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Test Plan – Reactor Pressure Vessel

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ACRONYMS

ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing and Materials
CCD	Conduction Cool Down (event)
CTF	Component Test Facility
DDN	Design Data Need
DOE	U.S. Department of Energy
FSV	Fort Saint Vrain
GA	General Atomics
GT-MHR	Gas Turbine Modular Helium Reactor
HTGR	High-Temperature, Gas-Cooled Reactor
HTTR	High Temperature Test Reactor
INL	Idaho National Laboratory
JAEA	Japan Atomic Energy Agency
KAERI	Korea Atomic Energy Research Institute
LWR	Light Water Reactor
NGNP	Next Generation Nuclear Plant
MHR	Modular Helium Reactor
MHTGR	Modular HTGR
NP-MHTGR	New Production Modular HTGR
ORNL	Oak Ridge National Laboratory
RPV	Reactor Pressure Vessel
TRL	Technology Readiness Level
VCS	Vessel Cooling System

1 INTRODUCTION

1.1 Purpose

This Test Plan provides a high-level description of a test program to support design, fabrication, and deployment of the reactor pressure vessel (RPV) for the Next Generation Nuclear Plant (NGNP). Startup of the NGNP is currently scheduled for 2021, so the test program must be well coordinated with a RPV design and fabrication schedule that supports this startup date.

This Test Plan is applicable to an RPV for the plant configuration shown in Figure 1, which General Atomics (GA) selected as its preferred configuration for the NGNP during the FY08-1 Conceptual Design Studies in early 2008 (GA Report 911120). This plant configuration is consistent with the high-level requirements for the NGNP that existed at that time, and it was selected at the onset of the NGNP technology development roadmapping task under which this Test Plan has been prepared as the basis for the technology development roadmapping effort.¹

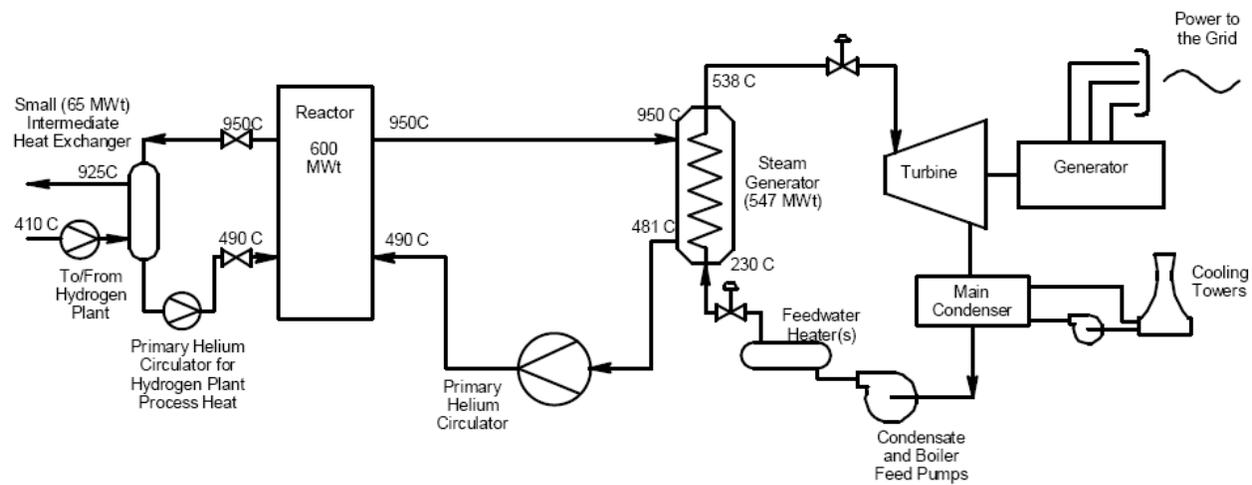


Figure 1. NGNP Configuration for Technology Development Roadmapping

Because the NGNP design process is at a very early stage, adequate design details to precisely define testing requirements are not currently available. Consequently, this Test Plan is intended primarily to identify the design and testing activities that are likely to be needed and to provide cost and schedule estimates, which are based primarily on engineering judgment at this point in

