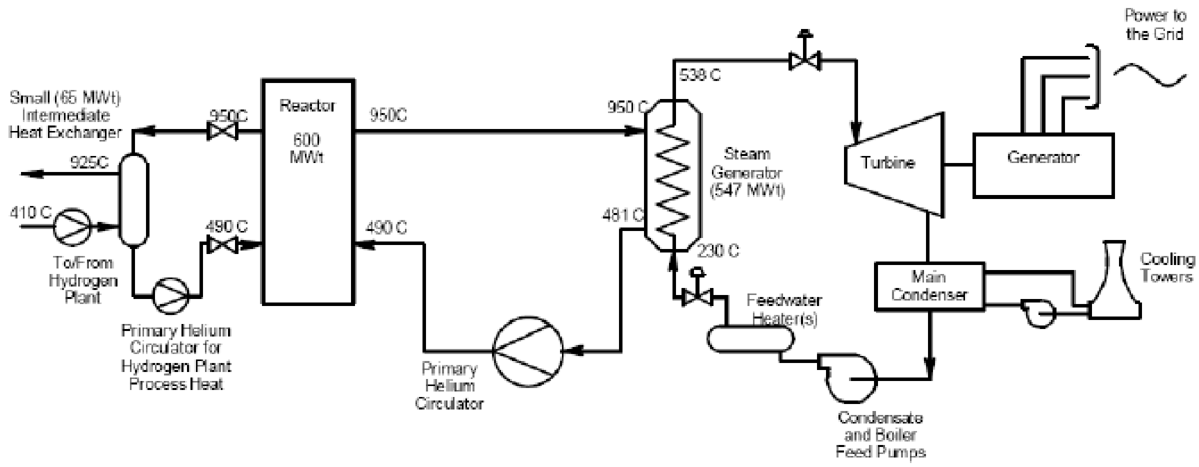


Figure 1-2. NGNP Configuration for Technology Development Roadmapping in 2008

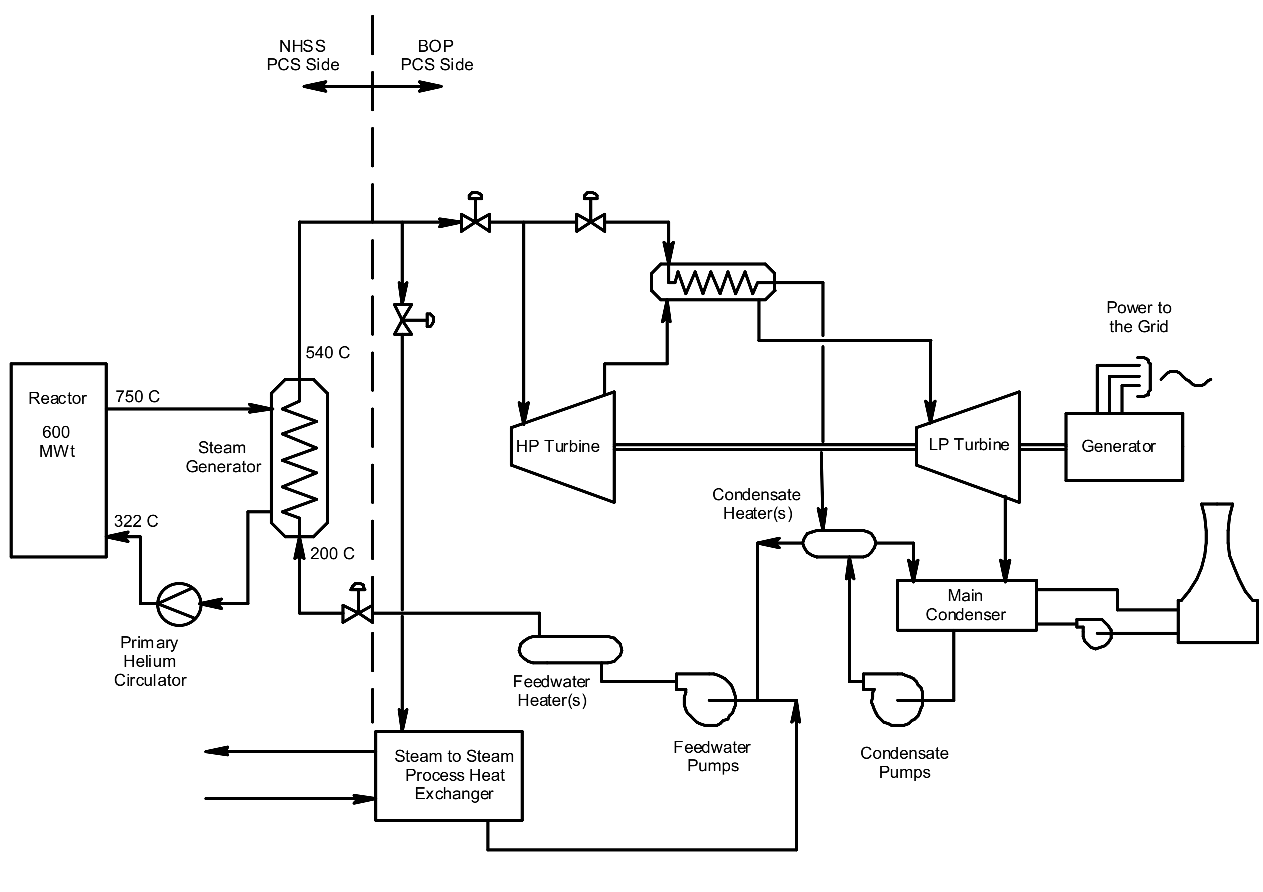


Figure 1-3. Current NGNP Configuration for Technology Development Roadmapping

This NGNP technology development road mapping effort covered in this report is based on the NGNP configuration shown in Figure 1-3. GA has selected this configuration as best meeting the process heat needs of potential end users of the HTGR technology based on the market survey performed by MPR Associates [MPR 2008]. As reflected by Figure 1-3, it is currently anticipated that the NGNP will be used to co-generate electricity and process heat in the form of steam, and that the steam will be provided to the end-user's facility via a tertiary loop. Heat will be transferred from an appropriate location (or locations) in the power conversion loop to the tertiary loop via a steam-to-steam heat exchanger. A steam-to-steam heat exchanger may also be needed in the power conversion loop to reheat the steam exiting the high pressure turbine as a means of improving power conversion cycle efficiency and reducing steam wetness at the back end of the cycle. This plant configuration is consistent with the high-level requirements specified in the latest revision of the NGNP Systems Requirements Manual [SRM 2009].

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 the primary heat transport loops will be helium, and for the secondary heat transport loop it will be steam.
- All vessels will be made out of LWR steel (i.e., SA-508/533). A vessel cooling system will not be necessary to control the reactor pressure vessel maximum temperatures below ASME code limits for SA-508/533

The SSCs and the current technology readiness levels for the SSCs are based on the NGNP configuration shown in Figure 1-3 and on the above assumptions, and the TDRMs reflect this NGNP configuration and these assumptions.

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 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 the initial technology development roadmapping task reported in [GA TDRM 2008].

- 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

Three of the above SSCs are not included in the new NGNP configuration shown in Figure 1-3. These include the IHX, the turbomachinery (for a combined-cycle PCS), and the S-I hydrogen production system. No new SSCs have been identified, but a technology development road map and the supporting TRL rating sheets have been prepared for the upper core restraint as part of the current work[2]. The new list of critical SSCs that have been addressed in this report is as follows.

- Reactor control equipment
- Reactor internals (control rods, upper core restraint)
- High temperature ducting (hot duct)
- Reactor core assembly
- Reactor graphite elements
- Reactor pressure vessel/reactor vessel cooling system
- Helium circulators (PHTS, SCS, SHTS)
- Shutdown cooling heat exchanger (SCHE)
- Reactor cavity cooling system (RCCS)
- Steam generator (SG)

[2] The UCR was also a critical SSC for the previous NGNP configuration, but preparation of a technology development roadmap for this component was not within the scope of the 2008 technology development road mapping task.

- High temperature valves
- Fuel handling and storage system
- Primary circuit and balance of plant instrumentation
- RPS, IPS, and PCDIS

Fuel, which is clearly a critical SSC for GA's NGNP design, was not addressed in the 2008 technology readiness assessment nor in the current assessment. This is because the NGNP/AGR Fuel Development and Qualification Program already has a detailed technical program plan [INL 2008d] (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 plant area, system, subsystem (structure), and components (PASSC). 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.

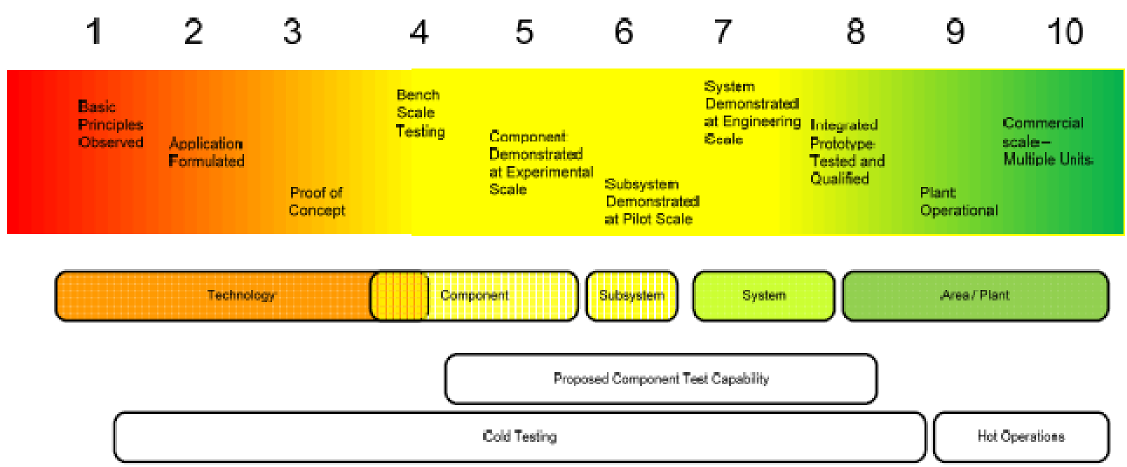


Figure 2-1. Comparison between TRL number, integration, and testing

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 statement of work (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. 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
Reactor pressure vessel	- 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 or metals)
High temperature ducting and insulation	- Material of construction for ducts - Type of insulation - Material of construction for cover plates

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 as part of the original NGNP road mapping task. 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.).

Several of the Test Plans identified in Table 4-1 of [GA TDRM 2008] remain applicable in their entirety to the new NGNP configuration, but some Test Plans required modification to delete technology development related to the higher reactor outlet helium temperature of the previous reference configuration. These Test Plans were modified as part of the current task.

3 BASELINE TRL RATINGS

Table 3-1 lists the initial TRL rating that GA has assigned to each critical SSC and shows how the critical SSCs and TRL ratings have changed relative to those in [GA TDRM 2008].

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

		Initial TRL Rating	
SSC #	**Critical SSC**	**Figure 1-2**	**Figure 1-3**
1	Reactor control equipment	4	4
2	Reactor internals		
	a1. Control rods – composite	2	2
	a2. Control rods – metallic	NA	4
	b. Control rod and RSM guide tubes	2	4
	c. Metallic core support structure (insulation)	3	3
	d1. Upper core restraint - composite	2	2
	d2. Upper core restraint - metallic	NA	4
	e. Upper plenum shroud (thermal barrier)	2	4
3	Hot duct	2	2
4	Reactor core and graphite		
	a. Reactor core	5	5
	b. Graphite	6	6
5	Reactor pressure vessel	5	5
6	Helium circulator	6	6
7	Intermediate heat exchanger	2	NA
8	Shutdown cooling system heat exchanger	4	4
9	Reactor cavity cooling system	4	4
10	Steam Generator	3	4
11	Turbomachinery (for combined cycle PCS)	4	NA
12	High temp. isolation valves and pressure relief valves	3	4
13	S-I hydrogen production system	3	NA
14	Fuel handling and storage system	4	4
15	Primary circuit and BOP protection inst.	3	3
16	RPS, IPS, PCDIS	4	4
N/A	Fuel	4	4

As indicated in Table 3-1, three of the critical SSCs for the previous NGNP configuration shown in Figure 1-2 are not critical SSCs for the new reference configuration shown in Figure 1-3. These include the intermediate heat exchanger, the S-I hydrogen production system, and the turbomachine, none of which are needed for the new plant configuration. On the other hand, variations of two of the critical SSCs have been added. These include metallic control rods and metallic upper core restraint elements. These SSC variations were added because GA considers it likely that these components will have to be made initially from high-temperature metals while ceramic composite components are being developed and qualified as a future design improvement [GA 2009]. This approach is considered necessary because it is considered unlikely that the ceramic composite components can be developed and qualified on a schedule that would make them available for an NGNP startup in 2021. However, It is anticipated that the parallel effort to develop ceramic composite control rods and upper core restraint elements will eventually lead to replacement of their metallic counterparts. Reflecting the need for less technology development for the metallic components, the initial TRL assigned to the metallic control rods and upper core restraint elements is 4 compared to 2 for the corresponding ceramic composite components.

With respect to changes in the initial TRL ratings, the initial TRL rating of the steam generator (3 to 4), the control rod and RSM guide tubes (2 to 4), and the upper plenum shroud (2 to 4) have been increased due to elimination of the need to develop and qualify high-temperature materials for these components. This is a direct benefit of the reduction in the reactor outlet helium temperature. Another component that is greatly impacted by the reduction in the reactor outlet helium temperature is the reactor pressure vessel because the lower reactor outlet temperature eliminates the need for a direct vessel cooling of the vessel. Although the initial TRL level assigned to the reactor pressure vessel was 5 for both the old and new reference NGNP configurations, the level of effort and cost associated with increasing the TRL from 5 to 8 is substantially reduced as a result of the reduction in the reactor outlet helium temperature.

As previously 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 [INL 2008d] 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

The TDRMs and TRL rating sheets developed for each critical SSC for the previous NGNP reference configuration were prepared using the methodology discussed in Section 2. For the SSC that remain critical SSC for the new NGNP reference configuration shown in Figure 1-3, the TDRMs and TRL rating sheets have been modified, as necessary, using the same methodology to reflect the new NGNP configuration. The critical SSC for the new reference configuration are identified as RC2-SSC-X (where RC2 stands for reference configuration 2 and X is the SSC number) to distinguish the TRL rating sheets and TDRMs for the new reference configuration from the TRL ratings sheets and TDRMs for the previous reference configuration. The SSC numbering for the previous configuration has been retained to avoid confusion.

A complete set of TRL rating sheets and TDRMs for the critical SSC in the new reference NGNP configuration 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. Several of the Test Plans prepared for the SSC in the old reference configuration in 2008 remain applicable in their entirety to the new NGNP configuration, but some Test Plans required modification to delete technology development related to the higher reactor outlet helium temperature of the previous reference configuration. These Test Plans were modified as part of the current task.

Table 4-1 identifies the Test Plans applicable to the new reference NGNP configuration shown in Figure 1-3. As indicated in Table 4-1, new Test Plans were prepared for the upper core restraint, the reactor pressure vessel, the steam generator, the hot duct, and the high-temperature valves.

Table 4-1. Test Plans for NGNP Critical SSC

RC2-SSC #	SSC Description	Originating Org.	Report #
1	Reactor control equipment	GA	911133
2a	Control rods (composites)	GA	911134
2d	Upper Core Restraint (composites)	GA	911172*
3	Hot duct	URS-WD	911177*
4a	Reactor core assembly	GA	911135
4b	Graphite elements	GA	911136
5	Reactor pressure vessel	GA	911173*
6	Helium circulator	GA	911138
8	Shutdown cooling heat exchanger	GA	911140
9	Reactor cavity cooling system	GA	911141
10	Steam generator	GA	911174*
12	High-temperature valves	URS-WD	911178*
14	Fuel handling and storage system	GA	911145
15	Primary circuit and BOP instrumentation	GA	911146
16	RPS, IPS, and PCDIS	GA	911147
* New Test Plans prepared for NGNP configuration having a reactor outlet helium temperature of 750°C			

4.1 RC2-SSC-1 Reactor Control Equipment

TRL Rating Sheets, TRL 4 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor: GA	**Document Number:** RC2-SSC-1.1	**Revision:** 0

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

Title: Reactor Control and Protection, Reactor Control Equipment

Description:
This SSC 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. It 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 ☐ PCS ☐ BOP

PASSC:	**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 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. The testing at the component level requires interaction with all these development efforts. Further 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 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	GA Vendor(s) Facility	CD 12-36mo CD 12-24mo CD 12-24mo	300 500 700

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 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. Support Structures, Movement Guidance Structures, Pressure Seals, Insulation, and Shielding: 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, 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.	GA Vendor(s) Facility	CD 24-36mo CD 12-36mo CD 12-36mo	500 1,000 1,500

TRL Rating Sheet

Vendor: GA	**Document Number:** RC2-SSC-1.2	**Revision:** 0

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

Title: Reactor Control and Protection, Reactor Control Equipment

Description:
For this SSC it 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. This SSC 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 ☐ PCS ☐ BOP

PASSC:	**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 is based on completion of previous TRL 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. The testing at the subsystem level requires interaction with all these development efforts. Further 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 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	GA Vendor(s) Facility	FD 12-30mo FD 24-30mo FD 12-30mo	300 500 700

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 this 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:**	RC2-SSC-1.3	**Revision:**	0

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

Title: Reactor Control and Protection, Reactor Control Equipment

Description:
For this SSC it 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. It 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 ☐ PCS ☐ BOP

PASSC:	**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 this SSC is based on the 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. The testing at the subsystem level requires interaction with all these development efforts. Further 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 this SSC. Related handling equipment will also be evaluated at level 7. Seismic testing for safety-related qualification of the 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 installation readiness process.

All 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 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 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	GA Facility	FD 54-60mo FD 54-72mo	300 1,000

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 the equipment. <u>Seismic Testing</u>: Seismic testing of the 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 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.	GA Facility Fabricators Seismic	FD 54-84mo FD 54-78mo FD 54-78mo FD 78-84mo	500 4,000 1,000 3,000

TRL Rating Sheet

Vendor: GA	**Document Number:** RC2-SSC-1.4	**Revision:** 1

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

Title: Reactor Control and Protection, Reactor Control Equipment

Description:

For this SSC it 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. It 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 ☐ PCS ☐ BOP

PASSC:	**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 technical rating for this SSC is based on the completion of activities required to achieve a level 7 technical rating. This work included testing of SRD, IFMU, Power Range neutron detectors, NCA, and NCA/RSCE prototype systems. These systems are contained in Reactor Control and Protection systems 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. Install NCA and NCA/RSCE equipment and perform hot-startup readiness tests at sub-power conditions. All testing, which is done after fuel installation, must be completed within minimum negative reactivity limitations at cold reactor conditions. (Cont.)	GA	FD 85-108mo	1,500

DDN(s) Supported: none	**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. The testing at the subsystem level requires interaction with all these development efforts. Further 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.

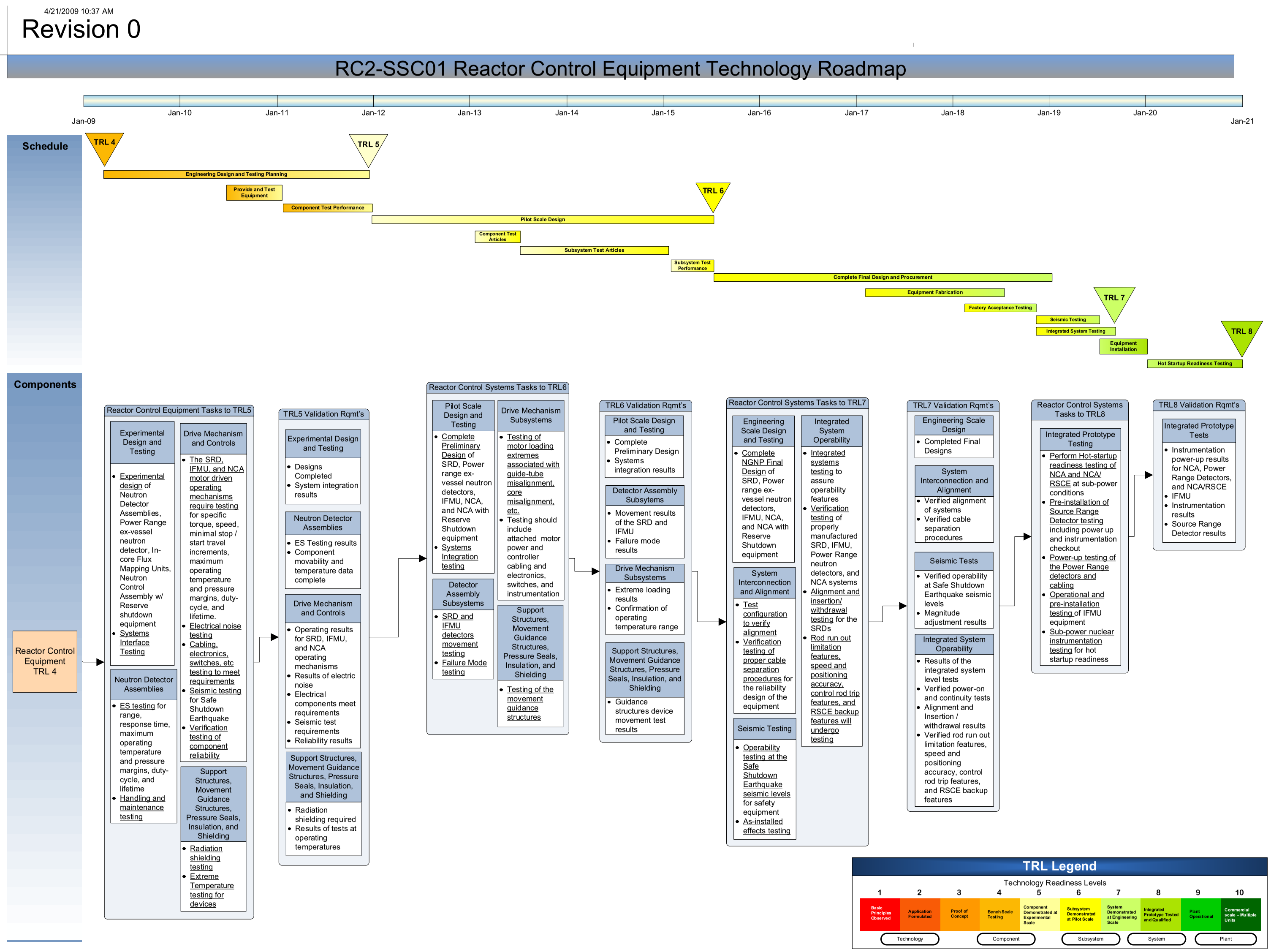
Additional Basis Sheet(s)

Basis:

Verification of NCA rod runout limitation features, control rod trip features, power cable and channel separation features. etc. were tested at the prototype level, and the required equipment adjustments were completed, documented, and incorporated into the equipment. Prominent subsystems -- neutron detector assembly subsystems, drive mechanism subsystems, and movement guidance subsystems – were also checked. The SRD neutron detectors, which operate through the lower vessel, were checked for alignment, retrieval features, etc. The IFMUs also contain movable equipment, and were checked at the prototype level. The handling equipment for all systems was checked. Seismic testing for safety-related qualification of the equipment was also completed at level 7.

Additional Action Sheet(s)

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Install Source Range Detector equipment and repeat pre-installation test procedure, including power-up and instrumentation checkouts, using PCDIS.	GA Fabricators	FD 85-108mo FD 85-108mo	800 200
3. Install Power Range detectors and cabling. Complete power-up checkout using PCDIS.	GA Fabricators	FD 85-108mo FD 85-108mo	800 200
4. Install IFMU equipment and repeat pre-installation checkout. Add operational testing deferred to level 8. Perform checkout testing with PCDIS.	GA Fabricators	FD 85-108mo FD 85-108mo	800 200
5. Confirm all sub-power nuclear instrumentation testing completion for hot startup readiness. Confirm source-level range and accuracy of SRD equipment, as part of checkout. Document.	GA	FD 96-102mo	500



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Revision 0
RC2-SSC01 Reactor Control Equipment Technology Roadmap
Jan-09 Jan-10 Jan-11 Jan-12 Jan-13 Jan-14 Jan-15 Jan-16 Jan-17 Jan-18 Jan-19 Jan-20 Jan-21
Schedule
TRL 4
TRL 5
TRL 6
TRL 7
TRL 8
Engineering Design and Testing Planning
Provide and Test Equipment
Component Test Performance
Pilot Scale Design
Component Test Articles
Subsystem Test Articles
Subsystem Test Performance
Complete Final Design and Procurement
Equipment Fabrication
Factory Acceptance Testing
Seismic Testing
Integrated System Testing
Equipment Installation
Hot Startup Readiness Testing
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 integration 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
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
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
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
TRL Legend
Technology Readiness Levels
1 2 3 4 5 6 7 8 9 10
Basic Principles Observed
Application Formulated
Proof of Concept
Bench Scale Testing
Component Demonstrated at Experimental Scale
Subsystem Demonstrated at Pilot Scale
System Demonstrated at Engineering Scale
Integrated Prototype Tested and Qualified
Plant Operational
Commercial scale - Multiple Units
Technology
Component
Subsystem
System
Plant

4.2 RC2-SSC-2a Control Rods and RC2-SSC-2d Upper Core Restraint

Control Rods (Composite), TRL Rating Sheets, TRL 2 through 7

Control Rods (Metallic), TRL Rating Sheets, TRL 4 through 7

Upper Core Restraint (Composite), TRL Rating Sheets, TRL 2 through 7

Upper Core Restraint (Metallic), TRL Rating Sheets, TRL 4 through 7

Technology Development Road Maps

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-2a.1-c	**Revision:**	0

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

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

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 ☐ PCS ☐ BOP

PASSC:	**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 ~1400°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:**	RC2-SSC-2a.2-c	**Revision:**	0

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

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

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 ☐ PCS ☐ BOP

PASSC:	**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 ~1400°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:**	RC2-SSC-2a.3-c	**Revision:**	0

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

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

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 ☐ PCS ☐ BOP

PASSC:	**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:** RC2-SSC-2a.4-c	**Revision:** 0

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

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

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 ☐ PCS ☐ BOP

PASSC:	**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:**	RC2-SSC-2a.5-c	**Revision:**	0

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

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

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 ☐ PCS ☐ BOP

PASSC:	**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

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-2a.6-c	**Revision:**	0

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

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

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 ☐ PCS ☐ BOP

PASSC:	**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) ☐

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.