¹ A decision has been made by the NGNP Project to reduce the nominal reactor outlet helium temperature for the NGNP from 950°C into the range of 750°C to 800°C with a corresponding reduction in the reactor inlet helium temperature. However, the technology roadmapping task was started and largely completed while the reactor outlet helium temperature objective for NGNP was still 950°C. Thus, the focus has been to define the technology development activities required for a reactor operating at that temperature.

time. It is assumed that this Test Plan will be updated periodically as the NGNP design progresses and that detailed test plans and test procedures will be prepared by the testing organizations for the specific tests that actually need to be conducted.

1.2 Scope

As discussed in Section 1.3.1, GA has concluded that the RPV for the NGNP must be fabricated from SA508/533 steel, which is the same low-alloy steel used to fabricate pressure vessels for Light Water Reactor (LWRs). 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 required size are within the capabilities of a major forging supplier (Japan Steel Works).

Although there is a large mechanical properties database for SA508/533, there is limited data available on thermal and environmental aging effects on the mechanical properties of this material. In particular, no data is available with respect to the potential long-term effects of the NGNP impure helium environment on the RPV. Consequently, additional data on thermal aging and environmental effects are likely to be needed to support licensing. As discussed in Section 1.3.2, INL and ORNL have also identified creep deformation as a potential concern for the NGNP SA508/533 RPV and have recommended an extensive program of stress-rupture testing to address this concern. This concern derives from the 60-year design lifetime for the RPV and the assumption that the temperature of the RPV will be about 350°C during normal reactor operation.

GA has also concluded that with the NGNP reactor operating with core outlet and inlet helium temperatures of 950°C and 490°C, respectively, a direct vessel cooling system (VCS) will be needed to keep maximum vessel temperatures within ASME code limits for SA508/533 steel with high confidence². Although previous MHR designs have not included a VCS, the VCS is not envisioned to be particularly complex or to require development of new technology.

GA has assigned a technology readiness level (TRL) of 5 to the RPV based on the extensive experience base for SA508/SA533 as the material of construction for current generation LWR RPVs and codification of this material in Section III of the ASME code. GA does not consider long-term creep effects to be a potential problem for the NGNP RPV based on the assumption that the VCS can be designed to keep RPV temperatures well below 350°C during normal reactor operations. Further, although some testing will be needed for confirmation and licensing

² For a reactor operating with a reactor outlet helium temperature in the range of 750°C to 800°C, it is likely that SA508/SA533 can be used as the material of construction for the RPV without a vessel cooling system (VCS). If lower helium temperatures result in RPV temperatures less than 350°C during normal reactor operation (with or without direct vessel cooling), the amount of material properties testing required to qualify SA508/SA533 steel for the NGNP RPV may be substantially reduced.

purposes, GA does not believe that there are likely to be any significant deleterious effects of impure helium on the mechanical properties of the SA508/533 vessel based on the experience with 2.25Cr-1Mo steel in the HTTR.

The required design and testing activities to advance the TRL of the RPV from 5 to 8 are outlined in this Test Plan.

1.3 Background

1.3.1 Basis for Selection of RPV Material

The reactor pressure (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, in-service inspection access, and source-range neutron detectors. For a 600-MWt prismatic modular helium reactor (MHR), the RPV would be larger in diameter (about 7.2-m I.D) than most LWR vessels, but the wall thickness would be comparable.

An evaluation of the potential cost and schedule risks associated with the key operating conditions currently under consideration for the NGNP concluded that the most significant risks are associated with selection of a coolant outlet temperature in excess of approximately 900°C and design choices that would require use of a higher-temperature, high-alloy steel for RPV (GA Report PC-000566). Although use of a high-alloy steel with higher temperature capability would place less burden on optimizing the reactor internal design, these materials pose a significant level of risk because of their very limited experience base for nuclear applications and/or lack of approval in Section III of the ASME code.