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 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.	GA and NGNP operator	Must be completed ~3 months prior to installation of NCAs and CRs in NGNP	200 (incremental cost for CR testing)

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:** RC2-SSC-2a.1-m	**Revision:** 0

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

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

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 ☐ PCS ☐ BOP

PASSC:	**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	**3**	**4**	**5**

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

As discussed in GA report 911175, GA has concluded that Hastelloy XR should be used initially as the material of construction for the NGNP CR metallic parts (with a conversion to ceramic composites at a later time). Hastelloy XR steel has a limited database in the United States as the material has been extensively developed in Japan for the HTTR, and it has not been codified in Section III of the ASME code. It is anticipated that a small materials development program will be needed to supplement the data and fill in any gaps that are present.

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.11.02.10, N.11.02.11, N.11.02.16	**Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein, Russ Vollman, Jessie Crozier

Date: 4-20-09	**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.	GA	3 months starting after completion of action 1	200
3. Conduct baseline physical and mechanical properties tests on material test specimens. The number of tests performed shall be adequate to obtain the data required for the statistically significant engineering database or supplement an existing database required for the material performance models	INL, ORNL, JAEA, CEA, KAERI, and/or commercial laboratories	6 months	500
4. Conduct irradiation tests to determine irradiation induced dimensional changes and creep and to determine the effect of irradiation on the baseline physical and mechanical properties (from action 3). The testing shall be sufficient to establish a statistically significant engineering data base or supplement an existing database for the material performance models	INL, JAEA, CEA, KAERI, and/or ORNL	2 years, must be completed by ~mid 2013	~2000
5. Conduct corrosion tests in a reactor helium environment at reactor operating temperatures (up to ~1400°C) to determine the effects on the baseline physical and mechanical properties, including emissivity data, (from action 3). The testing shall be sufficient to establish a statistically significant engineering data base or supplement an existing database for the material performance models	INL, JAEA, CEA, KAERI, and/or ORNL	1 year, must be completed by ~mid 2013	750
6. Complete updating material behavior and failure models based on the data from actions 3, 4, and 5	GA	6 months starting as soon as data from actions 3, 4, and 5 are available	400

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-2a.2-m	**Revision:**	0

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

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

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 ☐ PCS ☐ BOP

PASSC:	**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 when the following conditions are met: (1) Adequate physical and materials properties testing of unirradiated, irradiated, and corrosion of Hastelloy XR parts have been performed to establish a statistically significant material properties engineering data base; and (2) Material behavior and failure models have been updated based on the 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. 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, Jessie Crozier

Date: 4-20-09	**Originating Organization:** General Atomics

Additional Actions Sheets(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. 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

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-2a.3-m	**Revision:**	0

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

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

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 ☐ PCS ☐ BOP

PASSC:	**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 1) 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, and 2) 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 2019 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: 4-20-09 **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 2019 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 2019 prior to fab. of CRs for NGNP	1900

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-2a.4-m	**Revision:**	0

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

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

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 ☐ PCS ☐ BOP

PASSC:	**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) ☐

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.

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 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.	GA and NGNP operator	Must be completed ~3 months prior to installation of NCAs and CRs in NGNP	200 (incremental cost for CR testing)

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

Subject Matter Expert Making Determination: John Saurwein, Russ Vollman

Date: 4-20-09	**Originating Organization:** General Atomics

TRL Rating Sheet

Vendor: GA	**Document Number:** RC2-SSC-2d.1-c	**Revision:** 0

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

Title: Reactor Internals - Upper Core Restraint (UCR) - Composite

Description:

The UCR consists of interconnecting composite plenum elements which are located on the top layer of the reflector and permanent side reflector graphite blocks. All elements have hexagonal shapes except for those above the permanent side reflectors, which match the configuration of the permanent side reflectors. The UCR provides lateral restraint and stability to the core array, provides neutron flux attenuation to the reactor internals and reactor vessel, channels helium primary coolant from the upper plenum to the entrance of the reactor core, and maintains the capability to refuel the reactor core. In addition, the UCR provides channels to insert control rods, reserve shutdown pellets, and the In-Core Flux Mapping Unit (IFMU). The UCR assembly provides core stability and limits core movements. The plenum elements are keyed together to limit relative lateral motion within the UCR assembly. Connections between the core barrel and the plenum elements over the permanent side reflectors limit the translation and rotation of the UCR assembly with respect to the core barrel.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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 UCR design will be essentially the same as the one designed in the NP-MHTGR, but the higher temperatures to which the UCR will be subjected during conduction cooldown events in the NGNP may require use of a ceramic composite as the material of construction. 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 UCR 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 upper core restraint operating conditions (e.g., temperatures, flow conditions, helium impurities, etc.) and develop upper core restraint requirements	General Atomics (GA)	6 months starting near beginning of CD	350

DDN(s) Supported: N.11.02.25, N.11.02.26, N.11.02.27, and N.11.02.28,	**Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein, Russ Vollman, Jessie Crozier

Date: 04-20-09	**Originating Organization:** General Atomics

Additional Actions Sheets(s)

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Develop upper core restraint conceptual design. Perform FEA and seismic analyses to calculate the expected mechanical and seismic loads. Initiate development of UCR 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 UCR elements. 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 ~1000°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:** RC2-SSC-2d.2-c	**Revision:** 0

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

Title: Reactor Internals - Upper Core Restraint (UCR) - Composite

Description:

The UCR consists of interconnecting composite plenum elements which are located on the top layer of the reflector and permanent side reflector graphite blocks. All elements have hexagonal shapes except for those above the permanent side reflectors, which match the configuration of the permanent side reflectors. The UCR provides lateral restraint and stability to the core array, provides neutron flux attenuation to the reactor internals and reactor vessel, channels helium primary coolant from the upper plenum to the entrance of the reactor core, and maintains the capability to refuel the reactor core. In addition, the UCR provides channels to insert control rods, reserve shutdown pellets, and the In-Core Flux Mapping Unit (IFMU). The UCR assembly provides core stability and limits core movements. The plenum elements are keyed together to limit relative lateral motion within the UCR assembly. Connections between the core barrel and the plenum elements over the permanent side reflectors limit the translation and rotation of the UCR assembly with respect to the core barrel.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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 UCR elements fabricated from these materials have a high probability of satisfying UCR 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 UCR elements. 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.11.02.25, N.11.02.26, N.11.02.27, and N.11.02.28,	**Technology Case File:**
Subject Matter Expert Making Determination: John Saurwein, Russ Vollman, Jessie Crozier	
Date: 04-20-09	**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 ~1000°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:**	RC2-SSC-2d.3-c	**Revision:**	0

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

Title: Reactor Internals - Upper Core Restraint (UCR) - Composite

Description:

The UCR consists of interconnecting composite plenum elements which are located on the top layer of the reflector and permanent side reflector graphite blocks. All elements have hexagonal shapes except for those above the permanent side reflectors, which match the configuration of the permanent side reflectors. The UCR provides lateral restraint and stability to the core array, provides neutron flux attenuation to the reactor internals and reactor vessel, channels helium primary coolant from the upper plenum to the entrance of the reactor core, and maintains the capability to refuel the reactor core. In addition, the UCR provides channels to insert control rods, reserve shutdown pellets, and the In-Core Flux Mapping Unit (IFMU). The UCR assembly provides core stability and limits core movements. The plenum elements are keyed together to limit relative lateral motion within the UCR assembly. Connections between the core barrel and the plenum elements over the permanent side reflectors limit the translation and rotation of the UCR assembly with respect to the core barrel.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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 UCR elements 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:	**Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein, Russ Vollman, Jessie Crozier

Date: 03-10-09 **Originating Organization:** General Atomics

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-2d.4-c	**Revision:**	0

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

Title: Reactor Internals - Upper Core Restraint (UCR) - Composite

Description:

The UCR consists of interconnecting composite plenum elements which are located on the top layer of the reflector and permanent side reflector graphite blocks. All elements have hexagonal shapes except for those above the permanent side reflectors, which match the configuration of the permanent side reflectors. The UCR provides lateral restraint and stability to the core array, provides neutron flux attenuation to the reactor internals and reactor vessel, channels helium primary coolant from the upper plenum to the entrance of the reactor core, and maintains the capability to refuel the reactor core. In addition, the UCR provides channels to insert control rods, reserve shutdown pellets, and the In-Core Flux Mapping Unit (IFMU). The UCR assembly provides core stability and limits core movements. The plenum elements are keyed together to limit relative lateral motion within the UCR assembly. Connections between the core barrel and the plenum elements over the permanent side reflectors limit the translation and rotation of the UCR assembly with respect to the core barrel.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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 upper core restraint 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)**
1. Engineering analyses will be performed to show fluid flow, freedom of movement for different thermal expansions, and seismic load / movement requirements. This analysis will show analytically how the UCR elements perform during normal differential thermal expansion, dynamic fluid flow, seismic shaking, and movement from graphite columns.	GA	6 months. Must be complete about 1.5 years before end of final design	500

DDN(s) Supported:	**Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein, Russ Vollman, Jessie Crozier

Date: 04-20-09 **Originating Organization:** General Atomics

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Engineering analyses will be performed 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. This analysis will show analytically how the CR and RSM guide tubes respond during normal differential thermal expansion, dynamic fluid flow, seismic shaking, and non-normal events.	GA,	6 months. Must be complete about 1.5 years before end of final design	200

TRL Rating Sheet

Vendor: GA	**Document Number:** RC2-SSC-2d.5-c	**Revision:** 0

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

Title: Reactor Internals - Upper Core Restraint (UCR) - Composite

Description:

The UCR consists of interconnecting composite plenum elements which are located on the top layer of the reflector and permanent side reflector graphite blocks. All elements have hexagonal shapes except for those above the permanent side reflectors, which match the configuration of the permanent side reflectors. The UCR provides lateral restraint and stability to the core array, provides neutron flux attenuation to the reactor internals and reactor vessel, channels helium primary coolant from the upper plenum to the entrance of the reactor core, and maintains the capability to refuel the reactor core. In addition, the UCR provides channels to insert control rods, reserve shutdown pellets, and the In-Core Flux Mapping Unit (IFMU). The UCR assembly provides core stability and limits core movements. The plenum elements are keyed together to limit relative lateral motion within the UCR assembly. Connections between the core barrel and the plenum elements over the permanent side reflectors limit the translation and rotation of the UCR assembly with respect to the core barrel.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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 shows that the composite UCR elements meet the design and safety requirements. In addition, the UCR elements should be demonstrated to be capable of having the control rods 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.

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 full-scale UCR array with guide tubes, and above columns 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	1500

DDN(s) Supported: C.11.03.44	**Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein, Russ Vollman, Jessie Crozier

Date: 04-20-09	**Originating Organization:** General Atomics

Additional Action Sheet(s)

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Conduct UCR structural integrity testing. A full-size UCR array 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	1000
3. Conduct UCR flow distribution and pressure drop testing. A full-size UCR array will be subjected to operational and accident-condition loads and temperatures to quantify pressure drop coefficients and coolant hole flow distributions as function of the Reynolds number.	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	1000

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-2d.6-c	**Revision:**	0

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

Title: Reactor Internals - Upper Core Restraint (UCR) - Composite

Description:

The UCR consists of interconnecting composite plenum elements which are located on the top layer of the reflector and permanent side reflector graphite blocks. All elements have hexagonal shapes except for those above the permanent side reflectors, which match the configuration of the permanent side reflectors. The UCR provides lateral restraint and stability to the core array, provides neutron flux attenuation to the reactor internals and reactor vessel, channels helium primary coolant from the upper plenum to the entrance of the reactor core, and maintains the capability to refuel the reactor core. In addition, the UCR provides channels to insert control rods, reserve shutdown pellets, and the In-Core Flux Mapping Unit (IFMU). The UCR assembly provides core stability and limits core movements. The plenum elements are keyed together to limit relative lateral motion within the UCR assembly. Connections between the core barrel and the plenum elements over the permanent side reflectors limit the translation and rotation of the UCR assembly with respect to the core barrel.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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) ☐

TRL 7 is achieved when the following conditions are met: (1) UCR flow-induced and seismic vibration testing has been completed and the results confirm the position of the UCR elements and that there is no impact on the capability to insert or withdraw the CRs in the reactor, (2) UCR structural integrity testing has been completed and the results confirm that the UCR design has adequate margin against operational failure, and (3) UCR testing of pressure drop and flow distribution through the UCR elements confirms with limits.

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 at the NGNP to verify integration of the UCR interfaces with other components as part of the integrated testing of the NCA, RSM, RSM guide tubes, CR guide tube, and CR. The test should include both normal and off-normal events. Position measurements and visual inspection are required.	GA and NGNP operator	Must be completed ~3 months prior to installation of NCAs and CRs in NGNP	1500

DDN(s) Supported:	**Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein, Russ Vollman, Jessie Crozier

Date: 03-10-09	**Originating Organization:** General Atomics

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-2d.1-m	**Revision:**	0

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

Title: Reactor Internals - Upper Core Restraint (UCR) - Metallic

Description:

The UCR consists of interconnecting metallic plenum elements which are located on the top layer of the reflector and permanent side reflector graphite blocks. All elements have hexagonal shapes except for those above the permanent side reflectors, which match the configuration of the permanent side reflectors. The UCR provides lateral restraint and stability to the core array, provides neutron flux attenuation to the reactor internals and reactor vessel, channels helium primary coolant from the upper plenum to the entrance of the reactor core, and maintains the capability to refuel the reactor core. In addition, the UCR provides channels to insert control rods, reserve shutdown pellets, and the In-Core Flux Mapping Unit (IFMU). The UCR assembly provides core stability and limits core movements. The plenum elements are keyed together to limit relative lateral motion within the UCR assembly. Connections between the core barrel and the plenum elements over the permanent side reflectors limit the translation and rotation of the UCR assembly with respect to the core barrel.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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	**3**	**4**	**5**

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

As discussed in GA report 911175, GA has concluded that Hastelloy XR should be used as the material of construction for the NGNP UCR elements (with a conversion to ceramic composites at a later time). Hastelloy XR steel has a limited database in the United States as the material has been extensively developed in Japan for the HTTR, and it has not been codified in Section III of the ASME code. It is anticipated that a small materials development program will be needed to supplement the data and fill in any gaps that are present.

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 upper core restraint operating conditions (e.g., temperatures, flow conditions, helium impurities, etc.) and develop upper core restraint requirements	General Atomics (GA)	6 months starting near beginning of CD	350

DDN(s) Supported: N.11.02.10, N.11.02.11, and N.11.02.16	**Technology Case File:**
Subject Matter Expert Making Determination:	John Saurwein, Russ Vollman, Jessie Crozier
Date: 4-20-09	**Originating Organization:** General Atomics

Additional Action Sheet(s):

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Develop upper core restraint conceptual design. Perform FEA and seismic analyses to calculate the expected mechanical and seismic loads. Initiate development of UCR performance models	GA	3 months starting after completion of action 1	200
3. Conduct physical and mechanical properties testing on Hastelloy XR specimens. The number of tests performed shall be adequate to obtain the data required for the statistically significant engineering database or supplement an existing database required for the material performance models	INL, ORNL, JAEA, CEA, KAERI, and/or commercial laboratories	1 year	1000
4. Conduct irradiation tests to determine irradiation induced dimensional changes and creep and to determine the effect of irradiation on the baseline physical and mechanical properties on Hastelloy XR (from action 3). The testing shall be sufficient to establish a statistically significant engineering database or supplement an existing database required for the material performance models	INL, JAEA, CEA, KAERI, and/or ORNL	2 years, must be completed by ~mid 2013	~2000
5. Conduct corrosion tests in a reactor helium environment at reactor operating temperatures (up to ~1000°C) to determine the effects on the baseline physical and mechanical properties, including emissivity, on Hastelloy XR (from action 3) The testing shall be sufficient to establish a statistically significant engineering database or supplement an existing database required for the material performance models	INL, JAEA, CEA, KAERI, and/or ORNL	2 years, must be completed by ~mid 2013	500
6. Complete updating metal behavior and failure models based on the data from actions 3, 4, and 5	GA	6 months starting as soon as data from actions 3, 4, and 5 are available	200

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-2d.2-m	**Revision:**	0

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

Title: Reactor Internals - Upper Core Restraint (UCR) - Metallic

Description:

The UCR consists of interconnecting metallic plenum elements which are located on the top layer of the reflector and permanent side reflector graphite blocks. All elements have hexagonal shapes except for those above the permanent side reflectors, which match the configuration of the permanent side reflectors. The UCR provides lateral restraint and stability to the core array, provides neutron flux attenuation to the reactor internals and reactor vessel, channels helium primary coolant from the upper plenum to the entrance of the reactor core, and maintains the capability to refuel the reactor core. In addition, the UCR provides channels to insert control rods, reserve shutdown pellets, and the In-Core Flux Mapping Unit (IFMU). The UCR assembly provides core stability and limits core movements. The plenum elements are keyed together to limit relative lateral motion within the UCR assembly. Connections between the core barrel and the plenum elements over the permanent side reflectors limit the translation and rotation of the UCR assembly with respect to the core barrel.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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	**4**	**5**	**6**

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

TRL 5 is achieved when the following conditions are met for the Hastelloy XR UCR elements: (1) 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 (2) Material behavior and failure models have been completed based on the 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)**
1. 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. This analysis will show analytically how the UCR elements perform during normal differential thermal expansion, dynamic fluid flow, seismic shaking, and movement from graphite columns.	General Atomics	12 months starting about half-way through FD	1000

DDN(s) Supported:	**Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein, Russ Vollman, Jessie Crozier

Date: 4-20-09 **Originating Organization:** General Atomics

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Engineering analyses will be performed 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. This analysis will show analytically how the CR and RSM guide tubes respond during normal differential thermal expansion, dynamic fluid flow, seismic shaking, and non-normal events.	GA,	6 months. Must be complete about 1.5 years before end of final design	500

TRL Rating Sheet

Vendor: GA	**Document Number:** RC2-SSC-2d.3-m	**Revision:** 0

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

Title: Reactor Internals - Upper Core Restraint (UCR) - Metallic

Description:

The UCR consists of interconnecting metallic plenum elements which are located on the top layer of the reflector and permanent side reflector graphite blocks. All elements have hexagonal shapes except for those above the permanent side reflectors, which match the configuration of the permanent side reflectors. The UCR provides lateral restraint and stability to the core array, provides neutron flux attenuation to the reactor internals and reactor vessel, channels helium primary coolant from the upper plenum to the entrance of the reactor core, and maintains the capability to refuel the reactor core. In addition, the UCR provides channels to insert control rods, reserve shutdown pellets, and the In-Core Flux Mapping Unit (IFMU). The UCR assembly provides core stability and limits core movements. The plenum elements are keyed together to limit relative lateral motion within the UCR assembly. Connections between the core barrel and the plenum elements over the permanent side reflectors limit the translation and rotation of the UCR assembly with respect to the core barrel.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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.

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 full-scale UCR array with guide tubes and above columns of graphite control-rod fuel elements.	GA, Wyle Laboratories, Hazen Research, or INL CTF	18 months, must be completed 2019 prior to fab. of CRs for NGNP	1500

DDN(s) Supported: C.11.03.44 **Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein, Russ Vollman, Jessie Crozier

Date: 4-20-09 **Originating Organization:** General Atomics

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Conduct UCR structural integrity testing. A full-size UCR array 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 2019 prior to fab. of CRs for NGNP	1000
3. Conduct UCR flow distribution and pressure drop testing. A full-size UCR array will be subjected to operational and accident-condition loads and temperatures to quantify pressure drop coefficients and coolant hole flow distributions as function of the Reynolds number.	GA, Wyle Laboratories, Hazen Research, or INL CTF	18 months, must be completed by 2019 prior to fab. of CRs for NGNP	1000

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-2d.4-m	**Revision:**	0

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

Title: Reactor Internals - Upper Core Restraint (UCR) - Metallic

Description:

The UCR consists of interconnecting metallic plenum elements which are located on the top layer of the reflector and permanent side reflector graphite blocks. All elements have hexagonal shapes except for those above the permanent side reflectors, which match the configuration of the permanent side reflectors. The UCR provides lateral restraint and stability to the core array, provides neutron flux attenuation to the reactor internals and reactor vessel, channels helium primary coolant from the upper plenum to the entrance of the reactor core, and maintains the capability to refuel the reactor core. In addition, the UCR provides channels to insert control rods, reserve shutdown pellets, and the In-Core Flux Mapping Unit (IFMU). The UCR assembly provides core stability and limits core movements. The plenum elements are keyed together to limit relative lateral motion within the UCR assembly. Connections between the core barrel and the plenum elements over the permanent side reflectors limit the translation and rotation of the UCR assembly with respect to the core barrel.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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) ☐

TRL 7 is achieved when the following conditions are met: (1) UCR flow-induced and seismic vibration testing has been completed and the results confirm the position of the UCR elements and that there is no impact on the capability to insert or withdraw the CRs in the reactor, (2) UCR structural integrity testing has been completed and the results confirm that the UCR design has adequate margin against operational failure, and (3) UCR testing for estimating pressure drop and flow distribution through the UCR elements.