Consequently, it was recommended that SA508/533 steel be selected as the material for the NGNP RPV. The primary advantage for SA508/533 steel is its extensive experience base as the material used for current generation LWR RPVs. However, if SA508/533 steel is selected as the RPV material, the NGNP design must ensure the RPV temperatures remain within ASME code limits. Calculations performed by KAERI (GA Report 911118) suggest that active vessel cooling may not be required if the reactor core inlet temperature is limited to 490°C, but the confidence level associated with the calculations was 50% and the RPV temperature margin was small. The small operating temperature margin is a concern because, as discussed in Section 1.3.1, creep effects may need to be considered for an NGNP RPV fabricated from SA508/533 if RPV temperatures are close to the ASME code limit of 371°C and the design lifetime of the RPV is 60 years. Consequently, GA has concluded that a VCS is needed and

that the VCS should be designed to keep maximum vessel operating temperatures well below 371°C.

In principle, an active VCS should not impact the case for passive safety, since a SA508/533 RPV could operate for extended periods with the VCS offline without exceeding RPV damage limits. However, the VCS should be considered an investment protection system and should be designed with a high degree of reliability. During NGNP operation, RPV temperatures can be measured with the VCS online and offline to confirm whether or not a VCS is actually required.

1.3.2 Issue Associated with Use of SA508/533 Low-Alloy Steel

SA508/533 low-alloy steel (forgings and plate, respectively) is approved under Section III of the ASME Boiler and Pressure Vessel code for use in nuclear components. The maximum temperature limit permitted by Subsection NB for Class 1 components is 371°C. Per the code, there is no time limit for the operation of components made of SA508/533 as long as the operating temperatures are maintained below 371°C, the temperature below which creep deformation is negligible. Code Case N-499 was developed to provide rules of construction for SA508/533 and their weldments for short-term temperature excursions above the temperature limit of 371°C. Only Level B (upset), C (emergency), and D (faulted) service events are allowed. Metal temperatures are limited to 427°C during Level B events and 538°C during Level C and D events. The total duration of such temperature excursions is limited to 3,000 hours in the temperature range of 371°C to 427°C and 1,000 hours in the range of 427°C to 538°C. The number of Level C and D events above 427°C is limited to three.

While the ASME code and code cases cover many aspects related to structural integrity, they do not explicitly address issues such as degradation of properties due to service conditions or environment. However, these structural issues are highlighted in the Code and it is the responsibility of a nuclear plant to demonstrate to the NRC that these additional issues are adequately addressed. According to INL Report PLN-2803, recent laboratory data indicate that the LWR coolant environment (primary water) has a significant adverse impact on the fatigue life of reactor structural materials and that fatigue crack growth rates for 300-series stainless steels in LWR environments are 20 times higher than in air, and these issues, which have not been resolved within ASME code committees, have significantly raised the visibility of environmental effects within the NRC, ASME, and the nuclear industry. Thus, it may be necessary for the NGNP Project to proactively collect data to demonstrate that these are not issues for low alloy steels in impure helium environments.

Recently, materials experts at INL and ORNL have evaluated the adequacy of the SA508/533 data base developed for LWRs to assess the adequacy of the data for licensing an NGNP reactor pressure vessel made from this material. Specifically, a re-analysis of the SA-533 database reported in the data package that was used to support the development of Code Case

N-499 was performed. The basis for this re-analysis was a reference NGNP RPV with an assumed normal operating temperature of 350°C. A key issue addressed by this evaluation was the assumption that if the temperature is within the bounds of Subsection NB (371°C for RPV materials), then creep effects do not need to be considered. While this is the case for typical LWR operating temperatures, it may not be true for the reference NGNP operating at the higher 350°C operating temperature and having a 60-year design life. The reason is that creep deformation depends on stress, time and temperature and does not have a strict temperature cut-off that separates creep from non-creep regimes. This could potentially affect the primary stress limits and impact RPV sizing. The potential impact could also likely show up at structural or metallurgical discontinuities. Furthermore, if there is a real problem in the RPV due to creep effects at 350°C, it is not likely to show up until the component is well into its operating life.