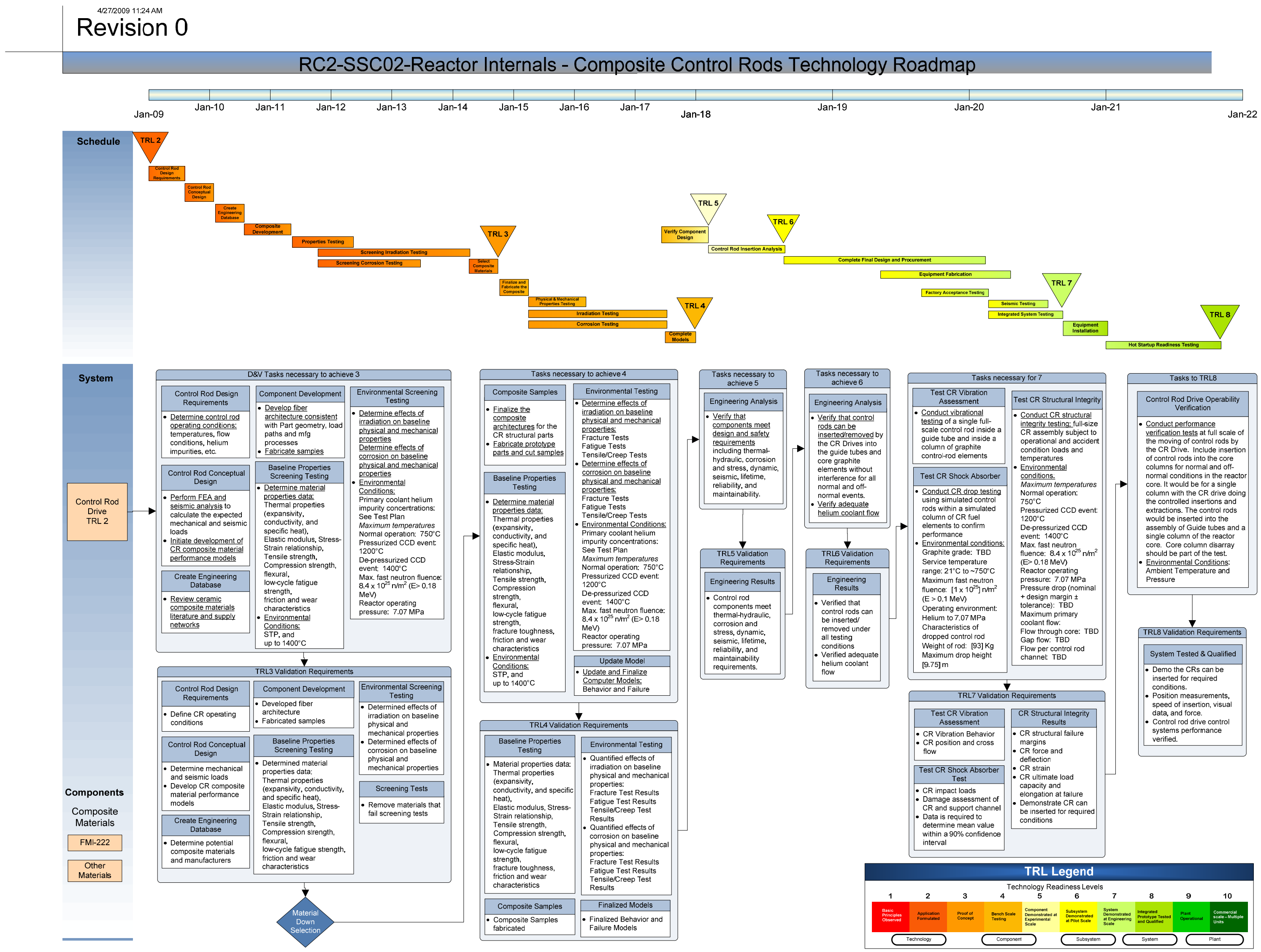
Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Perform testing at the NGNP to verify integration of the UCR interfaces with other components as part of the integrated testing of the NCA, RSM, RSM guide tubes, CR guide tube, and CR. The test should include both normal and off-normal events. Position measurements and visual inspection are required.	GA and NGNP operator	Must be completed ~3 months prior to installation of NCAs and CRs in NGNP	1500

DDN(s) Supported:	**Technology Case File:**

Subject Matter Expert Making Determination: John Saurwein, Russ Vollman, Jessie Crozier

Date: 4-20-09 **Originating Organization:** General Atomics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Revision 0
RC2-SSC02-Reactor Internals - Composite Control Rods Technology Roadmap
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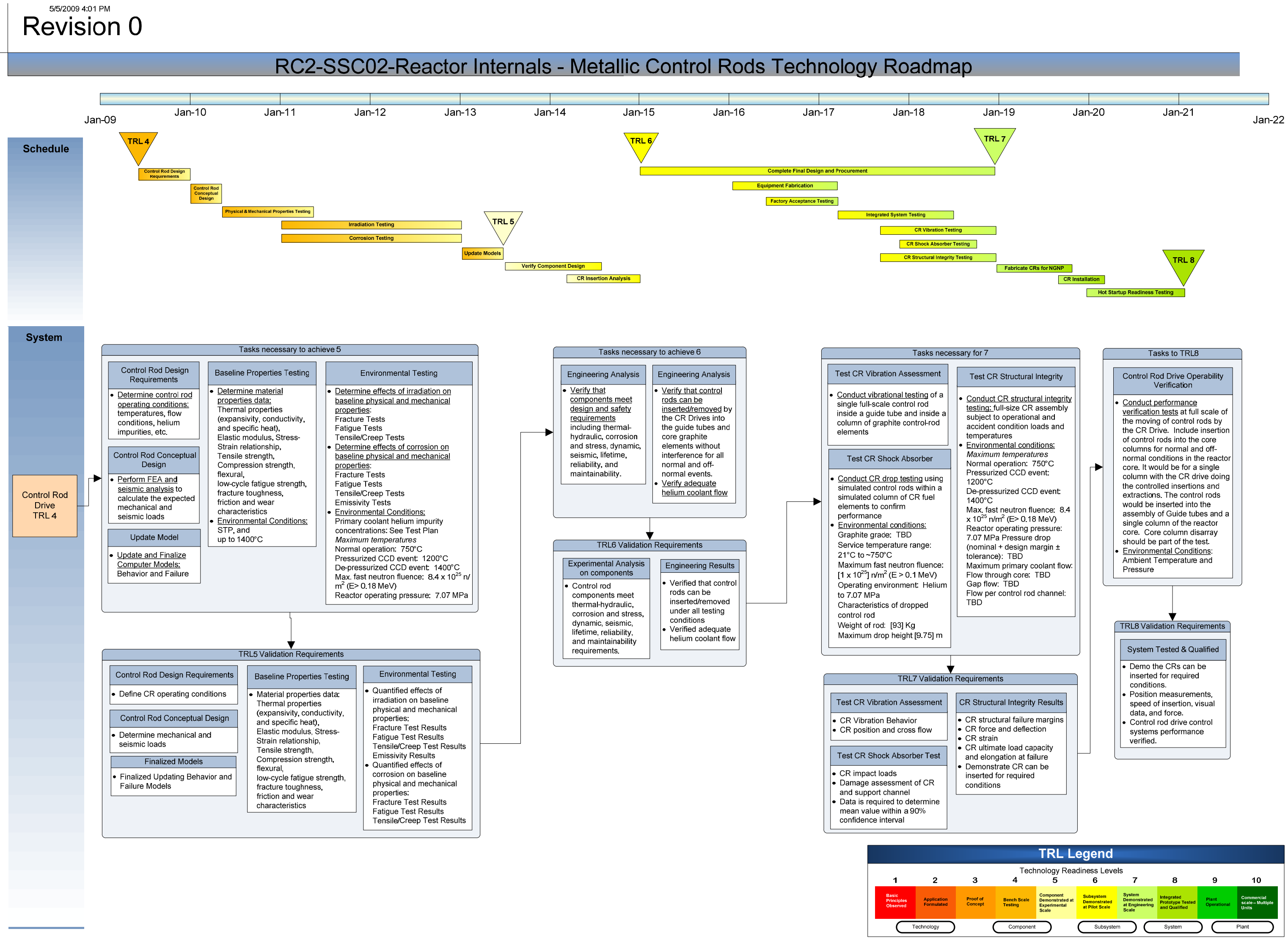
Schedule
TRL 2
Control Rod Design Requirements
Control Rod Conceptual Design
Create Engineering Database
Composite Development
Properties Testing
Screening Irradiation Testing
Screening Corrosion Testing
TRL 3
Select Composite Materials
Finalize and Fabricate the Composite
Physical & Mechanical Properties Testing
Irradiation Testing
Corrosion Testing
TRL 4
Complete Models
TRL 5
Verify Component Design
TRL 6
Control Rod Insertion Analysis
Complete Final Design and Procurement
Equipment Fabrication
Factory Acceptance Testing
TRL 7
Seismic Testing
Integrated System Testing
Equipment Installation
TRL 8
Hot Startup Readiness Testing
System
Control Rod Drive TRL 2
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 1400°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: 750°C
Pressurized CCD event: 1200°C
De-pressurized CCD event: 1400°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
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 1400°C
Environmental Testing
• Determine effects of irradiation on baseline physical and mechanical properties: Fracture Tests, Fatigue Tests, Tensile/Creep Tests
• Determine effects of corrosion on baseline physical and mechanical properties: Fracture Tests, Fatigue Tests, Tensile/Creep Tests
• Environmental Conditions: Primary coolant helium impurity concentrations: See Test Plan
Maximum temperatures
Normal operation: 750°C
Pressurized CCD event: 1200°C
De-pressurized CCD event: 1400°C
Max. fast neutron fluence: 8.4 x 10^{25} n/m^2 (E> 0.18 MeV)
Reactor operating pressure: 7.07 MPa
Update Model
• Update and Finalize Computer Models: Behavior and Failure
TRL4 Validation Requirements
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: Fracture Test Results, Fatigue Test Results, Tensile/Creep Test Results
• Quantified effects of corrosion on baseline physical and mechanical properties: Fracture Test Results, Fatigue Test Results, Tensile/Creep Test Results
Composite Samples
• Composite Samples fabricated
Finalized Models
• Finalized Behavior and Failure Models
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
Engineering Results
• Control rod components meet thermal-hydraulic, corrosion and stress, dynamic, seismic, lifetime, reliability, and maintainability requirements.
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
Engineering Results
• Verified that control rods can be inserted/ removed under all testing conditions
• Verified adequate helium coolant flow
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 ~750°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 temperatures
Normal operation: 750°C
Pressurized CCD event: 1200°C
De-pressurized CCD event: 1400°C
Max. fast neutron fluence: 8.4 x 10^{25} n/m^2 (E> 0.18 MeV)
Reactor operating pressure: 7.07 MPa
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
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
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. It 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.
Components
Composite Materials
FMI-222
Other Materials
TRL Legend
Technology Readiness Levels
1
2
3
4
5
6
7
8
9
10
Basic Principles Observed
Application Formulated
Proof of Concept
Bench Scale Testing
Component Demonstrated at Experimental Scale
Subsystem Demonstrated at Pilot Scale
System Demonstrated at Engineering Scale
Integrated Prototype Tested and Qualified
Plant Operational
Commercial Scale - Multiple Units
Technology
Component
Subsystem
System
Plant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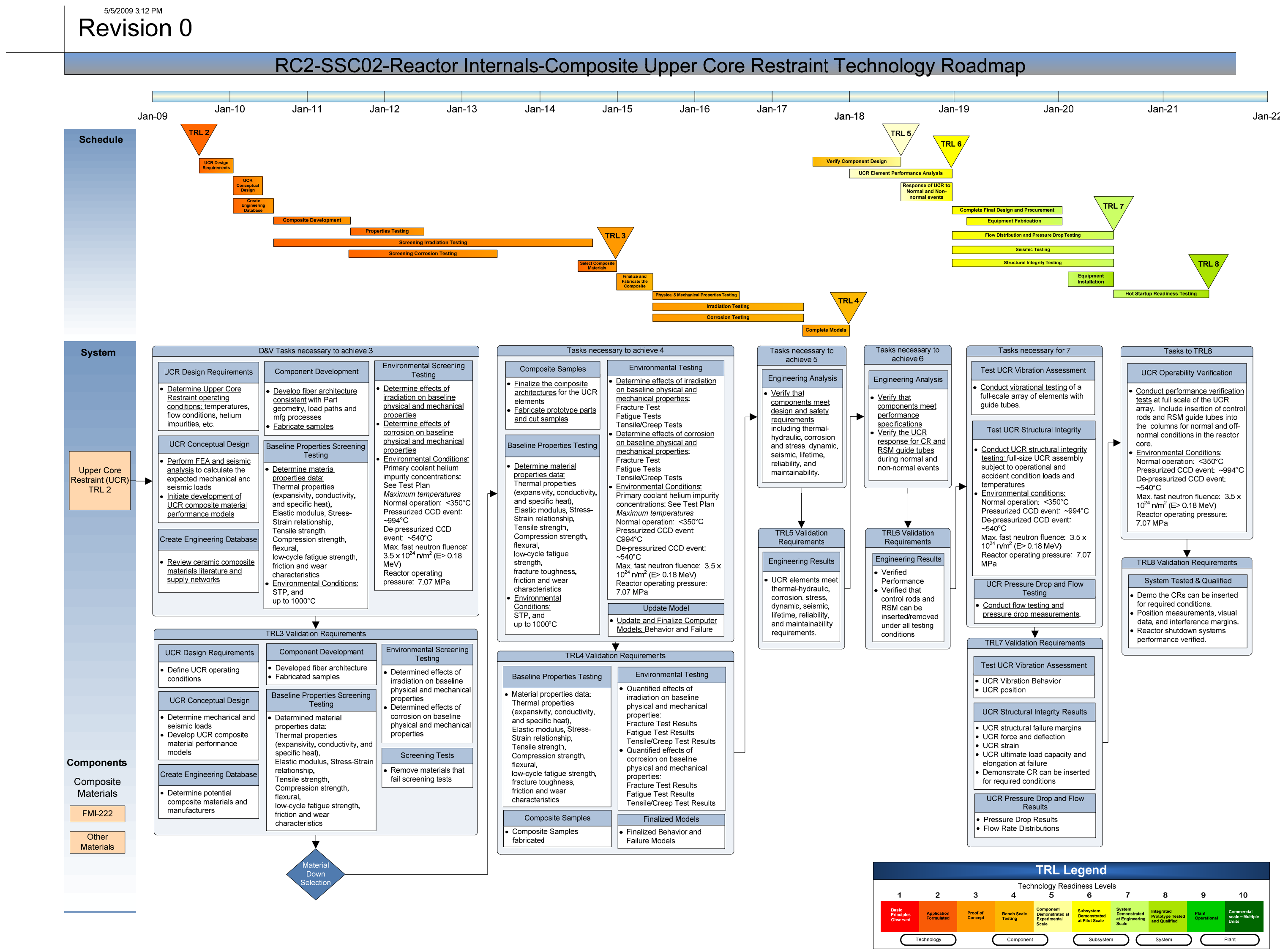
Revision 0
RC2-SSC02-Reactor Internals - Metallic Control Rods Technology Roadmap
Jan-09 Jan-10 Jan-11 Jan-12 Jan-13 Jan-14 Jan-15 Jan-16 Jan-17 Jan-18 Jan-19 Jan-20 Jan-21 Jan-22
Schedule
TRL 4
Control Rod Design Requirements
Control Rod Conceptual Design
Physical & Mechanical Properties Testing
Irradiation Testing
Corrosion Testing
TRL 5
Update Models
Verify Component Design
CR Insertion Analysis
TRL 6
Complete Final Design and Procurement
Equipment Fabrication
Factory Acceptance Testing
Integrated System Testing
CR Vibration Testing
CR Shock Absorber Testing
CR Structural Integrity Testing
TRL 7
Fabricate CRs for NGNP
CR Installation
Hot Startup Readiness Testing
TRL 8
System
Control Rod Drive TRL 4
Tasks necessary to achieve 5
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
Update Model
• Update and Finalize Computer Models: Behavior and Failure
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 1400°C
Environmental Testing
• Determine effects of irradiation on baseline physical and mechanical properties: Fracture Tests Fatigue Tests Tensile/Creep Tests
• Determine effects of corrosion on baseline physical and mechanical properties: Fracture Tests Fatigue Tests Tensile/Creep Tests Emissivity Tests
• Environmental Conditions: Primary coolant helium impurity concentrations: See Test Plan Maximum temperatures Normal operation: 750°C Pressurized CCD event: 1200°C De-pressurized CCD event: 1400°C Max. fast neutron fluence: 8.4 x 10^{25} n/m² (E> 0.18 MeV) Reactor operating pressure: 7.07 MPa
TRL5 Validation Requirements
Control Rod Design Requirements
• Define CR operating conditions
Control Rod Conceptual Design
• Determine mechanical and seismic loads
Finalized Models
• Finalized Updating Behavior and Failure Models
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: Fracture Test Results Fatigue Test Results Tensile/Creep Test Results Emissivity Results
• Quantified effects of corrosion on baseline physical and mechanical properties: Fracture Test Results Fatigue Test Results Tensile/Creep Test Results
Tasks necessary to achieve 6
Engineering Analysis
• Verify that components meet design and safety requirements including thermal-hydraulic, corrosion and stress, dynamic, seismic, lifetime, reliability, and maintainability.
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
Experimental Analysis on components
• Control rod components meet thermal-hydraulic, corrosion and stress, dynamic, seismic, lifetime, reliability, and maintainability requirements.
Engineering Results
• Verified that control rods can be inserted/removed under all testing conditions
• Verified adequate helium coolant flow
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 grace: TBD Service temperature range: 21°C to ~750°C Maximum fast neutron fluence: [1 x 10^{25}] n/m² (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 temperatures Normal operation: 750°C Pressurized CCD event: 1200°C De-pressurized CCD event: 1400°C Max. fast neutron fluence: 8.4 x 10^{25} n/m² (E> 0.18 MeV) Reactor operating pressure: 7.07 MPa 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
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. It 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.
TRL Legend
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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RC2-SSC02-Reactor Internals-Composite Upper Core Restraint Technology Roadmap
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Jan-17
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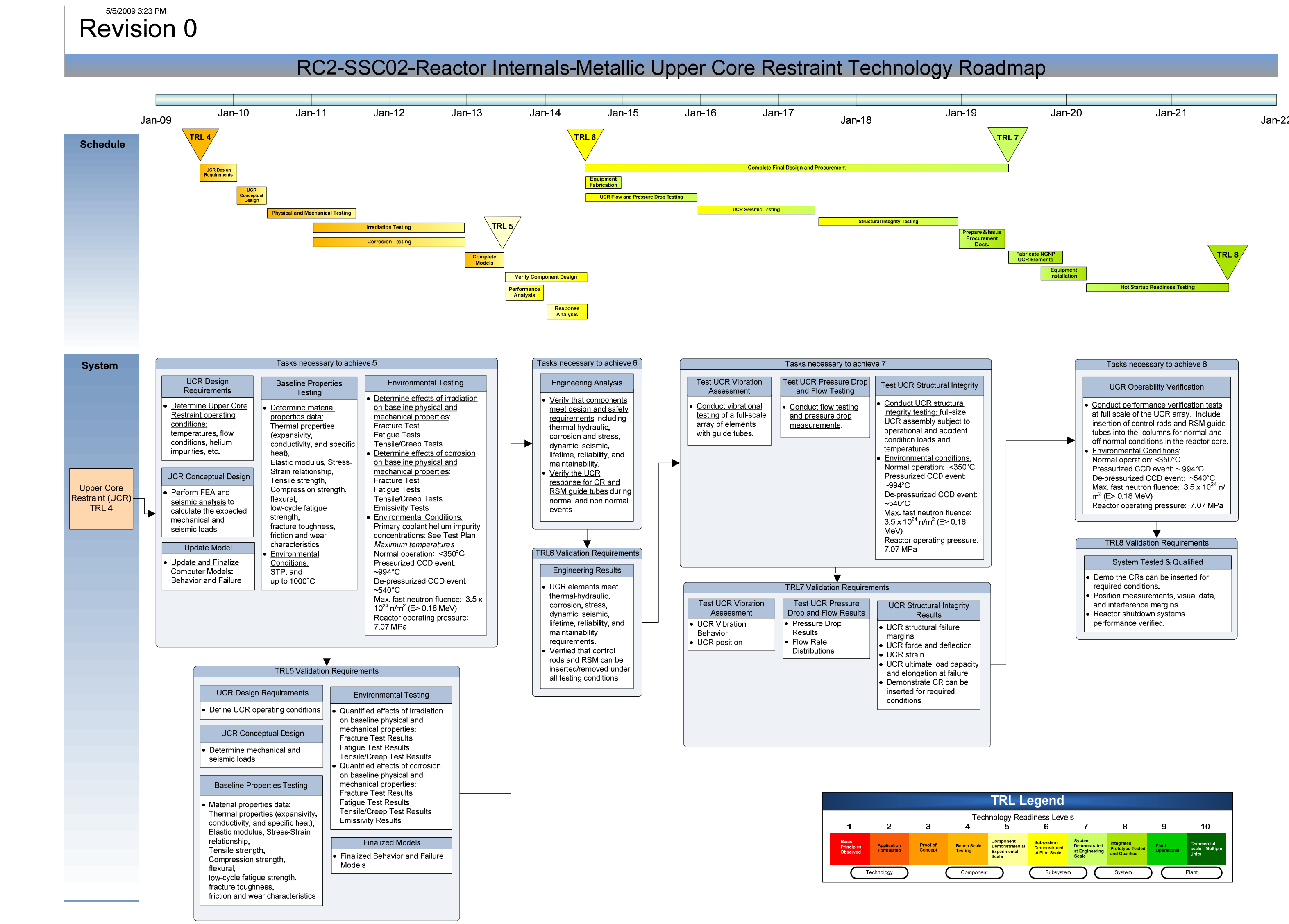
Jan-19
Jan-20
Jan-21
Jan-22
Schedule
TRL 2
UCR Design Requirements
UCR Conceptual Design
Create Engineering Database
Composite Development
Properties Testing
Screening Irradiation Testing
Screening Corrosion Testing
TRL 3
Select Composite Materials
Finalize and Fabricate the Composite
Physical & Mechanical Properties Testing
Irradiation Testing
Corrosion Testing
TRL 4
Complete Models
TRL 5
Verify Component Design
TRL 6
UCR Element Performance Analysis
Response of UCR to Normal and Non-normal events
Complete Final Design and Procurement
Equipment Fabrication
TRL 7
Flow Distribution and Pressure Drop Testing
Seismic Testing
Structural Integrity Testing
Equipment Installation
TRL 8
Hot Startup Readiness Testing
System
Upper Core Restraint (UCR) TRL 2
Components
Composite Materials
FMI-222
Other Materials
D&V Tasks necessary to achieve 3
UCR Design Requirements
• Determine Upper Core Restraint operating conditions: temperatures, flow conditions, helium impurities, etc.
UCR Conceptual Design
• Perform FEA and seismic analysis to calculate the expected mechanical and seismic loads
• Initiate development of UCR 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 1000°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: <350°C
Pressurized CCD event: ~994°C
De-pressurized CCD event: ~540°C
Max. fast neutron fluence: 3.5 x 10^{24} n/m² (E> 0.18 MeV)
Reactor operating pressure: 7.07 MPa
TRL3 Validation Requirements
UCR Design Requirements
• Define UCR operating conditions
UCR Conceptual Design
• Determine mechanical and seismic loads
• Develop UCR 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 (expansvity, 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
Tasks necessary to achieve 4
Composite Samples
• Finalize the composite architectures for the UCR elements
• 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 1000°C
Environmental Testing
• Determine effects of irradiation on baseline physical and mechanical properties: Fracture Test, Fatigue Tests, Tensile/Creep Tests
• Determine effects of corrosion on baseline physical and mechanical properties: Fracture Test, Fatigue Tests, Tensile/Creep Tests
• Environmental Conditions: Primary coolant helium impurity concentrations: See Test Plan
Maximum temperatures
Normal operation: <350°C
Pressurized CCD event: ⊄994°C
De-pressurized CCD event: ~540°C
Max. fast neutron fluence: 3.5 x 10^{24} n/m² (E> 0.18 MeV)
Reactor operating pressure: 7.07 MPa
Update Model
• Update and Finalize Computer Models: Behavior and Failure
TRL4 Validation Requirements
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: Fracture Test Results, Fatigue Test Results, Tensile/Creep Test Results
• Quantified effects of corrosion on baseline physical and mechanical properties: Fracture Test Results, Fatigue Test Results, Tensile/Creep Test Results
Composite Samples
• Composite Samples fabricated
Finalized Models
• Finalized Behavior and Failure Models
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
Engineering Results
• UCR elements meet thermal-hydraulic, corrosion, stress, dynamic, seismic, lifetime, reliability, and maintainability requirements.
Tasks necessary to achieve 6
Engineering Analysis
• Verify that components meet performance specifications
• Verify the UCR response for CR and RSM guide tubes during normal and non-normal events
TRL6 Validation Requirements
Engineering Results
• Verified Performance
• Verified that control rods and RSM can be inserted/removed under all testing conditions
Tasks necessary for 7
Test UCR Vibration Assessment
• Conduct vibrational testing of a full-scale array of elements with guide tubes.
Test UCR Structural Integrity
• Conduct UCR structural integrity testing: full-size UCR assembly subject to operational and accident condition loads and temperatures
• Environmental conditions: Normal operation: <350°C
Pressurized CCD event: ~994°C
De-pressurized CCD event: ~540°C
Max. fast neutron fluence: 3.5 x 10^{24} n/m² (E> 0.18 MeV)
Reactor operating pressure: 7.07 MPa
UCR Pressure Drop and Flow Testing
• Conduct flow testing and pressure drop measurements.
TRL7 Validation Requirements
Test UCR Vibration Assessment
• UCR Vibration Behavior
• UCR position
UCR Structural Integrity Results
• UCR structural failure margins
• UCR force and deflection
• UCR strain
• UCR ultimate load capacity and elongation at failure
• Demonstrate CR can be inserted for required conditions
UCR Pressure Drop and Flow Results
• Pressure Drop Results
• Flow Rate Distributions
Tasks to TRL8
UCR Operability Verification
• Conduct performance verification tests at full scale of the UCR array. Include insertion of control rods and RSM guide tubes into the columns for normal and off-normal conditions in the reactor core.
• Environmental Conditions: Normal operation: <350°C
Pressurized CCD event: ~994°C
De-pressurized CCD event: ~540°C
Max. fast neutron fluence: 3.5 x 10^{24} n/m² (E> 0.18 MeV)
Reactor operating pressure: 7.07 MPa
TRL8 Validation Requirements
System Tested & Qualified
• Demo the CRs can be inserted for required conditions.
• Position measurements, visual data, and interference margins.
• Reactor shutdown systems performance verified.
TRL Legend
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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RC2-SSC02-Reactor Internals-Metallic Upper Core Restraint Technology Roadmap
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Schedule
TRL 4
UCR Design Requirements
UCR Conceptual Design
Physical and Mechanical Testing
Irradiation Testing
Corrosion Testing
TRL 5
Complete Models
Verify Component Design
Performance Analysis
Response Analysis
TRL 6
Complete Final Design and Procurement
Equipment Fabrication
UCR Flow and Pressure Drop Testing
UCR Seismic Testing
Structural Integrity Testing
TRL 7
Prepare & Issue Procurement Docs.
Fabricate NGNP UCR Elements
Equipment Installation
TRL 8
Hot Startup Readiness Testing
System
Upper Core Restraint (UCR) TRL 4
Tasks necessary to achieve 5
UCR Design Requirements
• Determine Upper Core Restraint operating conditions: temperatures, flow conditions, helium impurities, etc.
UCR Conceptual Design
• Perform FEA and seismic analysis to calculate the expected mechanical and seismic loads
Update Model
• Update and Finalize Computer Models: Behavior and Failure
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 1000°C
Environmental Testing
• Determine effects of irradiation on baseline physical and mechanical properties: Fracture Test, Fatigue Tests, Tensile/Creep Tests
• Determine effects of corrosion on baseline physical and mechanical properties: Fracture Test, Fatigue Tests, Tensile/Creep Tests, Emissivity Tests
• Environmental Conditions: Primary coolant helium impurity concentrations: See Test Plan
Maximum temperatures
Normal operation: <350°C
Pressurized CCD event: ~994°C
De-pressurized CCD event: ~540°C
Max. fast neutron fluence: 3.5 x 10^24 n/m^2 (E> 0.18 MeV)
Reactor operating pressure: 7.07 MPa
TRL5 Validation Requirements
UCR Design Requirements
• Define UCR operating conditions
UCR Conceptual Design
• Determine mechanical and seismic loads
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: Fracture Test Results, Fatigue Test Results, Tensile/Creep Test Results
• Quantified effects of corrosion on baseline physical and mechanical properties: Fracture Test Results, Fatigue Test Results, Tensile/Creep Test Results, Emissivity Results
Finalized Models
• Finalized Behavior and Failure Models
Tasks necessary to achieve 6
Engineering Analysis
• Verify that components meet design and safety requirements including thermal-hydraulic, corrosion and stress, dynamic, seismic, lifetime, reliability, and maintainability.
• Verify the UCR response for CR and RSM guide tubes during normal and non-normal events
TRL6 Validation Requirements
Engineering Results
• UCR elements meet thermal-hydraulic, corrosion, stress, dynamic, seismic, lifetime, reliability, and maintainability requirements.
• Verified that control rods and RSM can be inserted/removed under all testing conditions
Tasks necessary to achieve 7
Test UCR Vibration Assessment
• Conduct vibrational testing of a full-scale array of elements with guide tubes.
Test UCR Pressure Drop and Flow Testing
• Conduct flow testing and pressure drop measurements.
Test UCR Structural Integrity
• Conduct UCR structural integrity testing: full-size UCR assembly subject to operational and accident condition loads and temperatures
• Environmental conditions: Normal operation: <350°C, Pressurized CCD event: ~994°C, De-pressurized CCD event: ~540°C, Max. fast neutron fluence: 3.5 x 10^24 n/m^2 (E> 0.18 MeV), Reactor operating pressure: 7.07 MPa
TRL7 Validation Requirements
Test UCR Vibration Assessment
• UCR Vibration Behavior
• UCR position
Test UCR Pressure Drop and Flow Results
• Pressure Drop Results
• Flow Rate Distributions
UCR Structural Integrity Results
• UCR structural failure margins
• UCR force and deflection
• UCR strain
• UCR ultimate load capacity and elongation at failure
• Demonstrate CR can be inserted for required conditions
Tasks necessary to achieve 8
UCR Operability Verification
• Conduct performance verification tests at full scale of the UCR array. Include insertion of control rods and RSM guide tubes into the columns for normal and off-normal conditions in the reactor core.
• Environmental Conditions: Normal operation: <350°C, Pressurized CCD event: ~ 994°C, De-pressurized CCD event: ~540°C, Max. fast neutron fluence: 3.5 x 10^24 n/m^2 (E> 0.18 MeV), Reactor operating pressure: 7.07 MPa
TRL8 Validation Requirements
System Tested & Qualified
• Demo the CRs can be inserted for required conditions.
• Position measurements, visual data, and interference margins.
• Reactor shutdown systems performance verified.
TRL Legend
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

4.3 RC2-SSC-3 Hot Duct TRL

TRL Rating Sheets, TRL 2 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-3.1	**Revision:**	0

☐ 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 750-800°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 ~340°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 800°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 ☐ PCS ☐ BOP

PASSC:	**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 2) Establish code applicability (ASME pressure vessel, nuclear, piping, QA)	GA GA/URS-WD	6 months	$87 (Not including GA scope of work)

DDN(s) Supported: C.11.02.02, C.11.02.12, C.11.02.13, C.11.02.14, C.11.02.15, N.11.02.12, N.11.02.13, N.11.02.14	**Technology Case File:**

Subject Matter Expert Making Determination: Greg Walz

Date: 4-27-09 **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:** RC2-SSC-3.2	**Revision:** 0

☐ 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 750-800°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 ~340°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 800°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 ☐ PCS ☐ BOP

PASSC:	**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, C.11.02.15, N.11.02.12, N.11.02.13, N.11.02.14	**Technology Case File:**

Subject Matter Expert Making Determination: Greg Walz

Date: 4-27-09	**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:**	RC2-SSC-3.3	**Revision:**	0

☐ 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 750-800°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 ~340°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 800°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 ☐ PCS ☐ BOP

PASSC:	**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, C.11.02.15, N.11.02.12, N.11.02.13, N.11.02.14

Technology Case File:

Subject Matter Expert Making Determination: Greg Walz

Date: 4-27-09 **Originating Organization:** Washington Division of URS

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

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-3.4	**Revision:**	0

☐ 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 750-800°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 ~340°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 800°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 ☐ PCS ☐ BOP

PASSC:	**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, C.11.02.15, N.11.02.12, N.11.02.13, N.11.02.14

Technology Case File:

Subject Matter Expert Making Determination: Greg Walz

Date: 4-27-09 **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		Included Above

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-3.5	**Revision:**	0

☐ 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 750-800°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 ~340°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 800°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 ☐ PCS ☐ BOP

PASSC:	**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, C.11.02.15, N.11.02.12, N.11.02.13, N.11.02.14	**Technology Case File:**

Subject Matter Expert Making Determination: Greg Walz

Date: 4-27-09	**Originating Organization:** Washington Division of URS

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

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-3.6	**Revision:**	0

☐ 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 750-800°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 ~340°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 800°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 ☐ PCS ☐ BOP

PASSC:	**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	GA/URS-WD	2 Years (coordinate with others)	500 (GA, INL/BEA Scope not included)
2) Validate concept of in-service inspection	GA/URS-WD		
3) Test uniformity of duct heating	GA/URS-WD		

DDN(s) Supported: C.11.02.02, C.11.02.12, C.11.02.13, C.11.02.14, C.11.02.15, N.11.02.12, N.11.02.13, N.11.02.14

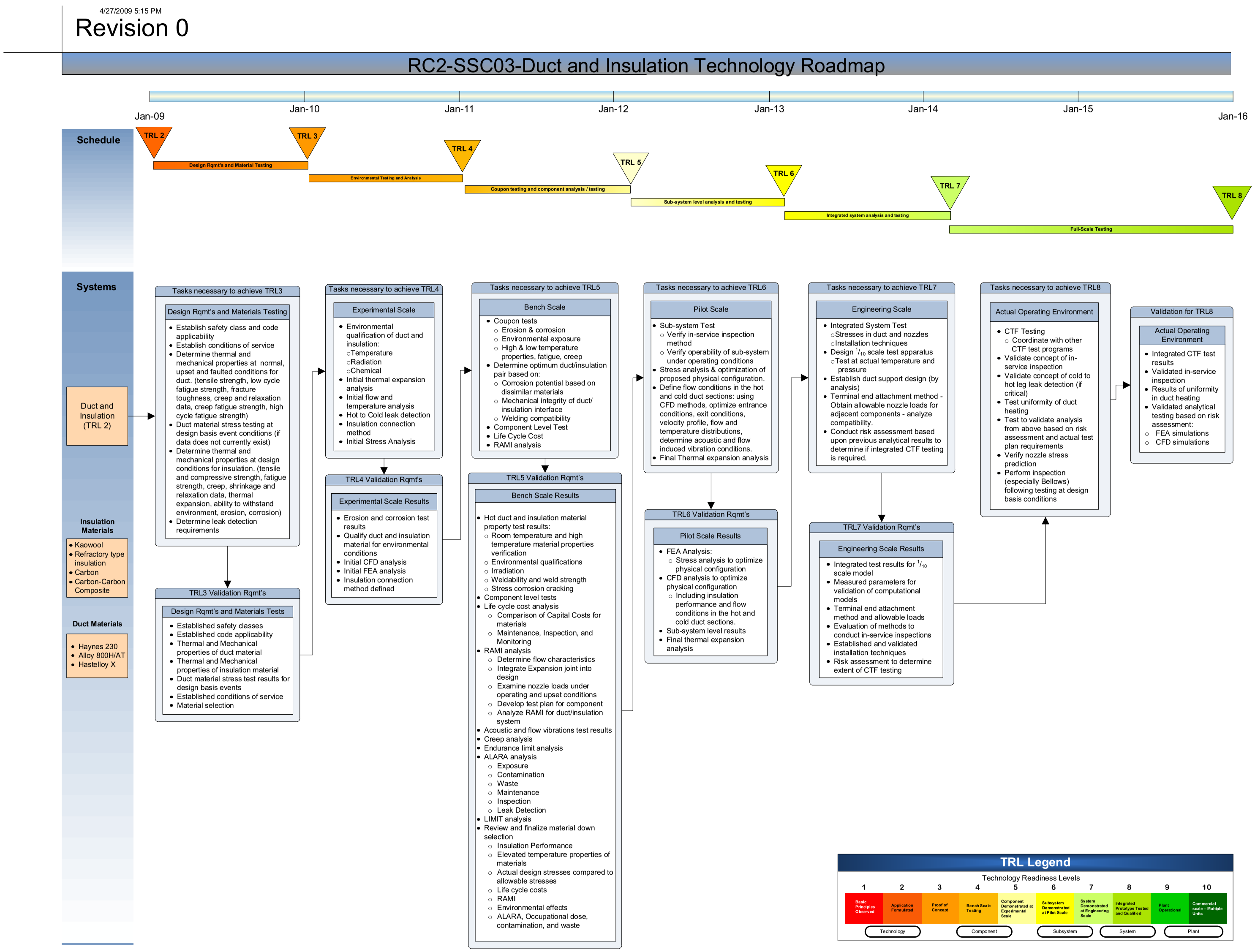
Technology Case File:

Subject Matter Expert Making Determination: Greg Walz

Date: 4-27-09 **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		Included above


4/27/2009 5:15 PM
Revision 0
RC2-SSC03-Duct and Insulation Technology Roadmap
Jan-09
Jan-10
Jan-11
Jan-12
Jan-13
Jan-14
Jan-15
Jan-16
Schedule
TRL 2
TRL 3
TRL 4
TRL 5
TRL 6
TRL 7
TRL 8
Design Rqmt's and Material Testing
Environmental Testing and Analysis
Coupon testing and component analysis / testing
Sub-system level analysis and testing
Integrated system analysis and testing
Full-Scale Testing
Systems
Duct and Insulation (TRL 2)
Insulation Materials
• Kaowool
• Refractory type insulation
• Carbon
• Carbon-Carbon Composite
Duct Materials
• Haynes 230
• Alloy 800H/AT
• Hastelloy X
Tasks necessary to achieve TRL3
Design Rqmt's and Materials Testing
• 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
TRL3 Validation Rqmt's
Design Rqmt's and Materials Tests
• Established safety classes
• Established code applicability
• Thermal and Mechanical properties of duct material
• Thermal and Mechanical properties of insulation material
• Duct material stress test results for design basis events
• Established conditions of service
• Material selection
Tasks necessary to achieve TRL4
Experimental Scale
• Environmental qualification of duct and insulation:
○Temperature
○Radiation
○Chemical
• Initial thermal expansion analysis
• Initial flow and temperature analysis
• Hot to Cold leak detection
• Insulation connection method
• Initial Stress Analysis
TRL4 Validation Rqmt's
Experimental Scale Results
• Erosion and corrosion test results
• Qualify duct and insulation material for environmental conditions
• Initial CFD analysis
• Initial FEA analysis
• Insulation connection method defined
Tasks necessary to achieve TRL5
Bench Scale
• Coupon tests
○ Erosion & corrosion
○ Environmental exposure
○ High & low temperature properties, fatigue, creep
• Determine optimum duct/insulation pair based on:
○ Corrosion potential based on dissimilar materials
○ Mechanical integrity of duct/ insulation interface
○ Welding compatibility
• Component Level Test
• Life Cycle Cost
• RAMI analysis
TRL5 Validation Rqmt's
Bench Scale Results
• Hot duct and insulation material property test results:
○ Room temperature and high temperature material properties verification
○ Environmental qualifications
○ Irradiation
○ Weldability and weld strength
○ Stress corrosion cracking
• Component level tests
• Life cycle cost analysis
○ Comparison of Capital Costs for materials
○ Maintenance, Inspection, and Monitoring
• RAMI analysis
○ Determine flow characteristics
○ Integrate Expansion joint into design
○ Examine nozzle loads under operating and upset conditions
○ Develop test plan for component
○ Analyze RAMI for duct/insulation system
• Acoustic and flow vibrations test results
• Creep analysis
• Endurance limit analysis
• ALARA analysis
○ Exposure
○ Contamination
○ Waste
○ Maintenance
○ Inspection
○ Leak Detection
• LIMIT analysis
• Review and finalize material down selection
○ Insulation Performance
○ Elevated temperature properties of materials
○ Actual design stresses compared to allowable stresses
○ Life cycle costs
○ RAMI
○ Environmental effects
○ ALARA, Occupational dose, contamination, and waste
Tasks necessary to achieve TRL6
Pilot Scale
• Sub-system Test
○ 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.
• Final Thermal expansion analysis
TRL6 Validation Rqmt's
Pilot Scale Results
• FEA Analysis:
○ Stress analysis to optimize physical configuration
• CFD analysis to optimize physical configuration
○ Including insulation performance and flow conditions in the hot and cold duct sections.
• Sub-system level results
• Final thermal expansion analysis
Tasks necessary to achieve TRL7
Engineering Scale
• 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)
• 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.
TRL7 Validation Rqmt's
Engineering Scale Results
• Integrated test results for $^1/_{10}$ scale model
• Measured parameters for validation of computational models
• Terminal end attachment method and allowable loads
• Evaluation of methods to conduct in-service inspections
• Established and validated installation techniques
• Risk assessment to determine extent of CTF testing
Tasks necessary to achieve TRL8
Actual Operating Environment
• CTF Testing
○ Coordinate with other CTF test programs
• Validate concept of in-service inspection
• Validate concept of cold to hot leg leak detection (if critical)
• Test uniformity of duct heating
• 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
Validation for TRL8
Actual Operating Environment
• Integrated CTF test results
• Validated in-service inspection
• Results of uniformity in duct heating
• Validated analytical testing based on risk assessment:
○ FEA simulations
○ CFD simulations
TRL Legend
Technology Readiness Levels
1
2
3
4
5
6
7
8
9
10
Basic Principles Observed
Application Formulated
Proof of Concept
Bench Scale Testing
Component Demonstrated at Experimental Scale
Subsystem Demonstrated at Pilot Scale
System Demonstrated at Engineering Scale
Integrated Prototype Tested and Qualified
Plant Operational
Commercial scale - Multiple Units
Technology
Component
Subsystem
System
Plant

4.4 RC2-SSC-4a Reactor Core Assembly and RC2-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:**	RC2-SSC-4a.1	**Revision:**	0

☐ 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 ☐ PCS ☐ BOP

PASSC:	**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:**	RC2-SSC-4a.2	**Revision:**	0

☐ 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 ☐ PCS ☐ BOP

PASSC:	**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:**	RC2-SSC-4a.3	**Revision:**	0

☐ 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 ☐ PCS ☐ BOP

PASSC:	**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:**	RC2-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 ☐ PCS ☐ BOP

PASSC:	**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.

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-4b.2	**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	☐ PCS	☐ BOP

PASSC:	**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	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 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.

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 physical, mechanical, thermal, and chemical properties. Perform inspections of selected fuel and reflector elements at the end of startup testing.	Vendor/ Operator	NGNP Startup Phase	15,000

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

Subject Matter Expert Making Determination: Matt Richards

Date: 10/30/08 **Originating Organization:** General Atomics

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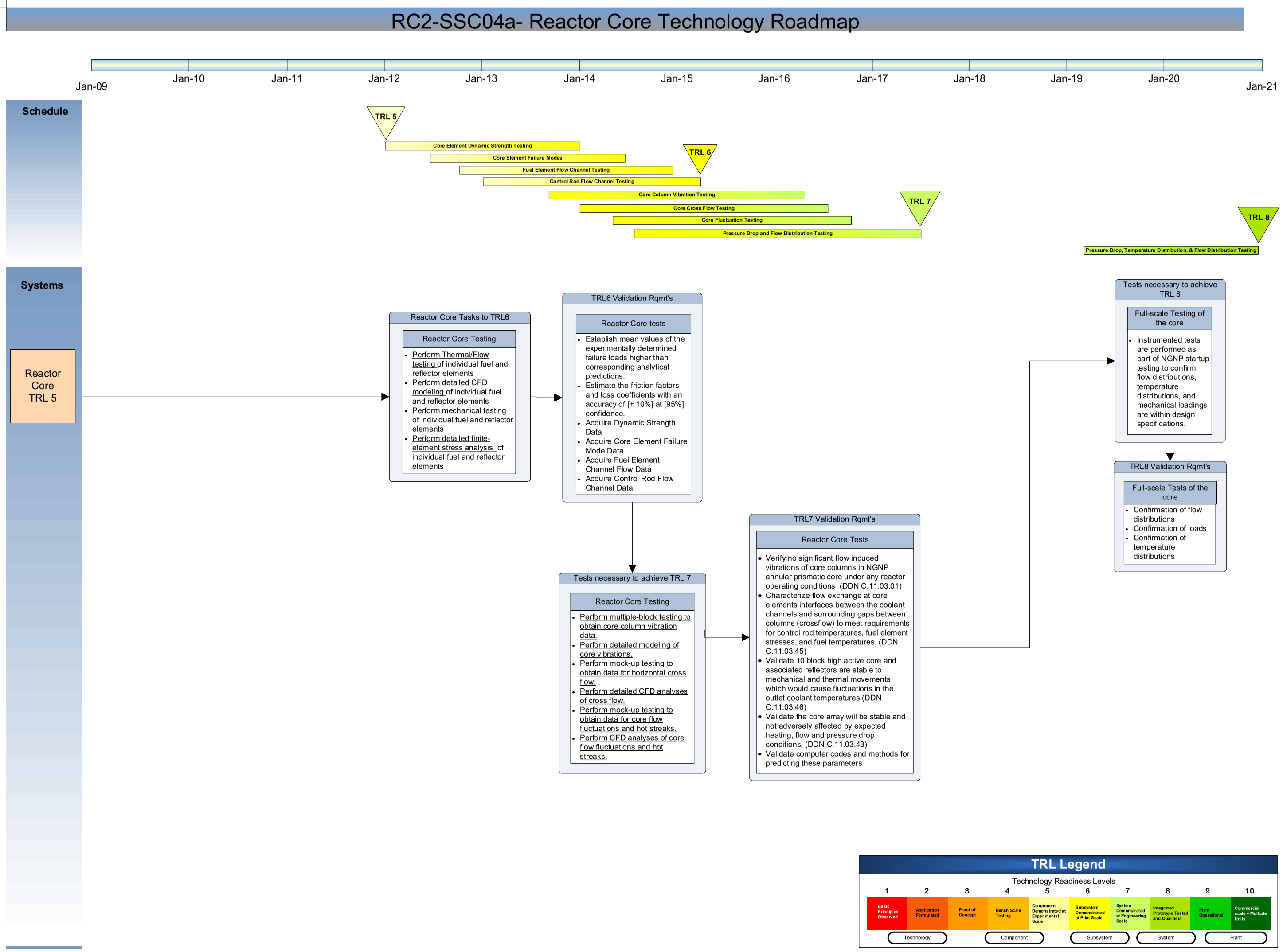

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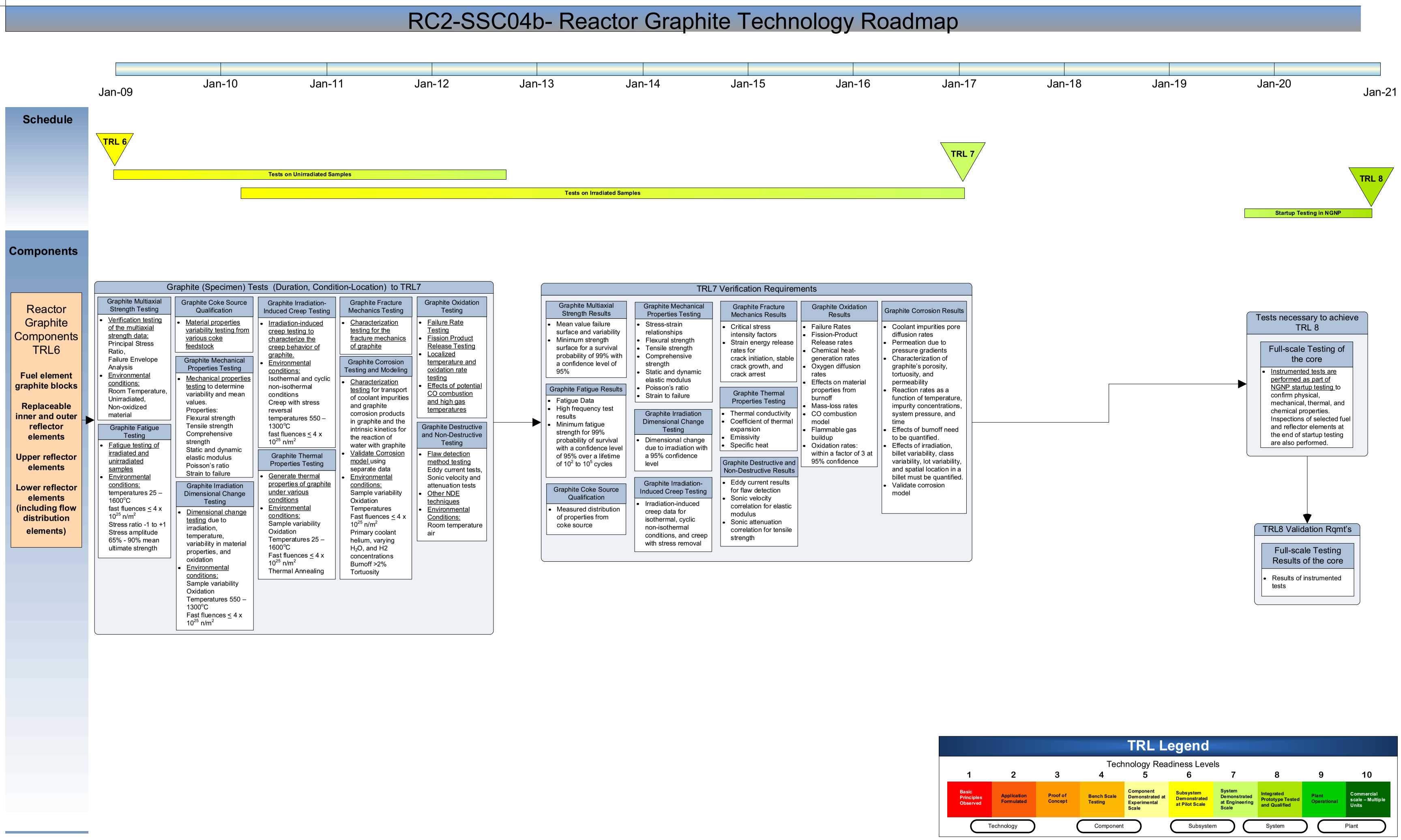
Revision 0



4.5 RC2-SSC-5 Reactor Pressure Vessel

TRL Rating Sheets, TRL 5 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-5.1	**Revision:**	0

☐ 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.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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 design requirements. 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	750

DDN(s) Supported: None	**Technology Case File:**
Subject Matter Expert Making Determination: John Saurwein	
Date: 4-2-09	**Originating Organization:** General Atomics

Additional Basis Sheet(s)

Basis:

required size are within the capabilities of a major forging supplier (Japan Steel Works).

GA has also concluded that it will not be necessary to include an active vessel cooling system (VCS) in the NGNP design with a reactor outlet helium temperature of 750°C. The vessel operating temperature is only a concern if the temperatures are pushing against the 371°C ASME code limit for SA508/533 steel and the design lifetime of the RPV is very long (e.g., 60 years). With a reactor outlet helium temperature of 750°C, the reactor inlet helium temperature will be limited to 350°C, which will result in maximum vessel temperatures of about 320°C. Vessel temperatures at this level should be sufficiently low to eliminate any concerns with regard to vessel creep damage.

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 environmental conditions within the RPV will be under 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 RPV temperature is 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 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 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:**	RC2-SSC-5.2	**Revision:**	0

☐ 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.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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 system have been defined, 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 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	250

DDN(s) Supported: None
Technology Case File:

Subject Matter Expert Making Determination: John Saurwein

Date: 4-2-09 **Originating Organization:** General Atomics

Additional Action Sheet(s)			
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
2. Develop the detailed design of the RPV 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	500

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-5.3	**Revision:**	0

☐ 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.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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 RPV analyses have been completed and 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.