Indeed, the results of the analysis indicated that the values of the creep rupture stress given in Code Case N-499 are non-conservative relative to the rupture data at 371°C and point to a potential need to develop longer term confirmatory creep rupture data and to follow-up on the creep-fatigue issue to ensure that creep effects are properly accounted for in design for very long operating lives. The majority of the additional information that is required is related to long term aging behavior at NGNP vessel temperatures, which are somewhat above those commonly encountered in the existing data base for LWR experience. Additional data are also required for the anticipated environment.

Based on the above considerations, INL and ORNL have proposed a RPV materials research and development program (INL Report PLN-2803) to develop additional data for SA508/533 as may be necessary to license an NGNP RPV made from these materials. This plan is based on the assumptions that the NGNP RPV will have a design lifetime of 60 years and that RPV temperatures will be about 350°C during normal reactor operation. The test program recommended in PLN-2803 includes an extensive series of tests to develop the following data:

- Creep-rupture data
 - Long-term creep-rupture tests of both the base metal and weldments at temperatures of 350°C, 371°C, and 390°C to provide some acceleration of the creep process
 - Environmental creep rupture tests to assess the potential impact of NGNP helium on creep-rupture strength
 - Creep rupture tests on samples that have been exposed to strain-controlled cycling to simulate creep-fatigue damage during the short-term higher temperature conditions allowed by Code Case N-499
 - Longer term creep rupture tests in air at 350°C to generate five-year and 20-year data
- Relaxation strength data at 350°C, 371°C, 427°C, 538°C, covering the normal operating temperature and the temperatures permitted in Code Case N-499.
- Creep-fatigue tests at 350°C to assess fatigue-stress relaxation

- Effects on tensile properties
 - Tensile tests at 20°C, 150°C, 250°C, 350°C, 450°C, and 550°C to determine baseline tensile properties, the creep-fatigue damaged condition, and the thermally aged conditions of both the base material and weldments.
 - Two thermal aging protocols, 20,000 hours at 450°C and 70,000 hours at 450°C are proposed.
 - In addition to providing data to assess the potential tensile properties degradation, these tensile data will be needed in the analysis of the fracture toughness data.
- Fracture toughness
- Cyclic stress-strain curve at 20°C, 350°C, 371°C, 427°C, and 538°C

PLN-2803 notes that the ASME Code does not provide detailed guidance in dealing with fracture and that this issue is traditionally handled between the NRC and the nuclear plant “owner”. Fracture toughness has been studied extensively for SA508/533 in LWRs; however, the data may not cover the conditions that the NGNP RPV would encounter because the temperatures of interest for LWR are 300°C and below. If RPV temperatures in NGNP are around 350°C as assumed in PLN-2803, then an issue for the NGNP RPV is the potential negative impact on fracture toughness from long-term thermal embrittlement (thermal aging) accumulated during normal reactor operation for very long time (~60 years) and creep-fatigue damage accumulated during the short-term, high-temperature excursions permitted by Code Case N-499.

2 APPLICABLE DOCUMENTS

Table 1. Documents Applicable to RPV/VCS Technology Development

Document Number	Title	Date
ANL/EXT-06-45	Preliminary Materials Selection Issues for the Next Generation Nuclear Plant Reactor Pressure Vessel	September 2006
GA Report PC-000566	White Paper – Characterizing the Effect of NGNP Operating Conditions on the Uncertainty of Meeting Project Cost and Schedule Objectives	April 2008
HTTR2008 Conf. Paper HTR2008-58281	Creep Effects on Design Below the Temperature Limits of ASME Section III Subsection NB*	October 2008
GA Report 911118	RPV and IHX Pressure Vessel Alternatives Study	April 2008
INL Report PLN-2803	Next Generation Nuclear Plant Reactor Pressure Vessel Materials Research and Development Plan	April 2008
GA Report 911135	Test Plan for the Reactor Core Assembly	December 2008

3 TEST PLAN TO ADVANCE FROM TRL 5 TO TRL 6

A TRL of 6 is achieved when components have been integrated into a subsystem and demonstrated at a pilot scale in a relevant environment. For the RPV, these requirements will be met by a combination of RPV/VCS design and engineering analysis, and materials testing as required to confirm the suitability of SA508/533 as the material of construction for the RPV in the NGNP operating environment. Specifically, the following activities will be conducted to advance the TRL of the RPV from 5 to 6.