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 in the actual operating environment (i.e., in the NGNP during start-up testing) to verify vessel temperatures are 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: 4-2-09	**Originating Organization:** General Atomics

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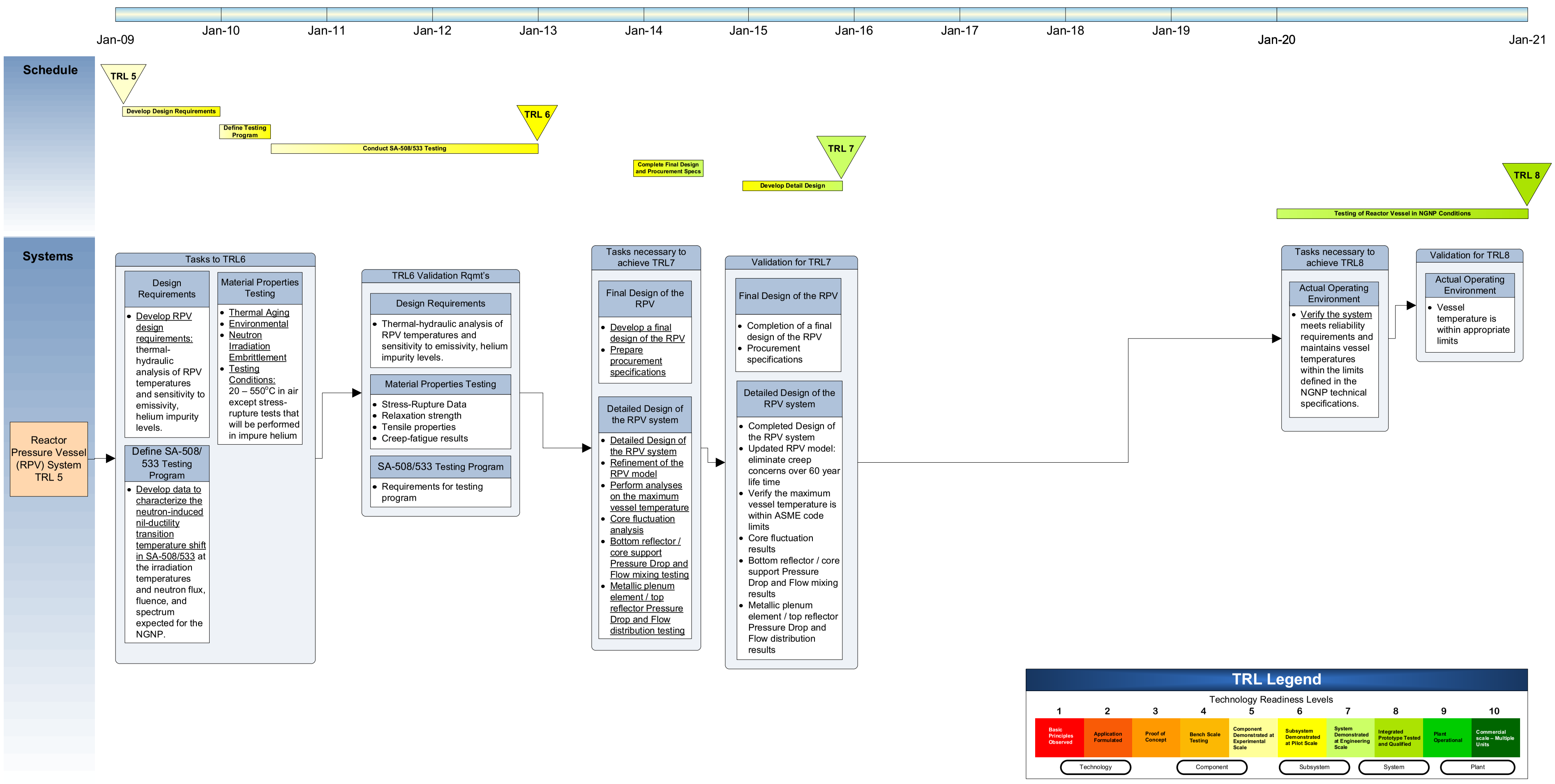
RC2-SSC05-Reactor Pressure Vessel Technology Roadmap



4.6 RC2-SSC-6 Helium Circulator

TRL Rating Sheets, TRL 6 and 7

Technology Development Road Map

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-6.1	**Revision:**	0

☐ 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 and, hence, to the external turbo-generator set.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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:**	RC2-SSC-6.2	**Revision:**	0

☐ 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 and, hence, to the external turbo-generator set.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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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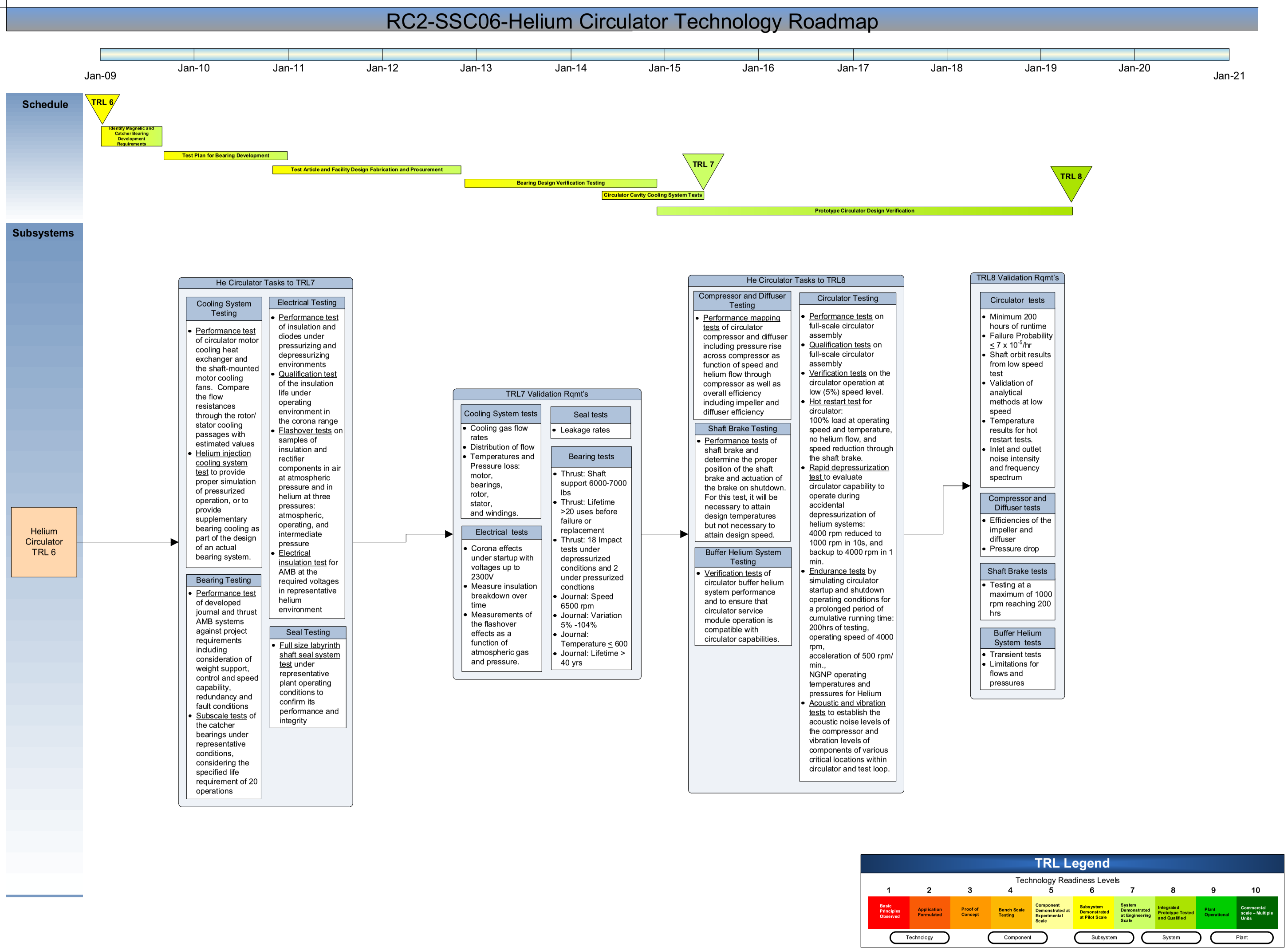

4.7 RC2-SSC-8 Shutdown Cooling Heat Exchanger

TRL Rating Sheets, TRL 4 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor: GA	**Document Number:** RC2-SSC-8.1	**Revision:** 0	

☐ 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 (ASTM A213, T22). The heat is removed by 60°C-pressurized (4.8MPa) water. The SCHE does not have a safety function.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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 (ASTM A213, T22).

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:	RC2-SSC-8.2	Revision:	0

☐ 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 (ASTM A213, T22). The heat is removed by 60^{o}C-pressurized (4.8MPa) water. The SCHE does not have a safety function.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	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:** RC2-SSC-8.3	**Revision:** 0

☐ 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 (ASTM A213, T22). The heat is removed by 60^{o}C-pressurized (4.8MPa) water. The SCHE does not have a safety function.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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: RC2-SSC-8.4	Revision: 0

☐ 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 (ASTM A213, T22). The heat is removed by 60°C-pressurized (4.8MPa) water. The SCHE does not have a safety function.

Area:	☒ NHSS	☐ HTS	☐ HPS	☐ PCS	☐ BOP

PASSC:	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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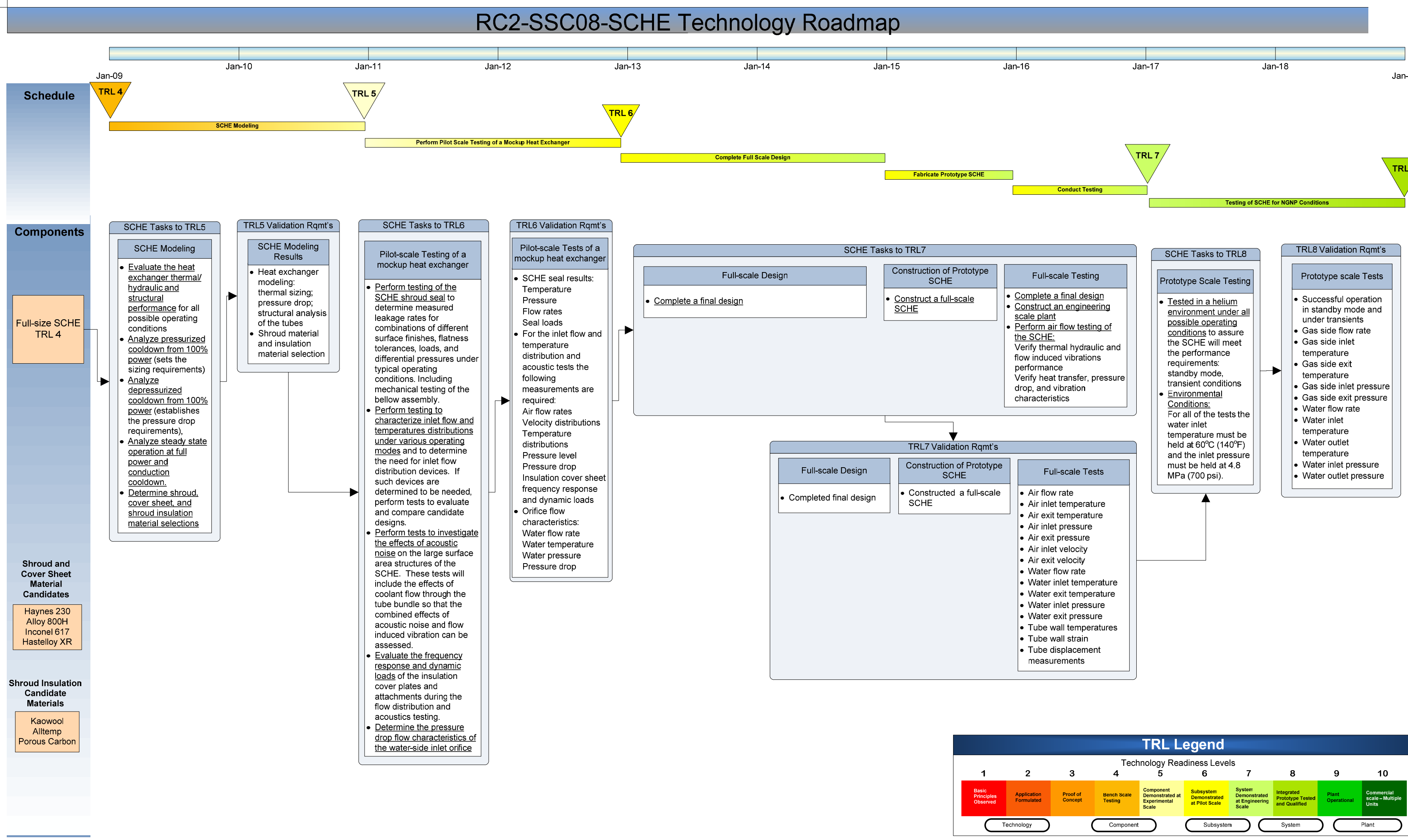
Revision 0



4.8 RC2-SSC-9 Reactor Cavity Cooling System

TRL Rating Sheets, TRL 4 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-9.1	**Revision:**	0

☐ 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 ☐ PCS ☐ BOP

PASSC:	**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:**	RC2-SSC-9.2	**Revision:**	0

☐ 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	☐ PCS	☐ BOP

PASSC:	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:**	RC2-SSC-9.3	**Revision:**	0

☐ 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 ☐ PCS ☐ BOP

PASSC:	**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:**	RC2-SSC-9.4	**Revision:**	0

☐ 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 ☐ PCS ☐ BOP

PASSC:	**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 RC2-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) ☐

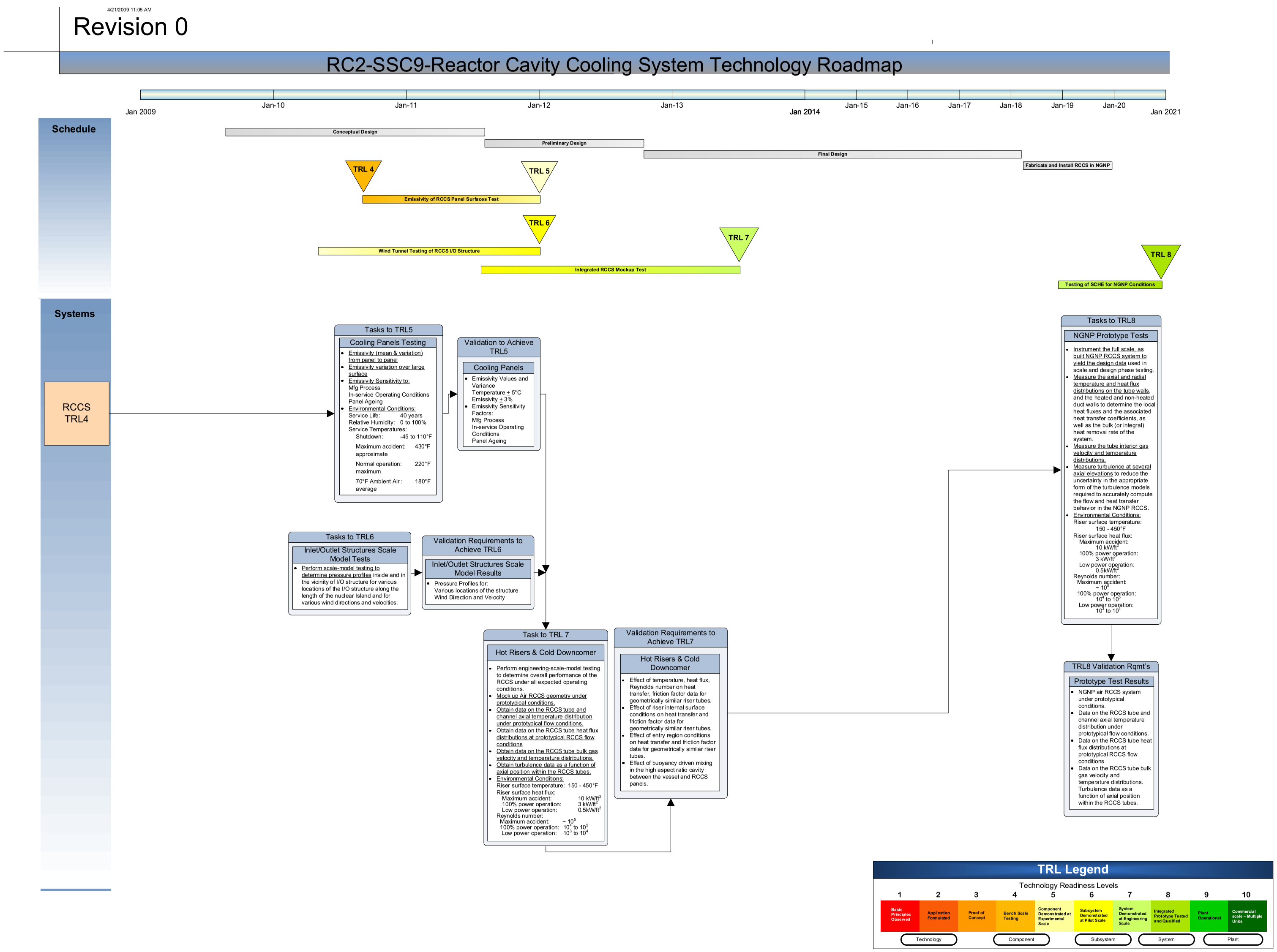
Actions *(list all)*	**Actionee**	**Schedule**	**Cost ($K)**
Perform testing of NGNP RCCS to verify design under all expected operating conditions.	As-built NGNP RCCS	Part of startup testing. Overall duration 24 months.	Part of startup testing.

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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Revision 0
RC2-SSC9-Reactor Cavity Cooling System Technology Roadmap
Jan 2009
Jan-10
Jan-11
Jan-12
Jan-13
Jan 2014
Jan-15
Jan-16
Jan-17
Jan-18
Jan-19
Jan-20
Jan 2021
Schedule
Conceptual Design
Preliminary Design
Final Design
Fabricate and Install RCCS in NGNP
TRL 4
TRL 5
Emissivity of RCCS Panel Surfaces Test
TRL 6
Wind Tunnel Testing of RCCS I/O Structure
TRL 7
Integrated RCCS Mockup Test
TRL 8
Testing of SCHE for NGNP Conditions
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
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
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/ft^2
100% power operation: 3 kW/ft^2
Low power operation: 0.5kW/ft^2
Reynolds number:
Maximum accident: ~ 10^5
100% power operation: 10^4 to 10^5
Low power operation: 10^3 to 10^4
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.
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/ft^2
100% power operation: 3 kW/ft^2
Low power operation: 0.5kW/ft^2
Reynolds number:
Maximum accident: ~ 10^5
100% power operation: 10^4 to 10^5
Low power operation: 10^3 to 10^4
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.
TRL Legend
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

4.9 RC2-SSC-10 Steam Generator

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

Technology Development Road Map

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-10.1.1	**Revision:**	0

☐ 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 the appropriate GA Test Plan.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-10.1.2	**Revision:**	0

☐ 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 the appropriate GA Test Plan.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**Parent:**	**WBS:**

Technology Readiness Level

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

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

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.

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) Demonstrate the ability to fabricate the helical coiled tubes.	GA/Vendor/ INL	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 and corrosion properties of alloy 800H under NGNP design conditions	GA/Test Lab/ INL	3 Years starting at start of CD	15,000
8) Perform testing to verify the mechanical and corrosion properties of alloy 2¼Cr - 1Mo under NGNP design conditions	GA/Test Lab/ INL		
9) Perform testing to verify the mechanical properties of the bimetallic weld under NGNP design conditions.	GA/Test Lab/ INL		
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/ INL	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	GA/Test Lab		

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-10.1.3	**Revision:**	0

☐ 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 the appropriate GA Test Plan.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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:**	RC2-SSC-10.1.4	**Revision:**	0

☐ 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 the appropriate GA Test Plan.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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**

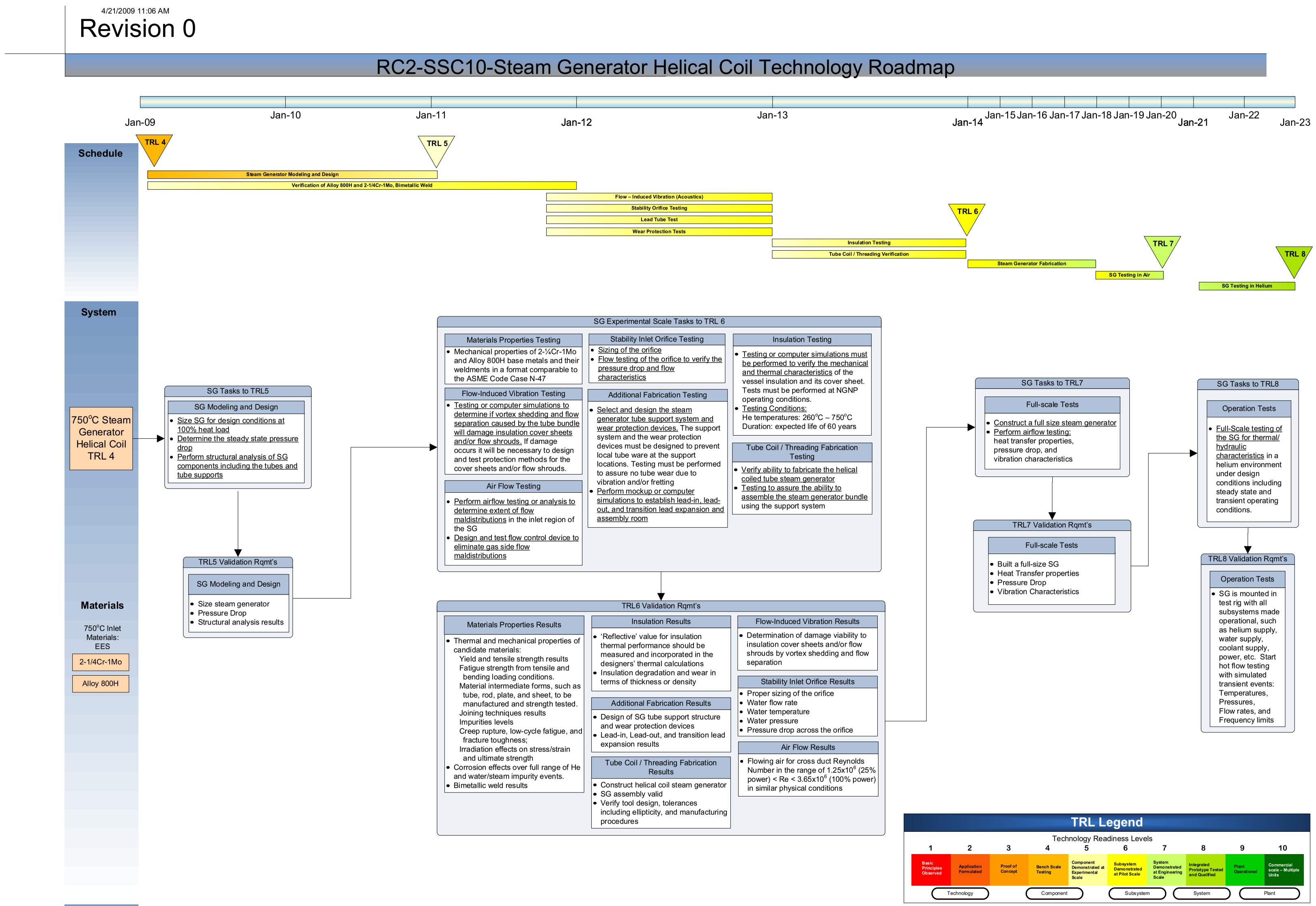
Basis for Rating (Check box if continued on additional sheets) ☐

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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Revision 0
RC2-SSC10-Steam Generator Helical Coil Technology Roadmap
Jan-09
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Schedule
TRL 4
TRL 5
Steam Generator Modeling and Design
Verification of Alloy 800H and 2-1/4Cr-1Mo, Bimetallic Weld
Flow – Induced Vibration (Acoustics)
Stability Orifice Testing
Lead Tube Test
Wear Protection Tests
TRL 6
Insulation Testing
Tube Coil / Threading Verification
Steam Generator Fabrication
TRL 7
SG Testing in Air
TRL 8
SG Testing in Helium
System
750°C Steam Generator Helical Coil TRL 4
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
SG Experimental Scale Tasks to TRL 6
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
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.
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
Stability Inlet Orifice Testing
• Sizing of the orifice
• Flow testing of the orifice to verify the pressure drop and flow characteristics
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 – 750°C
Duration: expected life of 60 years
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
TRL6 Validation Rqmt's
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
Flow-Induced Vibration Results
• Determination of damage viability to insulation cover sheets and/or flow shrouds by vortex shedding and flow separation
Stability Inlet Orifice Results
• Proper sizing of the orifice
• Water flow rate
• Water temperature
• Water pressure
• Pressure drop across the orifice
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
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
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
Materials
750°C Inlet Materials: EES
2-1/4Cr-1Mo
Alloy 800H
TRL Legend
Technology Readiness Levels
1
2
3
4
5
6
7
8
9
10
Basic Principles Observed
Application Formulated
Proof of Concept
Bench Scale Testing
Component Demonstrated at Experimental Scale
Subsystem Demonstrated at Pilot Scale
System Demonstrated at Engineering Scale
Integrated Prototype Tested and Qualified
Plant Operational
Commercial scale – Multiple Units
Technology
Component
Subsystem
System
Plant

4.10 RC2-SSC-12 High Temperature Valves

TRL Rating Sheets, TRL 4 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-12.1	**Revision:**	0

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

Title: High Temperature Isolation Valves and Pressure Relief Valve

Description:
High temperature isolation valves, located in the steam circuit, enable isolation of the steam generator. Isolation valves are also present on the inlet of the branch to the power turbine, and on the inlet to the reboiler used to produce process steam. The Reactor Pressure Vessel is protected by a high temperature pressure actuated relief valve. Helium temperatures are assumed to be 750-800°C and 300-325°C for the hot and cold legs respectively. 241.4 kg/sec of steam is produced at 541°C and 17.25MPa, consistent with the CoGen plant described in the NGNP PCS alternatives report. The plant life is 60 years. It is also assumed that for isolation valves on the steam generator, there will be three (3) valves on the steam side of each hot and cold leg. These valves will be an integral part of the plant protective system actions for steam loop isolation events. (Reference GA Report 911120/0). Isolation and control valves may be 2 way or 3 way, globe, ball or gate type, Relief valves may be spring or pilot operated. In general, valves may be manual, automatic (remote) or actuated. Valves in some locations may require remote service.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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) ☒

See attached Basis Sheet

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)**
See Action Sheets			

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

Subject Matter Expert Making Determination: David T. Carroccia

Date: 4-27-09	**Originating Organization:** Washington Division of URS

Additional Basis Sheet(s)

Basis for rating:

TRL-4 is assigned to the high temperature valves. Although the relief valve is expected to require more development then the steam isolation valves, it is assigned the same level as the steam valves. Steam isolation and control valves are commercially available in the size range needed with ratings that include the pressure and temperature noted above but material properties in the range necessary are not tabulated in the ASME Code Section III Subsection NH. Valves cannot be rated at TRL-5 because a complete specification for the high temperature high pressure high flow valves does not exist at the current time.
The critical characteristics for which these valves must be designed are not defined for the service conditions at NGNP , particularly, the following must be established:

- ASME Section III Code Acceptance of materials
- Allowable Valve Leakage
- Valve response times required
- Acceptable valve open pressure drop
- Accident excursion temperatures
- Accident excursion pressures
- Valve Configuration
- Actuator type
- Application requirements and mounting location

Additional Action Sheet(s)			
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 OM-1-2007, ASME OM-2008) a) Establish Safety Class	GA/URS-WD GA	1 year	\$300 to \$350 (not including GA Scope)
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 material properties - Endurance limit analysis - Welds, Cladding - Material corrosion - Stress corrosion cracking, fracture toughness - Elevated temperature properties & fatigue data - 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 - Remote Service Requirements - Determine applicability of EPRI PPM	GA/URS-WD		Material Testing included in above cost estimate
3) Establish conditions of service under normal and design basis event conditions	GA		
4) Valve material stress testing	GA/URS-WD		
5) Material durability tests	GA/URS-WD & Valve Suppliers		
6) Determine performance of gaskets, packing material and seals	GA/URS-WD		
Cont’d			

7) Material Selection and Valve Configuration (Body, Bonnet, Seat, Seal, Stem and Packing) 8) 3D modeling and analytical simulation including FEA stress analysis, heat transfer analysis and CFD modeling 9) Endurance Limit and Creep Analysis 10) Identify Maintenance Requirements, ALARA analysis and RAMI characteristics 11) Erosion and corrosion accelerated wear testing, Environmental qualification of valve materials, He leak tightness & Weld Methods, Dissimilar Materials and Differential Thermal Expansion 12) Interfaces with adjoining structures, piping. Insulation, installation, maintenance access, contamination control	GA/URS-WD GA/URS-WD GA/URS-WD, SME GA/URS-WD, GA/URS-WD GA/URS-WD	1.5year	\$300 to \$600k (not including GA Scope) Simulations, tests and analysis included in above cost estimate

TRL Rating Sheet

Vendor: GA	**Document Number:** RC2-SSC-12.2	**Revision:** 0

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

Title: High Temperature Isolation Valves and Pressure Relief Valve

Description:

High temperature isolation valves, located in the steam circuit, enable isolation of the steam generator. Isolation valves are also present on the inlet of the branch to the power turbine, and on the inlet to the reboiler used to produce process steam. The Reactor Pressure Vessel is protected by a high temperature pressure actuated relief valve. Helium temperatures are assumed to be 750-800°C and 300-325°C for the hot and cold legs respectively. 241.4 kg/sec of steam is produced at 541°C and 17.25MPa, consistent with the CoGen plant described in the NGNP PCS alternatives report. The plant life is 60 years. It is also assumed that for isolation valves on the steam generator, there will be three (3) valves on the steam side of each hot and cold leg. These valves will be an integral part of the plant protective system actions for steam loop isolation events. (Reference GA Report 911120/0). Isolation and control valves may be 2 way or 3 way, globe, ball or gate type, Relief valves may be spring or pilot operated. In general, valves may be manual, automatic (remote) or actuated. Valves in some locations may require remote service.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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-5 is achieved upon successful completion of the action items identified in the TRL rating sheet for TRL-4. A complete specification for the high temperature, high pressure, high flow valves exists when TRL-5 is achieved.

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	GA/URS-WD	1 year	400 - 450
2) Tests using test apparatus	GA/URS-WD		
3) Determination of applicable NDE methods	GA/URS-WD		
4) Verify 3D (scale) models based on test results	GA/URS-WD		
5) Determine Leak Rate Detection Method Validation	GA/URS-WD		

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

Technology Case File:

Subject Matter Expert Making Determination: David T. Carroccia

Date: 4-27-09 **Originating Organization:** Washington Division of URS

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-12.3	**Revision:**	0

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

Title: High Temperature Isolation Valves and Pressure Relief Valve

Description:

High temperature isolation valves, located in the steam circuit, enable isolation of the steam generator. Isolation valves are also present on the inlet of the branch to the power turbine, and on the inlet to the reboiler used to produce process steam. The Reactor Pressure Vessel is protected by a high temperature pressure actuated relief valve. Helium temperatures are assumed to be 750-800°C and 300-325°C for the hot and cold legs respectively. 241.4 kg/sec of steam is produced at 541°C and 17.25MPa, consistent with the CoGen plant described in the NGNP PCS alternatives report. The plant life is 60 years. It is also assumed that for isolation valves on the steam generator, there will be three (3) valves on the steam side of each hot and cold leg. These valves will be an integral part of the plant protective system actions for steam loop isolation events. (Reference GA Report 911120/0). Isolation and control valves may be 2 way or 3 way, globe, ball or gate type, Relief valves may be spring or pilot operated. In general, valves may be manual, automatic (remote) or actuated. Valves in some locations may require remote service.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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: N.42.02.01, N.42.02.02	**Technology Case File:**
Subject Matter Expert Making Determination: David T. Carroccia	
Date: 4-27-09	**Originating Organization:** Washington Division of URS

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

TRL Rating Sheet

Vendor: GA	Document Number: RC2-SSC-12.4	Revision: 0

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

Title: High Temperature Isolation Valves and Pressure Relief Valve

Description:

High temperature isolation valves, located in the steam circuit, enable isolation of the steam generator. Isolation valves are also present on the inlet of the branch to the power turbine, and on the inlet to the reboiler used to produce process steam. The Reactor Pressure Vessel is protected by a high temperature pressure actuated relief valve. Helium temperatures are assumed to be 750-800°C and 300-325°C for the hot and cold legs respectively. 241.4 kg/sec of steam is produced at 541°C and 17.25MPa, consistent with the CoGen plant described in the NGNP PCS alternatives report. The plant life is 60 years. It is also assumed that for isolation valves on the steam generator, there will be three (3) valves on the steam side of each hot and cold leg. These valves will be an integral part of the plant protective system actions for steam loop isolation events. (Reference GA Report 911120/0). Isolation and control valves may be 2 way or 3 way, globe, ball or gate type, Relief valves may be spring or pilot operated. In general, valves may be manual, automatic (remote) or actuated. Valves in some locations may require remote service.

Area: ☐ NHSS ☒ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	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)	GA/URS-WD /INL	2 years coordinated with other activities at CTF)	750 – 950 Not including GA,INL/BEA scope
2) Stress Analysis Validation by prescribed mechanical loading and strain gage measurements	GA/URS-WD		
3) Temperature and Flow Analysis Validation	GA/URS-WD		
4) In-Service Inspection Techniques Validation	GA/URS-WD		

DDN(s) Supported: N.42.02.01, N.42.02.02	**Technology Case File:**
Subject Matter Expert Making Determination: David T. Carroccia	
Date: 4-27-09	**Originating Organization:** Washington Division of URS

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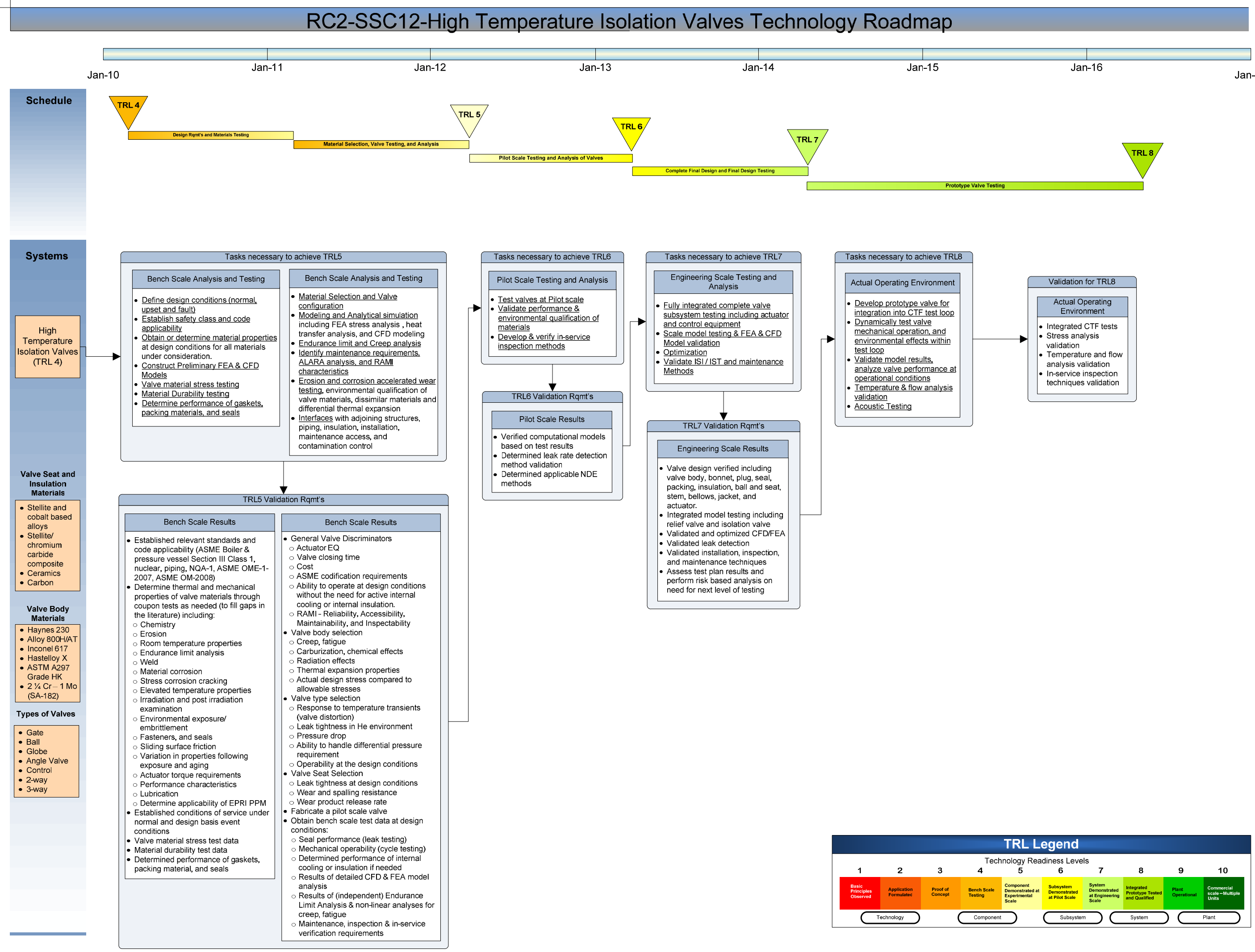
Revision 0



4.11 RC2-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: RC2-SSC-14.1	Revision: 0

☐ 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 ☐ PCS ☐ BOP

PASSC:	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

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-14.2	**Revision:**	0

☐ 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 ☐ PCS ☐ BOP

PASSC:	**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	Subsystems verified at pilot scale
TRL	**4**	**5**	**6**

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

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.

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 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.)	GA and FHEP vendor	18 months staring at beginning of FD	1900

DDN(s) Supported: C.21.01.01, C.21.01.02, C.21.01.03, C.21.01.06	**Technology Case File:**
Subject Matter Expert Making Determination: John Saurwein	
Date: 11-29-08	**Originating Organization:** General Atomics

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:** RC2-SSC-14.3	**Revision:** 0	

☐ 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 ☐ PCS ☐ BOP

PASSC:	**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:**	RC2-SSC-14.4	**Revision:**	0

☐ 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 ☐ PCS ☐ BOP

PASSC:	**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.

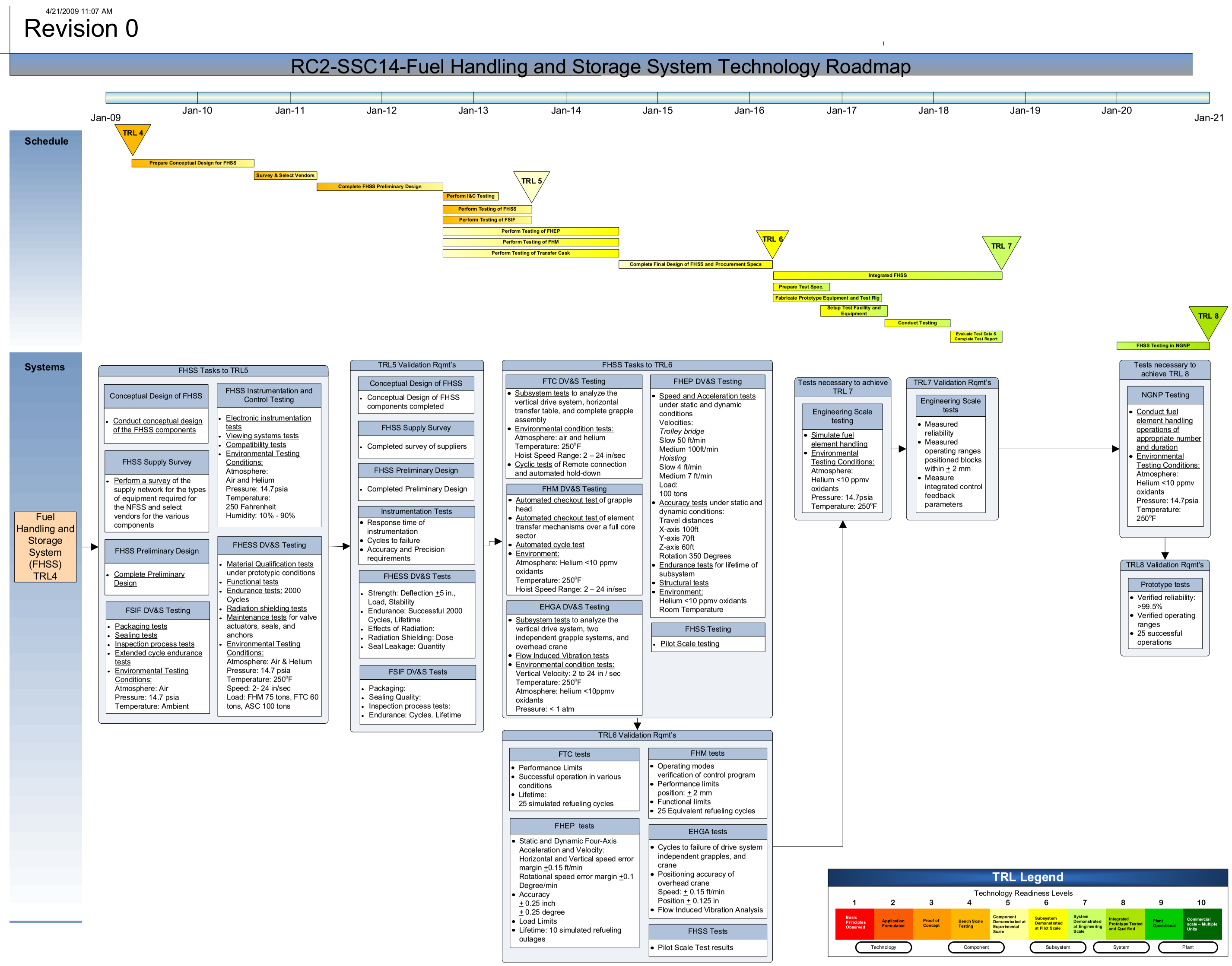
Outline of plan to get from current level to next level.
(Check box if continued on additional sheets) ☐

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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RC2-SSC14-Fuel Handling and Storage System Technology Roadmap
Jan-09
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Schedule
TRL 4
Prepare Conceptual Design for FHSS
Survey & Select Vendors
Complete FHSS Preliminary Design
TRL 5
Perform I&C Testing
Perform Testing of FHSS
Perform Testing of FSIF
Perform Testing of FHEP
Perform Testing of FHM
Perform Testing of Transfer Cask
TRL 6
Complete Final Design of FHSS and Procurement Specs
TRL 7
Integrated FHSS
Prepare Test Spec.
Fabricate Prototype Equipment and Test Rig
Setup Test Facility and Equipment
Conduct Testing
Evaluate Test Data & Complete Test Report
TRL 8
FHSS Testing in NGNP
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
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
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
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
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
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
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
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
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
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
TRL Legend
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

4.12 RC2-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:**	RC2-SSC-15.1	**Revision:**	0

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

Title: Reactor Control and Protection, Primary Circuit and Balance of Plant Instrumentation

Description:

This SSC 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 ☐ PCS ☐ BOP

PASSC:	**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 this SSC 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 Reactor Building design information, select leak detection instrumentation locations in the Reactor Building 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, it includes moisture monitoring and pressure instrumentation for steam leakage detection, operator information, and as a protection-logic, reactor-trip parameter. Furthermore, this SSC 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 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 isolation valve 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:**	RC2-SSC-15.2	**Revision:**	0

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

Title: Reactor Control and Protection, Primary Circuit and Balance of Plant Instrumentation

Description:
This SSC 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 ☐ PCS ☐ BOP

PASSC:	**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 this SSC 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:

The instrumentation developed in this SSC 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. Further it 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. It 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 this SSC 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 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:** RC2-SSC-15.3	**Revision:** 0

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

Title: Reactor Control and Protection, Primary Circuit and Balance of Plant Instrumentation

Description:

This SSC 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 ☐ PCS ☐ BOP

PASSC:	**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 this SSC 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:

The instrumentation developed in this SSC 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. It 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. Lastly, it 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.

This SSC 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:** RC2-SSC-15.4	**Revision:** 0

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

Title: Reactor Control and Protection, Primary Circuit and Balance of Plant Instrumentation

Description:

This SSC 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	☐ PCS	☐ BOP

PASSC:	**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 this SSC 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:

The instrumentation developed in this SSC 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. It also 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. Finally, it 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, this SSC requires 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).

This SSC 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	FD 60-78mo FD 72-80mo	100 1,600
4. Deliver instrumentation and repeat vendor acceptance tests on-site to validate operation.	GA	FD 80-84mo	200
5. Verify instrumentation mounting and cable installation capability.	GA	FD 82-84mo	50
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 42-46mo FD 46-78mo	40 100
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 Howden	FD 72-78mo FD 72-78mo	50 60
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.	GA BOP	FD 60-72mo FD 60-72mo	50 50

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-15.5	**Revision:**	0

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

Title: Reactor Control and Protection, Primary Circuit and Balance of Plant Instrumentation

Description:
This SSC 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 ☐ PCS ☐ BOP

PASSC:	**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 this SSC 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:

The instrumentation developed in this SSC provides detection of primary coolant leakage through measurement of pressure, temperature or radiation levels within the Reactor Building. Furthermore, it 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. It also includes helium flow rate instrumentation and radioactive plateout monitoring instrumentation.

Additional Basis Sheet(s)

Basis:

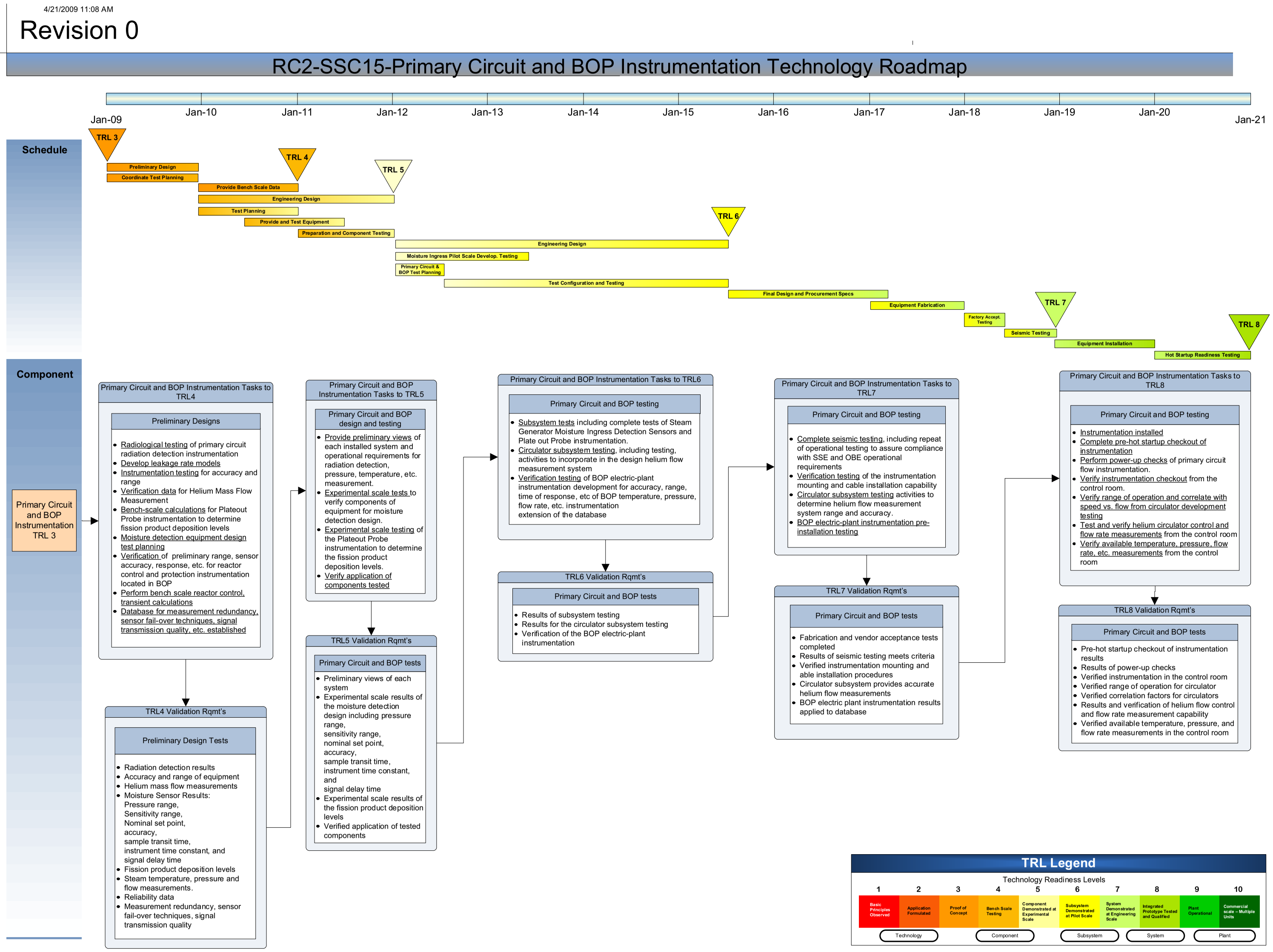
Instrumentation checkouts to confirm hot startup readiness will be completed to achieve a level 8 technical rating. This includes checkout of primary circuit instrumentation, BOP instrumentation, and helium flow rate instrumentation. Advancement to the level 8 technical rating will complete qualification 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.

The 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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Schedule
TRL 3
Preliminary Design
Coordinate Test Planning
TRL 4
Provide Bench Scale Data
Engineering Design
Test Planning
Provide and Test Equipment
Preparation and Component Testing
TRL 5
Engineering Design
Moisture Ingress Pilot Scale Develop. Testing
Primary Circuit & BOP Test Planning
Test Configuration and Testing
TRL 6
Final Design and Procurement Specs
Equipment Fabrication
Factory Accept. Testing
Seismic Testing
TRL 7
Equipment Installation
Hot Startup Readiness Testing
TRL 8
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
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
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
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
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
TRL Legend
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

4.13 RC2-SSC-16 RPS, IPS, and PCDIS

TRL Rating Sheets, TRL 4 through 7

Technology Development Road Map

TRL Rating Sheet

Vendor: GA	**Document Number:** RC2-SSC-16.1	**Revision:** 0

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

Title: Reactor Control and Protection, RPS, IPS AND PCDIS

Description:
This SSC 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. It 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. Lastly, it also includes the necessary testing and qualification to assure reliability and safety with this type of equipment.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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 this SSC 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 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:**	RC2-SSC-16.2	**Revision:**	0

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

Title: Reactor Control and Protection, RPS, IPS AND PCDIS

Description:

This SSC 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. It 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. Lastly, it includes the necessary testing and qualification to assure reliability and safety with this type of equipment.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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 this SSC 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	FD 30-42mo FD 12-42mo FD 12-42mo	400 1,200 500
4. Procure checkout interfaces for development simulator.	GA Vendor 4	FD 30-42mo FD 36-42mo	200 800
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 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.	GA	FD 12-42mo	900
6. Provide reactor control and protection analysis results to confirm final design readiness.	GA	FD 36-42mo	1,000

TRL Rating Sheet

Vendor:	GA	**Document Number:**	RC2-SSC-16.3	**Revision:**	0

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

Title: Reactor Control and Protection, RPS, IPS AND PCDIS

Description:
This SSC 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. It 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. Lastly, it includes the necessary testing and qualification to assure reliability and safety with this type of equipment.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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 this SSC 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 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:**	RC2-SSC-16.4	**Revision:**	0

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

Title: Reactor Control and Protection, RPS, IPS AND PCDIS

Description:
This SSC 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. It 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. Lastly, it includes the necessary testing and qualification to assure reliability and safety with this type of equipment.