1. Develop design requirements for the RPV and for RPV cooling
2. Develop the conceptual design of the vessel cooling system (VCS) and perform analyses to determine the expected operating temperatures for the RPV during normal operation and during conduction cooldown events
3. Define the required materials testing program for SA508/SA533 based on the results of the analyses performed in action 2
4. Conduct the required materials testing program

These activities are discussed in the following sections. An overall schedule and cost estimate summary is provided in Section 6.

3.1 Develop RPV and VCS Design Requirements and VCS Conceptual Design

3.1.1 Activity Description

The first part of this activity is to define the design requirements for the RPV and for RPV cooling. To confirm the need for vessel cooling and to establish the design requirements for the VCS, thermal hydraulic calculations will be performed to assess the sensitivity of vessel temperatures to key parameters, such as emissivities, bypass flow, flow leakage into the core barrel - RPV gap, etc. These analyses will determine the design parameters for the VCS, such as helium flow rate and temperature.

Also, once the point design for the NGNP has been finalized at the beginning of conceptual design, analyses will be performed to confirm that the design helium coolant impurity levels in Table 2 are applicable to the NGNP and to estimate the expected impurity levels specific to the NGNP design and operating conditions³.

³ The design impurity levels were obtained from GA Report 911125 and are applicable to both the MHTGR and GT-MHR. The expected impurity levels are applicable to the GT-MHR. The design impurity levels are considerably higher than the levels that would be expected during equilibrium operating conditions.

Table 2. Design and Expected Levels of Primary Coolant Impurities

Impurity	Design Value	Expected Value*	Units
H ₂ O	2.0	0.5	ppmV
CO ₂	2.0	1.0	ppmV
CO	5.0	2.0	ppmV
H ₂	10.0	3.0	ppmV
CH ₄	2.0	0.1	ppmV
N ₂	10.0	2.0	ppmV
Particulates	10.0	1.0	lb/yr
* For GT-MHR operating at 100% power with T _{in} = 490°C and T _{out} = 850°C			

After the requirements for vessel cooling have been established, the VCS conceptual design will be developed. The primary issue to be addressed is the source of the helium that will be used for direct vessel cooling. In one pre-conceptual VCS concept, the shutdown cooling system is the source of the cold helium. In another, the helium is obtained from the helium purification system. However, GA's current view is that the best approach for NGNP would be to use a dedicated redundant system having its own helium inventory.

The level of the design activity and the supporting thermal-hydraulic calculations will be sufficient to demonstrate a high probability that the VCS will satisfy the vessel cooling requirements. The thermal-hydraulic calculations will provide the expected operating temperatures for the RPV during normal operation and during conduction cooldown (CCD) events. The design activity will also identify and evaluate the operational, regulatory, and investment protection issues associated with vessel cooling.

3.1.2 Test Requirements

This activity does not require any testing. The analyses will be performed by the reactor vendor (assumed to be GA) or GA's designated subcontractor(s) using computer codes that have been verified and validated in accordance with the applicable requirements of ASME NQA-1.

3.1.3 Schedule and Cost

The activity described in Section 3.1.1 is a high-priority conceptual design activity that should start at the beginning of the NGNP conceptual design phase. It is estimated that it will take about nine months to complete this activity. The estimated cost is about \$1M.

3.2 Define Required SA508/533 Testing Program

3.2.1 Activity Description

As discussed in Section 1.3.2, INL and ORNL have proposed a RPV materials research and development program (INL Report PLN-2803) to develop additional data for SA508/533 as may be necessary to license an NGNP RPV made from these materials. This plan is based on the assumptions that the NGNP RPV will have a design lifetime of 60 years and that RPV temperatures will be about 350°C during normal reactor operation. Much of the testing proposed in this plan is to develop stress-rupture data to determine if creep is a potential problem for the NGNP RPV for 60 years of reactor operation with RPV temperatures approaching the ASME Code limit of 371°C. As previously noted in Section 1.2, GA does not consider long-term creep effects to be a potential problem for the NGNP RPV based on the assumption that the VCS can be designed to keep RPV temperatures well below 350°C during normal reactor operations. Further, although some testing will be needed for confirmation and licensing purposes, GA does not believe that there are likely to be any significant deleterious effects of impure helium on the mechanical properties of the SA508/533 RPV based on the experience with 2.25Cr-1Mo steel in the HTTR.

Following completion of the design and engineering analysis activities described in Section 3.1.1, the necessary SA508/533 testing program will be defined based on the calculated RPV temperatures for normal reactor operation and CCD events. It is anticipated that the resultant test program will include some, but not necessarily all, of the elements of the test program proposed in INL Report PLN-2803. The end result of this activity will be a revised NGNP RPV materials research and development plan that is specific to the NGNP RPV conditions established in the RPV/VCS design and analysis activity described in Section 3.1.1

3.2.2 Test Requirements

This activity does not require any testing. The testing program will be developed by the reactor vendor (assumed to be GA) working with materials scientists and materials testing experts at the INL and at ORNL.

3.2.3 Schedule and Cost

The activity described in Section 3.2.1 will start about one year into conceptual design and should take only about four months to complete given the new test program is likely to be a subset of the test program proposed in INL Report PLN-2803. The estimated cost is about \$350K.

3.3 SA508/533 Material Properties Testing

3.3.1 Test Objective

Obtain the data needed to confirm the suitability of SA508/533 steel for use as the material of construction for the NGNP RPV and to support licensing of the NGNP RPV.

3.3.2 Test Description

Details of the testing proposed by INL and ORNL to support the use of SA508/533 for the NGNP RPV are provided in Appendix A of PLN-2803. Section 6 of PLN-2803 provides a summary of the testing program. As discussed in Section 3.2.1, some of this testing may not be necessary if RPV temperatures are significantly lower than 350°C during normal operation, which GA believes will be the case with direct cooling of the RPV. The actual tests to be performed will be defined in the revised NGNP RPV materials research and development plan (see Section 3.2.1) that will be specific to the NGNP RPV conditions established in the RPV/VCS design and analysis activity described in Section 3.1.1. It is anticipated that as a minimum it will be necessary to perform tests of the type defined in Tables A3 and A4 of PLN-2803 to assess the potential impact of NGNP helium on creep-rupture strength, and tests of the type defined in Tables A5 and A6 and Tables A15 through A22 to determine the effect of thermal aging and creep-fatigue damage accumulated during the short-term temperature excursions permitted by Code Case N-499 on creep-rupture strength and tensile properties.

3.3.3 Test Conditions

The test conditions will be as given in PLN-2803. The mechanical properties measurements will be performed at temperatures from 20°C to 550°C in air with the exception of a series of stress-rupture tests that will be performed in impure helium. As appropriate for the specific tests, the test specimens will be thermally aged or treated to simulate in-reactor creep-fatigue damage as described in PLN-2803.

3.3.4 Test Configuration

The various mechanical properties (e.g., stress-rupture, relaxation strength, tensile properties, creep-fatigue, etc.) tests will be performed on standard test specimens using standard laboratory equipment. The 20-year creep-rupture tests (if performed) will need to be performed in a dedicated Long Term Aging Facility. Long-term thermal aging would also best be conducted in a dedicated Long Term Aging Facility.

3.3.5 Required Measurements

Details of the testing proposed by INL and ORNL to support the use of SA508/533 for the NGNP RPV are provided in Appendix A of PLN-2803. Section 6 of PLN-2803 provides a

summary of the testing program. The data to be potentially obtained is listed in Section 1.3. The actual tests to be performed will be defined in the revised NGNP RPV materials research and development plan (see Section 3.2.1) that will be specific to the NGNP RPV conditions established in the RPV/VCS design and analysis activity described in Section 3.1.1.