Area: ☒ NHSS ☐ HTS ☐ HPS ☐ PCS ☐ BOP

PASSC:	**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 this SSC 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

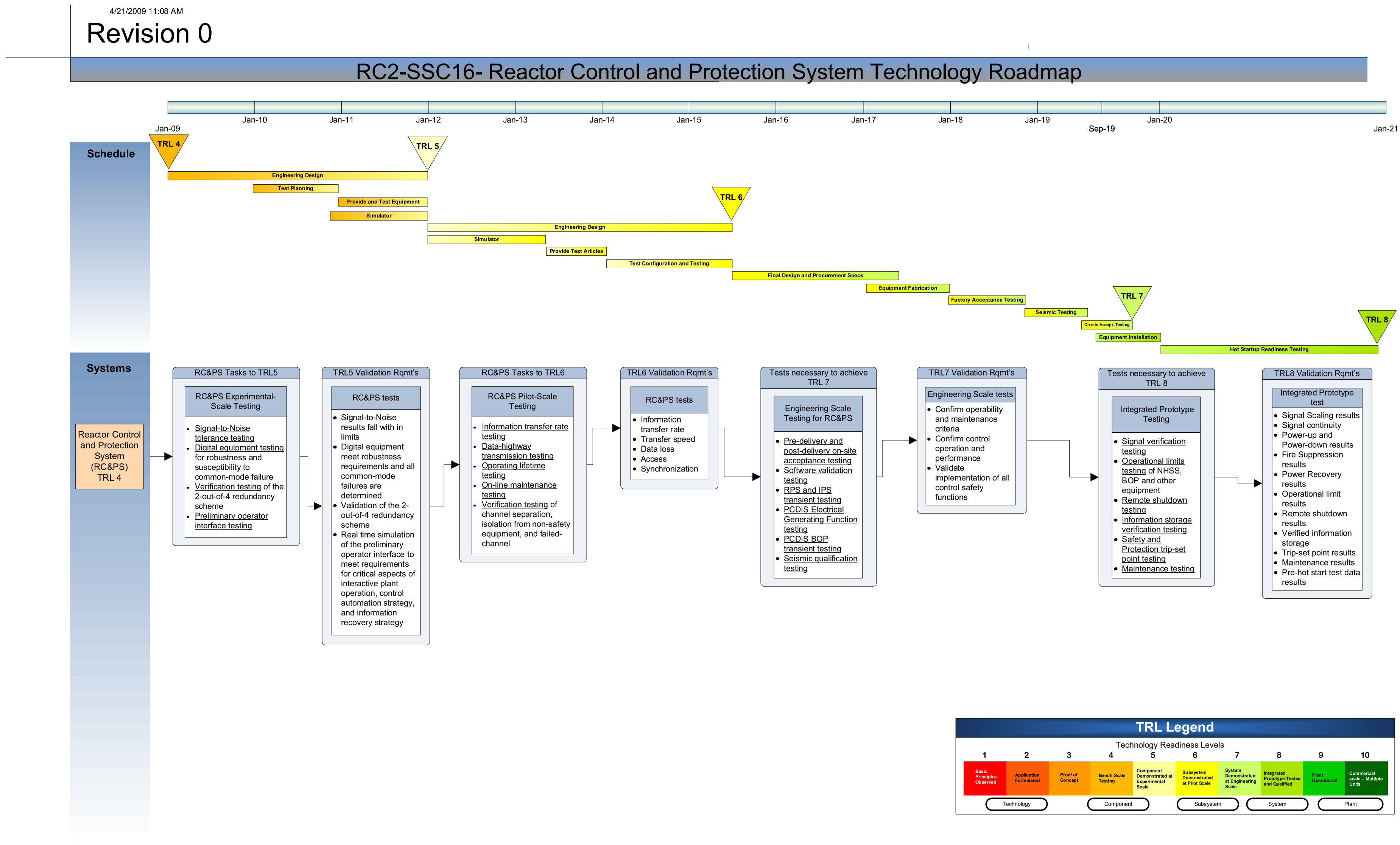
Additional Basis Sheet(s)

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 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 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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Factory Acceptance Testing
Seismic Testing
On-site Accept. Testing
Equipment Installation
Hot Startup Readiness Testing
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
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
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
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
TRL Legend
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

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.

INTEGRATED TEST SCHEDULE-NGNP

ID	SSC	TRL	CTF	Name	dur
1				**NGNP Top Level Schedule**	**167 mo**
2				Common Conceptual Design	10 mo
3				Common Preliminary Design	39 mo
4				NHSS Conceptual and Prelim Design	39 mo
5				Final Design	118 mo
6				Site Work & Construction	92 mo
7				Start-Up & Testing	28 mo
8	1			**Reactor Control Equipment**	**147 mo**
9		4-5		Engineering Design and Test Planning	36 mo
10				Provide and test equipment	6 mo
11				Component test performance	12 mo
12				Component test Monitoring	12 mo
13		5-6		Engineering Design	45 mo
14				Component Test Articles	6 mo
15				Subsystem Test Articles	18 mo
16				Component test Monitoring	6 mo
17			CTF	Subsystem Test Performance	6 mo
18		6-7		Final Design and Procurement	42 mo
19				Equipment fabrication	18 mo
20				Factory Acceptance Testing	9 mo
21				Test Monitoring	6 mo
22				Seismic Testing	6 mo
23				Integrated System Testing	12 mo
24		7-8		Equipment Installation	6 mo
25				Hot Startup Readiness Testing	12 mo
26	2a			**Reactor Internals - Control Rods -Composite**	**163 mo**
27		2-3		Finalize NGNP Point Design	5 mo
28				Develop control rod design requirements	8 mo

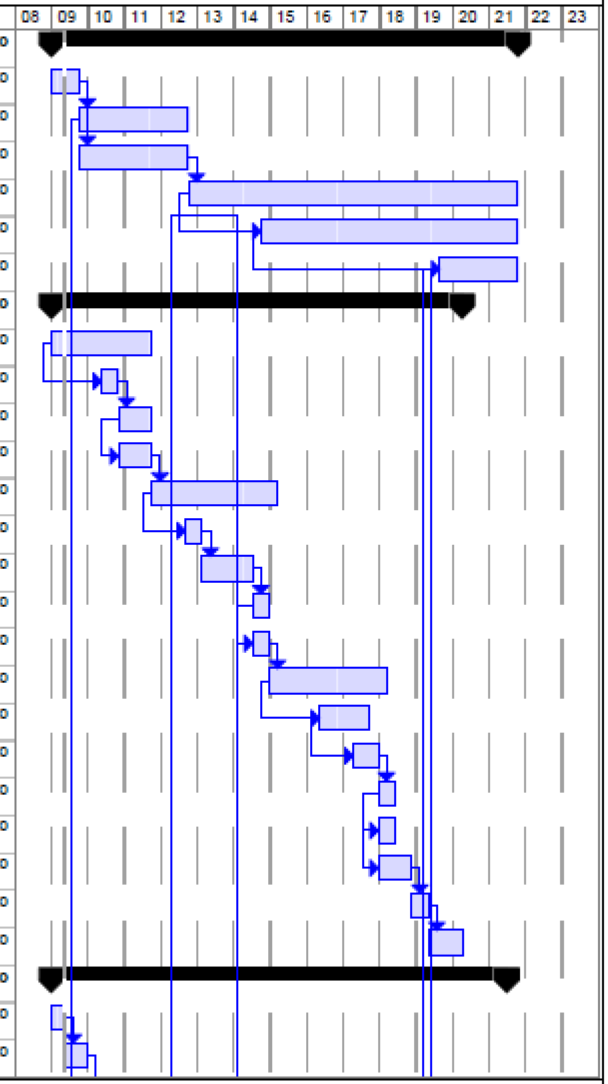

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Figure 5-1. Overall Schedule for NGNP Technology Development

INTEGRATED TEST SCHEDULE-NGNP

ID	SSC	TRL	CTF	Name	dur
29				Develop CR conceptual design	3 mo
30				Review composites knowledge base and supply network;select mfg	6 mo
31				Dvlp comp. architecture, fab. shapes, obtain test samples	13 mo
32				Baseline properties testing	13 mo
33				**Irradiation Testing**	**58.5 mo**
34				Irradiation capsule design	14 mo
35				Irradiation testing	30 mo
36				Post irradiation examination	19 mo
37				Prepare test reports	12 mo
38				Corrosion testing	26 mo
39				Select composite materials and architectures	2.5 mo
40		3-4		Develop final comp. architecture and fab. shapes	6 mo
41				Baseline properties testing for engineering data base	12 mo
42				Irradiation testing for engineering data base	24 mo
43				Corrosion testing for engineering data base	24 mo
44				Complete CR composites materials models	9 mo
45		4-5		Engineering analysis	14 mo
46		5-6		Engineering analysis	10 mo
47		**6-7**		**CR vibration testing**	**18.5 mo**
48				Develop test specification	2.5 mo
49				Fab. prototype control rod	9 mo
50				Set up test facility and prepare test procedures	9 mo
51			CTF	Conduct CR vibration testing and prepare report	7 mo
52				CR shock absorber test	15 mo
53				**CR structural integrity test**	**18 mo**
54				Develop test specification	2 mo
55				Fab. prototype control rod	9 mo
56				Set up test facility and prepare test procedures	9 mo

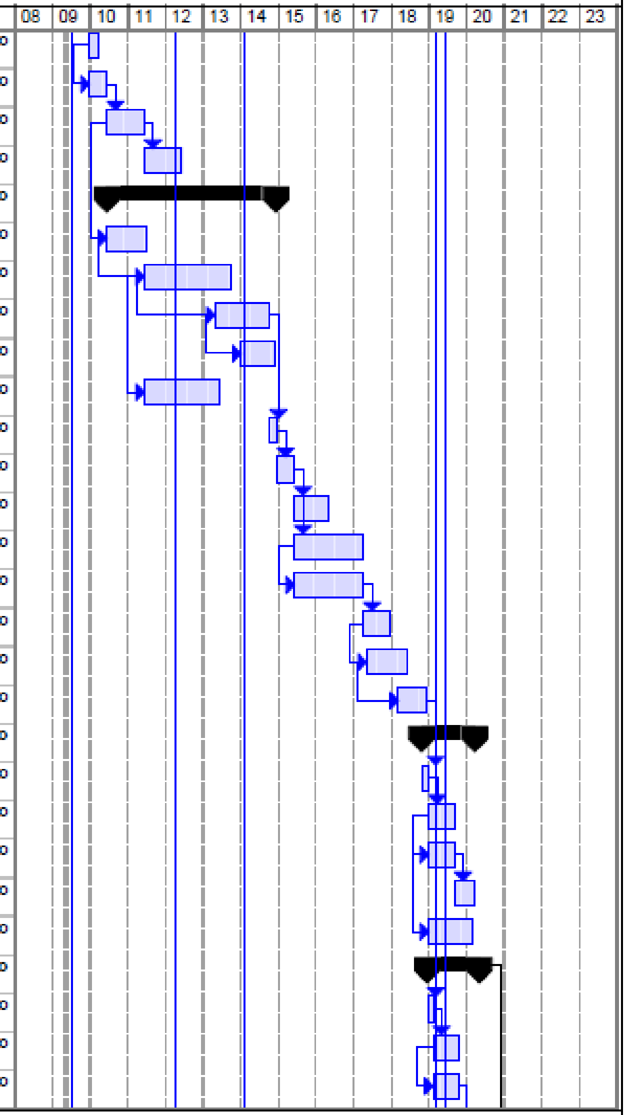

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Figure 5-1. Overall Schedule for NGNP Technology Development (2 of 11)

INTEGRATED TEST SCHEDULE-NGNP

ID	SSC	TRL	CTF	Name	dur
57			CTF	Conduct CR structural integrity test and prepare report	7 mo
58		7-8		Integrated NCA testing at NGNP	15 mo
59	**2a-M**			**Control Rod-Metallics Technology Development**	**150 mo**
60		4-5		Finalize NGNP Point Design	3 mo
61				Develop CR Requirements	6 mo
62				Develop CR Conceptual Design	3 mo
63				Baseline Properties Testing for Engineering Database	12 mo
64				Irradiation Testing for Engineering Database	24 mo
65				Corrosion Testing for Engineering Database	24 mo
66				Complete CR materials models	6 mo
67		5-6		Engineering Analysis 1	12 mo
68				Engineering Analysis 2	9 mo
69		**6-7**		**CR Vibration Testing**	**18 mo**
70				Develop Test specification	3 mo
71				Fabricate Prototype Control Rods	9 mo
72				Setup Test Facility and prepare test procedures	9 mo
73			CTF	Conduct testing and prepare report	6 mo
74				CR Shock Absorber Testing	12 mo
75				**CR Structural Integrity Testing**	**21 mo**
76				Develop Test specification	3 mo
77				Fabricate Prototype Control Rods	12 mo
78				Setup Test Facility and prepare test procedures	9 mo
79			CTF	Conduct testing and prepare report	6 mo
80		7-8		Integrated NCA testing at NGNP	12 mo
81	**2d**			**Upper Core Restraint-Composite Technology Development**	**166 mo**
82		2-3		Finalize NGNP Point Design	3 mo
83				Develop UCR Requirements	6 mo
84				Develop UCR Conceptual Design	3 mo

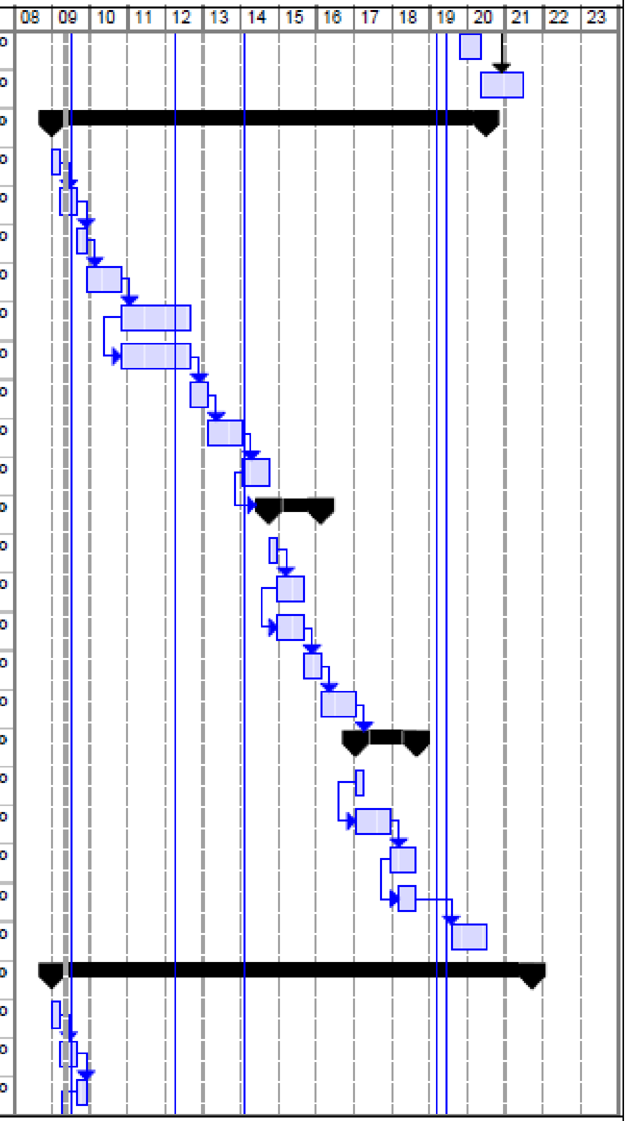

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Figure 5-1. Overall Schedule for NGNP Technology Development (3 of 11)

INTEGRATED TEST SCHEDULE-NGNP

ID	SSC	TRL	CTF	Name	dur
85				Review Composites Knowledge Base & Supply Chain; Select Manufacturers	6 mo
86				Develop Complete Architecture, Fabricate Shapes, Obtain test samples	12 mo
87				Baseline Properties Testing	12 mo
88				**Irradiation Testing**	**55 mo**
89				Irradiation Capsule Design	12 mo
90				Irratiation Testing	24 mo
91				Post Irradiation Examination	18 mo
92				Prepare Test Reports	9 mo
93				Corrosion Testing	24 mo
94				Select Composite Materials and Architectures	3 mo
95		3-4		Develop Final Composite Architecture and Fabricate Shapes	6 mo
96				Baseline Properties Testing for Engineering Database	12 mo
97				Irradiation Testing for Engineering Database	24 mo
98				Corrosion Testing for Engineering Database	24 mo
99				Complete UCR Composites materials models	6 mo
100		4-5		Engineering Analysis	12 mo
101		**5-6**		**Engineering Analysis**	**12 mo**
102				UCR Performance	6 mo
103				Response of CR & RSM guide tubes during normal and non-normal events	6 mo
104		**6-7**		**UCR Vibration Testing**	**18 mo**
105				Develop Test specification	3 mo
106				Fabricate Engineering-Scale UCR elements	9 mo
107				Setup Test Facility and prepare test procedures	9 mo
108			CTF	Conduct testing and prepare report	6 mo
109				**UCR Pressure Drop and Flow Testing**	**18 mo**
110				Develop Test Spec	3 mo
111				Setup Test Facility	9 mo
112			CTF	Conduct testing and prepare Report	6 mo

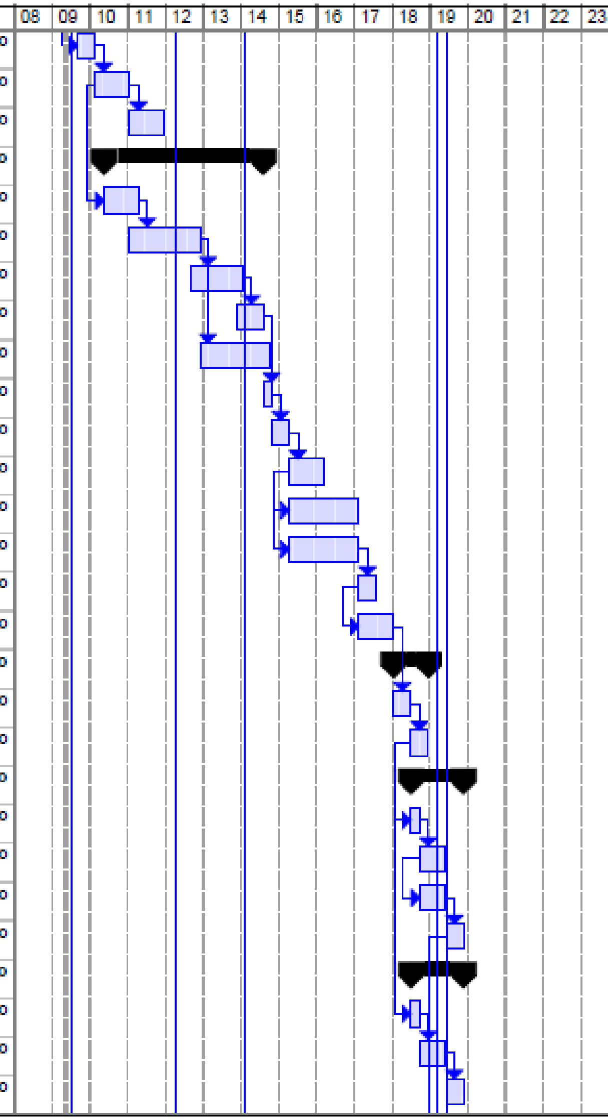

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Figure 5-1. Overall Schedule for NGNP Technology Development (4 of 11)

INTEGRATED TEST SCHEDULE-NGNP

ID	SSC	TRL	CTF	Name	dur
113				**UCR Structural Integrity Testing**	**18 mo**
114				Develop Test specification	3 mo
115				Setup Test Facility and prepare test procedures	9 mo
116			CTF	Conduct testing and prepare report	6 mo
117		7-8		Integrated UCR testing at NGNP	12 mo
118	**2d-M**			**UCR-Metallic Technology Development**	**142 mo**
119		4-5		Finalize NGNP Point Design	3 mo
120				Develop UCR Requirements	6 mo
121				Develop UCR Conceptual Design	3 mo
122				Baseline Properties Testing for Engineering Database	12 mo
123				Irradiation Testing for Engineering Database	24 mo
124				Corrosion Testing for Engineering Database	34 mo
125				Complete UCR materials models	6 mo
126		**5-6**		**Engineering Analysis**	**12 mo**
127				UCR Performance	6 mo
128				Response of CR & RSM guide tubes during normal /non-normal events	6 mo
129		**6-7**		**UCR Pressure Drop and Flow Testing**	**18 mo**
130				Develop Test specification	3 mo
131				Fabricate Prototype Upper Core Restraints	9 mo
132				Setup Test Facility and prepare test procedures	9 mo
133			CTF	Conduct testing and prepare report	6 mo
134				**UCR Vibration Testing**	**18 mo**
135				Develop Test specification	3 mo
136				Setup Test Facility and prepare test procedures	9 mo
137			CTF	Conduct testing and prepare report	6 mo
138				**UCR Structural IntegrityTesting**	**18 mo**
139				Develop Test specification	3 mo
140				Setup Test Facility and prepare test procedures	9 mo

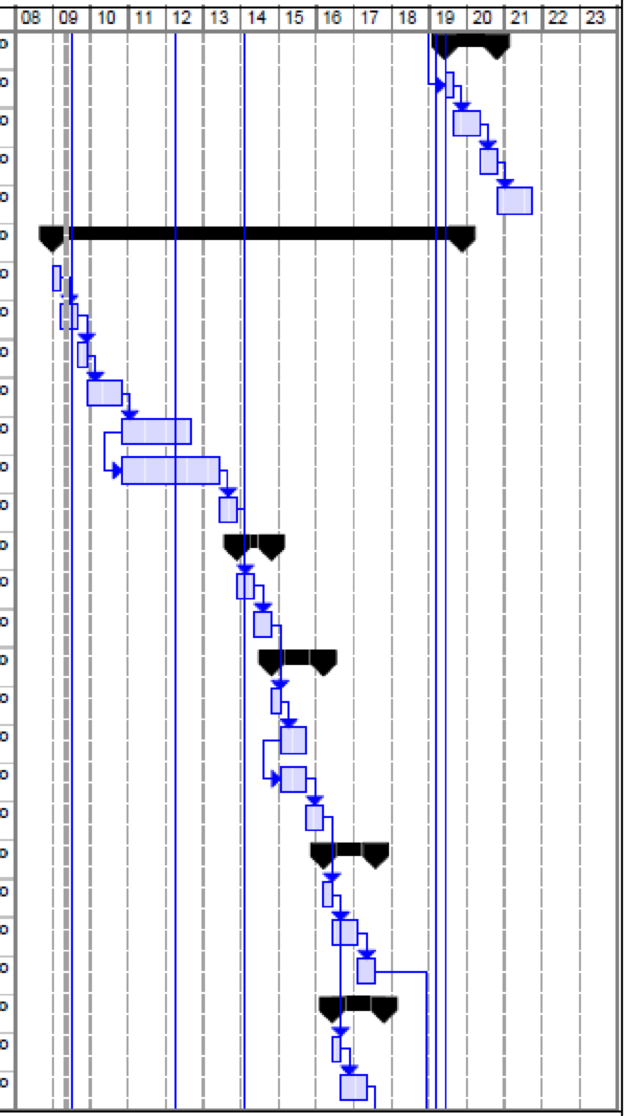

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Figure 5-1. Overall Schedule for NGNP Technology Development (5 of 11)

INTEGRATED TEST SCHEDULE-NGNP

ID	SSC	TRL	CTF	Name	dur
141			CTF	Conduct testing and prepare report	6 mo
142		7-8		Integrated UCR-Metallic elements testing at NGNP	12 mo
143	3			**High Temperature Ducts & Insulation**	**94 mo**
144		2-3		Condition of Service Documented; safety, code and standards, mat'ls	6 mo
145		3-4		Analytical Modeling	12 mo
146		4-5		Component & Coupon testing	12 mo
147		5-6		Subsystem Testing & Analysis	12 mo
148		6-7		Scale System-Level Vessels tests	24 mo
149		7-8	CTF	CTF testing (if required)	24 mo
150	4			**Reactor Core & Graphite**	**112 mo**
151	4a		CTF	**Reactor Core**	**72 mo**
152			CTF	Core Element Dynamic Strength	24 mo
153			CTF	Core Element Failure Modes	24 mo
154			CTF	Fuel Element Flow Channel Data	24 mo
155			CTF	Control Rod Flow Channel Data	24 mo
156			CTF	Core Column Vibration Data	25 mo
157			CTF	Core Crossflow Test Data	25 mo
158			CTF	Core Fluctuation Test Data	25 mo
159			CTF	Bottom Reflector/Core Support Pressure Drop/Flow Mixing Data	24 mo
160			CTF	Metallic Plenum Element & Top Reflector Pressure Drop/Flow Distribution	24 mo
161	4b			**Graphite**	**112 mo**
162				Graphite Coke Source Qualification	26 mo
163				Tests on Unirradiated Samples	54 mo
164				Tests on Irradiated Samples	90 mo
165	5			**Reactor Pressure Vessel**	**156 mo**
166		5-6		Finalize NGNP Point Design NA	6 mo
167				Develop design requirements and conceptual design	10 mo
168				Define SA-508/533 Testing Program	4 mo

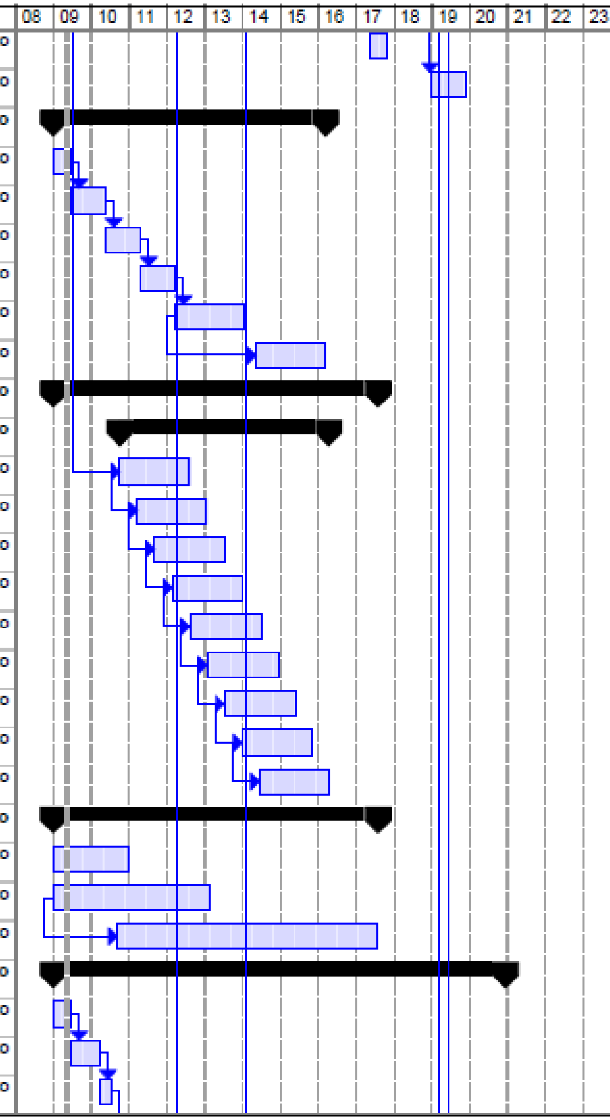

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Figure 5-1. Overall Schedule for NGNP Technology Development (6 of 11)