3.3.6 Test Location

The various mechanical properties (e.g., stress-rupture, relaxation strength, tensile properties, creep-fatigue, etc.), with the exception of the stress-rupture tests in an impure helium environment will be performed using standard laboratory equipment. These tests could be performed at the INL and ORNL, or at any of a number of commercial laboratories that are qualified to perform such testing. Two commercial laboratories that have extensive capability to perform stress-rupture and creep testing over a range of temperatures in accordance with ASTM standard methods include:

IMR Test Labs

Portland, Oregon

503-653-2904

E-mail contact: Bob Adrian at bob@khametal.com

Westmoreland Mechanical Testing & Research, Inc.

Youngstown, Ohio

724-537-3131

E-mail contact: admin@wmtr.com

Depending on the number of samples to be tested, it may be necessary to perform tests at both of the National Laboratories and at a number of commercial testing facilities to complete the tests in a reasonable time frame. The 20-year creep-rupture tests (if performed) will need to be performed in a dedicated Long Term Aging Facility. Long-term thermal aging would also best be conducted in a dedicated Long Term Aging Facility. It is assumed that this facility would most likely be located at ORNL.

3.3.7 Data Requirements

All work performed to support the NGNP R&D Program must be in accordance with The Next Generation Nuclear Plan (NGNP) Quality Assurance Program, INEEL/EXT-04-01776. This program invokes the national consensus standard ASME NQA 1997, "QA Program Requirements for Nuclear Facilities Applications," and Subpart 4.2 of ASME NQA 2000, "Guidance on Graded Application of Quality Assurance (QA) for Nuclear-Related Research and Development."

3.3.8 Test Evaluation Criteria

The conditions for successful completion of the SA508/533 material properties testing program are: (1) the required measurements as identified in Section 3.3.5 have been completed, (2) the data satisfies the data quality requirements as defined in Section 3.3.7, and (3) the data confirm that SA508/533 steel is a suitable material for the NGNP RPV.

3.3.9 Test Deliverables

A final Test Report shall be provided which includes:

- Detailed discussion of test method
- Equipment employed
- Equipment calibration verification
- Detailed test procedures
- Original test data
- Summarized and reduced test data
- A detailed discussion of test results, observations, and calculations that were completed throughout the course of testing.

3.3.10 Schedule, Cost, and Risk

3.3.10.1 Schedule

With the exception of the very-long-term stress rupture tests and the thermal aging tests, it should be possible to complete the tests in about 2.5 years if adequate testing capacity is available (see Section 3.3.6).

3.3.10.2 Cost

The estimated overall cost of the entire test program proposed in PLN-2803 is \$18M, including the cost of raw material, test equipment, analysis and reporting, ASME code interface, QA, and Program Management. The estimated cost just for sample preparation (excluding raw material cost) and testing is \$6M. Based on the assumption that much of this testing will not be necessary, the actual cost of the SA508/533 testing program should be considerably less than \$18M. The cost is roughly estimated to be about half of the estimated cost in PLN-2803, or about \$9M.

3.3.10.3 Risk

Failure to perform material tests could lead to the following:

- Inability to obtain NRC acceptance of the RPV
- Premature failure of the RPV in the NGNP

4 TEST PLAN TO ADVANCE FROM TRL 6 TO TRL 7

To achieve a TRL rating of 7, the system must complete integrated engineering-scale demonstration in a relevant environment. For the RPV, this requirement will be met by a combination of RPV/VCS engineering design and analysis, and engineering-scale testing in a relevant environment. Specifically, the following activities will be conducted to advance the TRL of the RPV from 6 to 7.

1. Develop the design of the RPV and prepare procurement specifications
2. Develop the detailed design of the RPV/VCS system, refine the RPV/VCS computer model, and perform analyses to verify that maximum vessel temperatures will be within ASME code limits with adequate margin
3. Conduct design verification testing of an engineering-scale model of the vessel and vessel cooling system

These activities are discussed in the following sections. An overall schedule and cost estimate summary is provided in Section 6.