INTEGRATED TEST SCHEDULE-NGNP

ID	SSC	TRL	CTF	Name	dur
169				SA-508/533 mechanical properties testing	35 mo
170		6-7		Develop detailed RPV design and RPV procurement specs	4 mo
171				Develop detailed RPV design	13 mo
172		7-8		RPV testing in NGNP	12 mo
173	6			**Helium Circulator**	**131 mo**
174				**Identify Magnetic and Catcher Bearing Development Requirements**	**6 mo**
175				Establish Initial Requirements	6 mo
176				**Test Plan for Bearing Development**	**15 mo**
177				Initial Plan	6 mo
178				Spec	3 mo
179				Test Program Cost Estimate	6 mo
180				**Test Article and Facility Design Fabrication and Procurement**	**24 mo**
181				Initial Design and Layout in Support of Cost Estimate	1.5 mo
182				Evaluate International Co-operative Testing Program	10.5 mo
183				Complete Facility and Test Article Design	10.5 mo
184				Complete Test Facility Fabrication Assembly and Shakedown	12 mo
185			CTF	**Bearings Design Verification Testing**	**24 mo**
186				Initial Confirmation of Rotor/Bearings Adequacy	3 mo
187				Initial Verification	6 mo
188				Complete Testing	9 mo
189				Extended Duration Testing or Test of Modified Bearing System (If Necessary)	6 mo
190			CTF	**Circulator Cavity Cooling System Tests**	**12 mo**
191				Complete Simulated Cavity Rig	6 mo
192				Establish Data for System Final Design	6 mo
193			CTF	**Prototype Circulator Design Verification**	**56 mo**
194				Initiate Work on New Facility or Modification of Existing Facility	0 mo
195				Complete Test Facility	12 mo
196				Complete Fabrication of Prototype Unit	7.5 mo

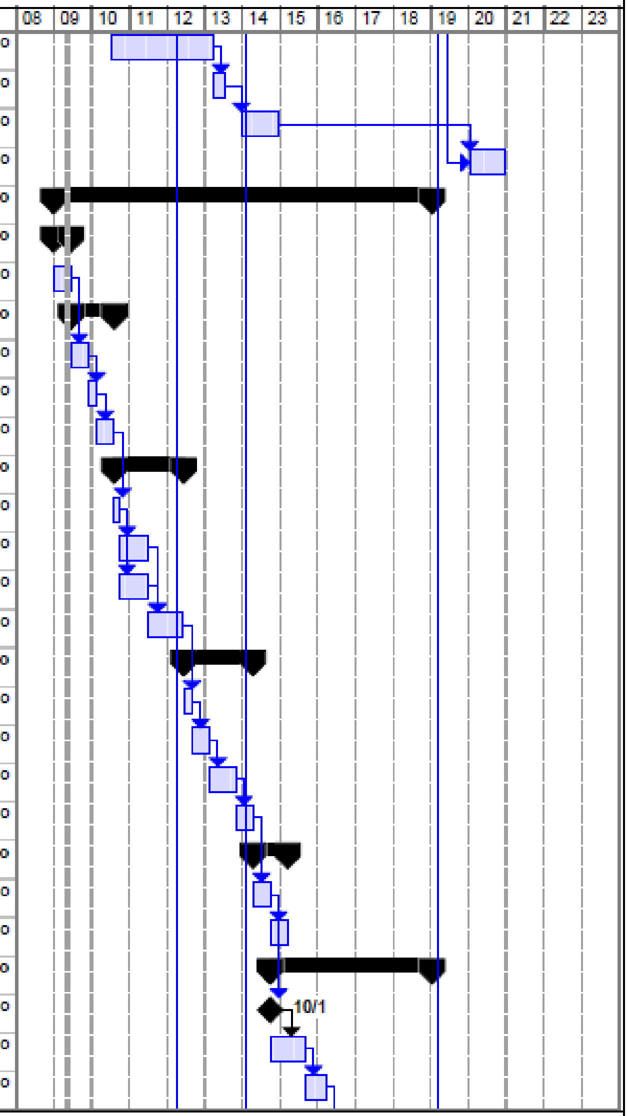

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Figure 5-1. Overall Schedule for NGNP Technology Development (7 of 11)

INTEGRATED TEST SCHEDULE-NGNP

ID	SSC	TRL	CTF	Name	dur
197				Install Prototype in Facility	8 mo
198				Initial Confirmation that Circulator Meets Requirements	3 mo
199				Complete Initial Test Verification	1.5 mo
200				Extended Duration Testing	3 mo
201				Test Modified MC If Necessary	9 mo
202				Proof Test Additional Circulators in Support of First Plant Operation	12 mo
203	8			**Shutdown Cooling System Heat Exchanger**	**130 mo**
204		4-5		**Design & Analysis**	**26 mo**
205				Thermal Design & Analysis of SCHE	13 mo
206				Structural Analysis	13 mo
207				Selection of Shroud Material and Shroud Insulation	13 mo
208		5-6		**Structural & Properties Tests**	**52 mo**
209				Shroud and its Insulation Properties Tests	13 mo
210				Shroud Leak Test	13 mo
211				Vibration and Acoustical Tests	13 mo
212				Related SG testing	13 mo
213				Inlet Flow and/or Temperature Distribution Tests	13 mo
214				Stability Oriface	6 mo
215				SCHE Final Design	26 mo
216				Build a Full Scale SCHE	13 mo
217		6-7	CTF	Airflow Testing for Full-Scale SCHE	13 mo
218		7-8	CTF	Full-Scale SCHE Test in Helium	26 mo
219	9			**Reactor Cavity Cooling System**	**147 mo**
220		4-5		Emissivity of RCCS Panel Surfaces	15 mo
221		5-6		Wind Tunnel Testing of Scale Model RCCS I/O Structure	21 mo
222		6-7		RCCS Cooling Panel Heat Transfer and Friction Factor and Integrated Test	24 mo
223		7-8		Integrated RCCS Performance	24 mo
224	10			**Steam Generator**	**170 mo**

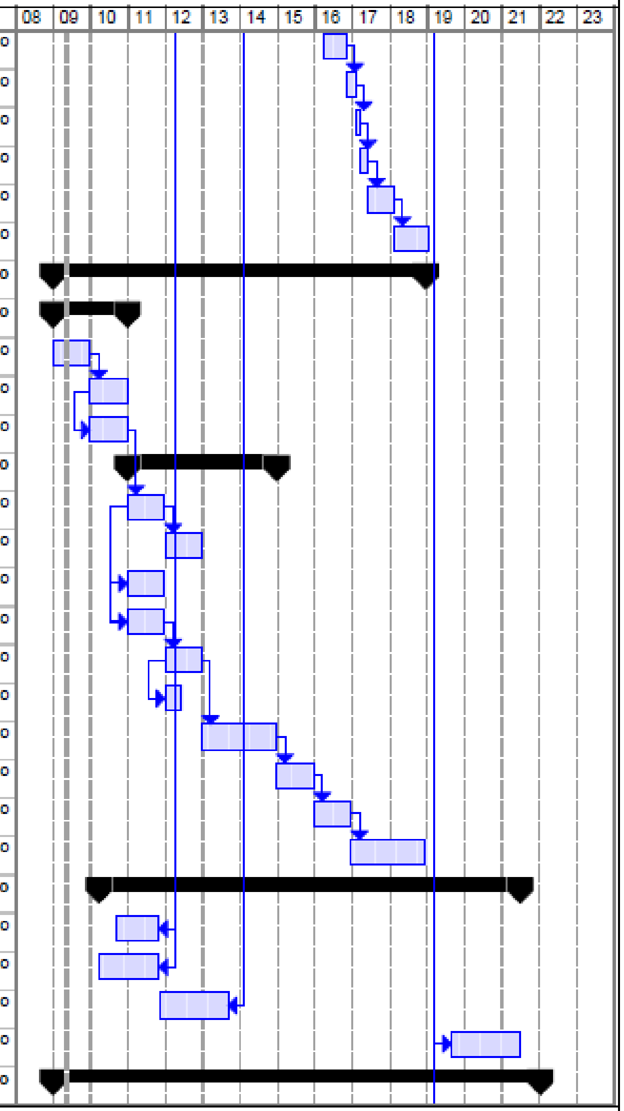

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Figure 5-1. Overall Schedule for NGNP Technology Development (8 of 11)

INTEGRATED TEST SCHEDULE-NGNP

ID	SSC	TRL	CTF	Name	dur
225		4-5		SG Design & Analysis	26 mo
226		5-6		Base Material and bimetallic weld test (750C)	39 mo
227				Air Flow Testing and Oriface testing	13 mo
228				Lead In Lead Out testing	13 mo
229				Tube Support System Test	13 mo
230				Insulation testing	13 mo
231				Large Helical Coil Fabrication testing	13 mo
232		6-7		SG Final Design and Fabrication	52 mo
233				Air Flow Testing	27 mo
234		7-8		Testing in Helium	13 mo
235	**12**			**High Temperature Valves**	**72 mo**
236		4-5		Conditions of service, safety, code & standards	12 mo
237				Analytical Modeling of Components Qualification	12 mo
238		5-6		Pilot-scale component testing	12 mo
239		6-7		Complete Valve Sub-System Integrated Experimental Scale Model Test	12 mo
240		7-8	CTF	Integrated CTF Testing	24 mo
241	**14**			**Fuel Handling and Storage System**	**151 mo**
242		4-5		Prepare conceptual design for NGNP FHSS	22 mo
243				Survey supply network and select vendors	6 mo
244				Complete FHSS preliminary design	20 mo
245				Perform I&C testing	6 mo
246				Perform testing of Fuel Handling Equip. Support Structure	12 mo
247				Perform testing of Fuel Sealing and Inspection Facility	12 mo
248		5-6		Perform testing of Fuel Handling Equipment Positioner	20 mo
249				Perform Testing of Equipment Hoist and Grapple Assembly	20 mo
250				Perform testing of Fuel Handling Machine	20 mo
251				Perform testing of Fuel Transfer Cask	20 mo
252				Complete final design of FHSS and procurement specs	26 mo

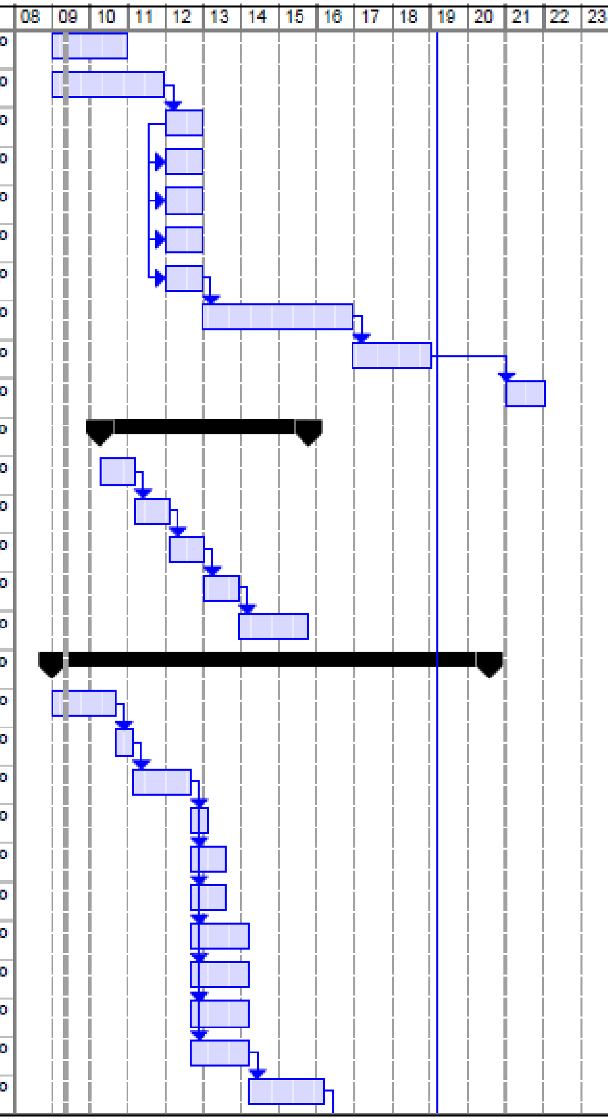

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Figure 5-1. Overall Schedule for NGNP Technology Development (9 of 11)

INTEGRATED TEST SCHEDULE-NGNP

ID	SSC	TRL	CTF	Name	dur
253		6-7		Integrated FHSS	32 mo
254				Prepare test specification	6 mo
255				Fabricate prototype equipment and test rig	14 mo
256				Set up test facility and equipment	9 mo
257				Conduct testing	11 mo
258				Evaluate test data and complete test report	6 mo
259		7-8		FHSS testing in NGNP	12 mo
260	15			Primary Circuit and BOP Protection Instrumentation	156 mo
261		3-4		**Bench-Scale Testing of available Instrumentation**	**26 mo**
262				Engineering Design	12 mo
263				Coordinate Test Planning	12 mo
264				Provide Bench Scale Data	14 mo
265		4-5		**Instrumentation Tests at Component Level**	**27 mo**
266				Engineering Design	27 mo
267				Test Planning	14 mo
268				Provide and Test Equipment	13 mo
269				Preparation and Component Testing	13 mo
270		5-6		**Pilot Scale Testing**	**45 mo**
271				Engineering Design	45 mo
272				Moisture Ingress Pilot Scale Devt Testing	21 mo
273				Primary Circuit & BOP Test Planning	6 mo
274				Test Configuration and Testing	36 mo
275		6-7		**Pre-Installation testing**	**48 mo**
276				Final Design and Procurement Specs	24 mo
277				Equipment Fabrication	13 mo
278				Factory Acceptance Testing	3 mo
279				Seismic Testing	9 mo
280		7-8		**Hot Start-up Readiness testing**	**26 mo**

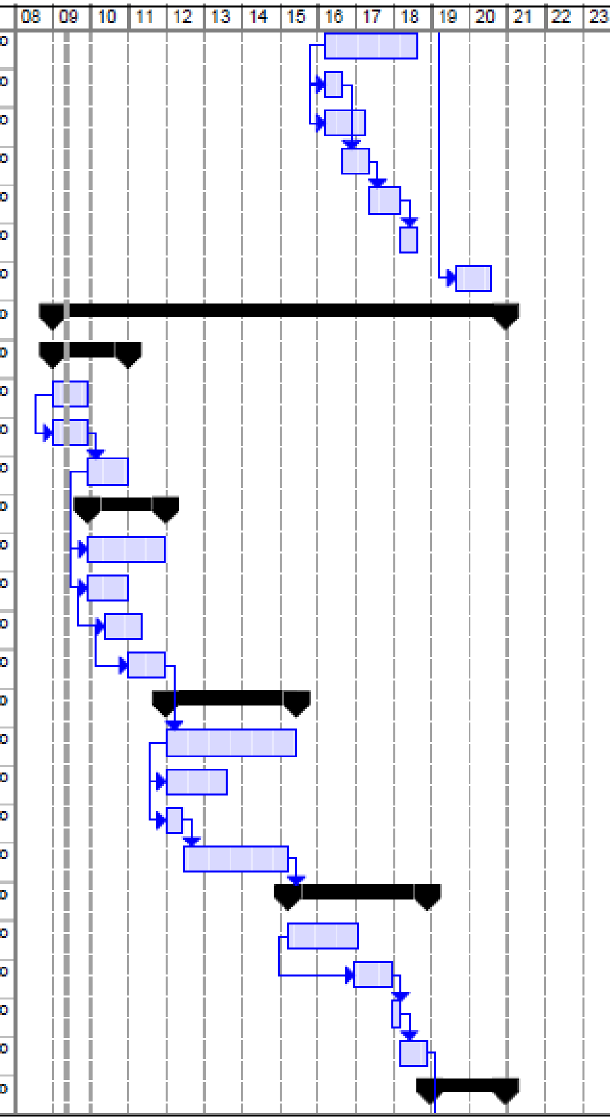

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Figure 5-1. Overall Schedule for NGNP Technology Development (10 of 11)

INTEGRATED TEST SCHEDULE-NGNP

ID	SSC	TRL	CTF	Name	dur
281				Equipment Installation	13 mo
282				Hot Starup Readiness Testing	13 mo
283	16			RPS, IPS, PCDIS	156 mo
284		4-5		Components Engineering Design	39 mo
285				Test Planning	14 mo
286				Provide and Test Equipment	14 mo
287				Simulator	16 mo
288		5-6		Pilot Scale Engineering Design	45 mo
289				Simulator	19 mo
290				Provide Test Articles	6 mo
291				Test Configuration and Testing	21 mo
292		6-7		Pre-Installation Final Design and Procurement Specs	21 mo
293				Equipment Fabrication	13 mo
294				Factory Acceptance Testing	9 mo
295				Seismic Testing	9 mo
296				On-Site Acceptance Testing	3 mo
297		7-8		Hot StartUp Readiness Equipment Installation	5 mo
298				Hot Starup Readiness Testing	13 mo

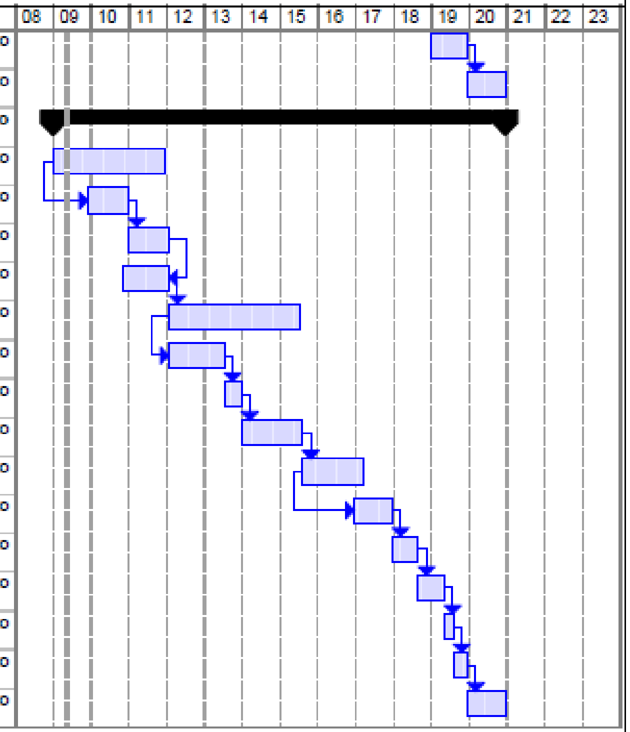

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Figure 5-1. Overall Schedule for NGNP Technology Development (11 of 11)

INTEGRATED TEST SCHEDULE-CTF

ID	SSC	TRL	CTF	Name	dur
8	1			Reactor Control Equipment	147 mo
17			CTF	Subsystem Test Performance	6 mo
26	2a			Reactor Internals - Control Rods -Composite	163 mo
47		6-7		CR vibration testing	18.5 mo
51			CTF	Conduct CR vibration testing and prepare report	7 mo
53				CR structural integrity test	18 mo
57			CTF	Conduct CR structural integrity test and prepare report	7 mo
59	2a-M			Control Rod-Metallics Technology Development	150 mo
69		6-7		CR Vibration Testing	18 mo
73			CTF	Conduct testing and prepare report	6 mo
75				CR Structural Integrity Testing	21 mo
79			CTF	Conduct testing and prepare report	6 mo
81	2d			Upper Core Restraint-Composite Technology Development	166 mo
104		6-7		UCR Vibration Testing	18 mo
108			CTF	Conduct testing and prepare report	6 mo
109				UCR Pressure Drop and Flow Testing	18 mo
112			CTF	Conduct testing and prepare Report	6 mo
113				UCR Structural Integrity Testing	18 mo
116			CTF	Conduct testing and prepare report	6 mo
118	2d-M			UCR-Metallic Technology Development	142 mo
129		6-7		UCR Pressure Drop and Flow Testing	18 mo
133			CTF	Conduct testing and prepare report	6 mo
134				UCR Vibration Testing	18 mo
137			CTF	Conduct testing and prepare report	6 mo
138				UCR Structural Integrity Testing	18 mo
141			CTF	Conduct testing and prepare report	6 mo
143	3			High Temperature Ducts & Insulation	94 mo
149		7-8	CTF	CTF testing (if required)	24 mo
150	4			Reactor Core & Graphite	112 mo
151	4a		CTF	Reactor Core	72 mo
152			CTF	Core Element Dynamic Strength	24 mo
153			CTF	Core Element Failure Modes	24 mo
154			CTF	Fuel Element Flow Channel Data	24 mo
155			CTF	Control Rod Flow Channel Data	24 mo
156			CTF	Core Column Vibration Data	25 mo
157			CTF	Core Crossflow Test Data	25 mo
158			CTF	Core Fluctuation Test Data	25 mo
159			CTF	Bottom Reflector/Core Support Pressure Drop/Flow Mixing Data	24 mo
160			CTF	Metallic Plenum Element & Top Reflector Pressure Drop/Flow Distribution	24 mo
173	6			Helium Circulator	131 mo
185			CTF	Bearings Design Verification Testing	24 mo
190			CTF	Circulator Cavity Cooling System Tests	12 mo
193			CTF	Prototype Circulator Design Verification	56 mo
203	8			Shutdown Cooling System Heat Exchanger	130 mo
217		6-7	CTF	Airflow Testing for Full-Scale SCHE	13 mo
218		7-8	CTF	Full-Scale SCHE Test in Helium	26 mo
235	12			High Temperature Valves	72 mo
240		7-8	CTF	Integrated CTF Testing	24 mo

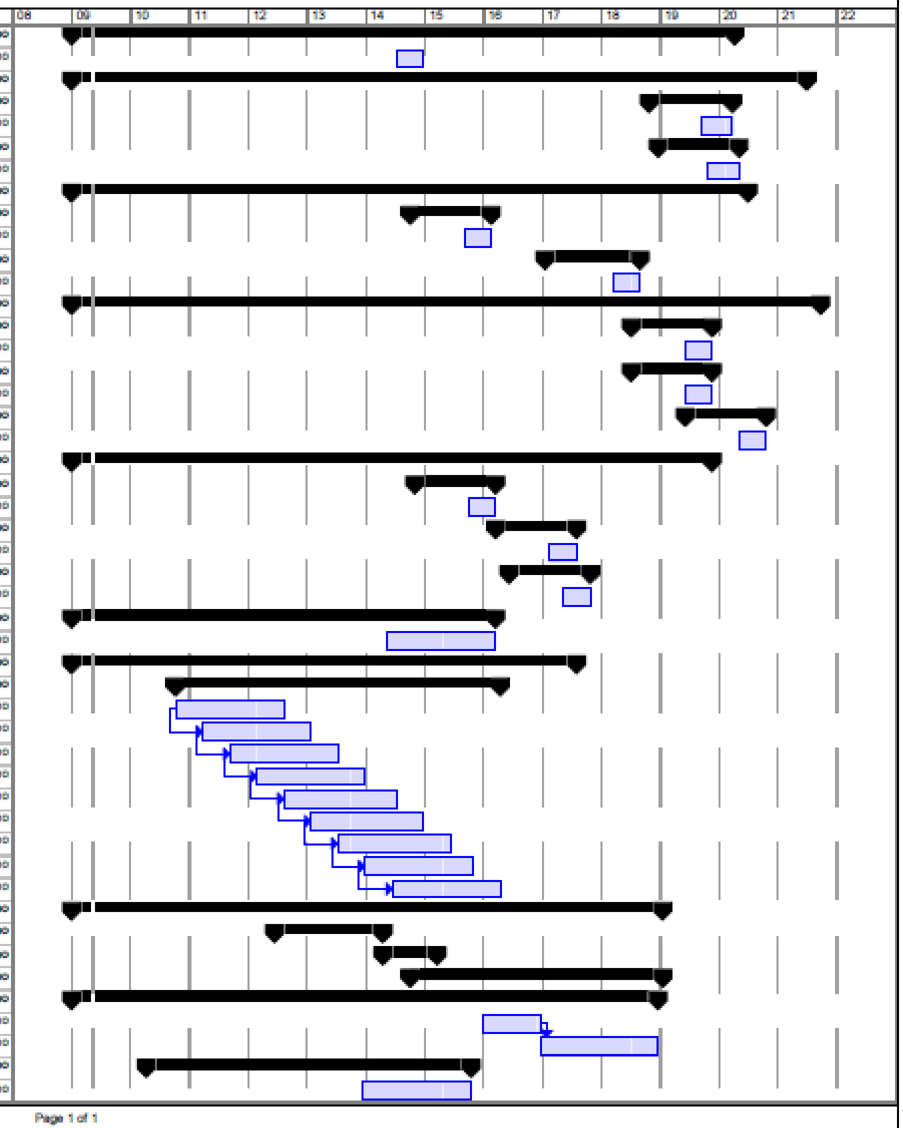

Figure 5-2. Schedule for Potential Testing in CTF

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