4.1 Develop RPV Design and Prepare Procurement Specifications

4.1.1 Activity Description

Evaluate and confirm the previously developed design requirements for the RPV based on the results of the mechanical properties testing of SA508/533 (that was performed to advance the TRL from 5 to 6). Complete the RPV design, including the wall thicknesses of the RPV components required to meet ASME code stress allowables with adequate margin. Prepare and issue the procurement specifications for the RPV.

4.1.2 Test Requirements

This activity does not require any testing. The design work and analyses will be performed by the reactor vendor (assumed to be GA) or GA's designated subcontractor(s) using computer codes that have been verified and validated in accordance with the applicable requirements of ASME NQA-1. The procurement specifications will be prepared and issued in accordance with the QA Program requirements applicable to document and procurement control.

4.1.3 Schedule and Cost

The activity described in Section 4.1.1 will start early in final design when the results are available from the SA508/533 testing program. It is estimated that it will take about six months to complete this activity. The estimated cost is about \$350K.

4.2 Complete RPV/VCS system design

4.2.1 Activity Description

Update the computer model of the RPV/VCS system based on the results of the mechanical properties testing of SA508/533 (that was performed to advance the TRL from 5 to 6) and to incorporate the details of the reactor core developed during preliminary and final design of the reactor core and the reactor internals. Develop the detailed design of the RPV/VCS and verify that the design satisfies all ASME code rules for the RPV. Perform analyses to verify that the maximum RPV temperatures will be within ASME code limits for SA508/533 with adequate margin to eliminate any concerns about creep effects over a 60-year lifetime.

4.2.2 Test Requirements

This activity does not require any testing. The analyses will be performed by the reactor vendor (assumed to be GA) or GA's designated subcontractor(s) using computer codes that have been verified and validated in accordance with the applicable requirements of ASME NQA-1.

4.2.3 Schedule and Cost

The activity described in Section 4.2.1 will start about 1.5 years into NGNP final design when the results are available from the SA508/533 testing program and detailed design information is available for the reactor core and reactor internals. It is estimated that it will take about one year to complete this activity. The estimated cost is about \$700K.

4.3 Engineering-Scale Test of RPV/VCS

4.3.1 Test Objective

Conduct design verification testing of the RPV/VCS system using an engineering-scale model of the reactor core and the RPV/VCS.

4.3.2 Test Description

It is anticipated that design verification testing of the RPV/VCS system will be performed concurrently with design verification testing of the reactor core. A number of engineering-scale tests, including a core fluctuation test, a bottom reflector/core support pressure drop and flow mixing test, and a metallic plenum element and top reflector pressure drop and flow distribution test have been defined for the reactor core in the Reactor Core Test Plan (GA Report 911135).

Based on their experience with the HTTR in Japan, JAEA has confirmed to GA the need for an engineering scale demonstration of the RPV/VCS system and has made the following specific recommendations to concerning the test.

- The reactor core and reactor internals designs should be confirmed by testing of a large-scale model so that differential pressure does not cause unexpected bypass flow and temperature rise of the RPV
- Analysis and testing of the flow distribution needed to cool the RPV, including the control rod standpipes, the core restraint mechanism, and the core support plates are necessary
- Evaluation of thermal mixing of coolant in the core plenums located above and below the core should be performed to prevent non-uniform temperatures or hot streaks in the RPV walls

It is anticipated that the engineering-scale model will include a heat source and will simulate the vessel, core barrel, permanent side reflector, and the upper and lower plenums to the extent necessary to precisely represent the flow path(s) for direct vessel cooling and the potential paths for in-leakage of primary coolant into the direct vessel cooling flow path(s). The engineering-scale model will include the necessary instrumentation to record vessel temperatures and helium flow rates during the test. The first part of this activity will be to design the test and to prepare the Test Specification.

4.3.3 Test Conditions

The reactor vendor will specify the requirements for the test, including the test conditions, in a Test Specification. The testing organization will prepare a test plan and a test procedure that are responsive to the test requirements provided in the Test Specification.

4.3.4 Test Configuration

It is anticipated that design verification testing of the RPV/VCS system will be performed concurrently with design verification testing of the reactor core. A number of engineering-scale tests, including a core fluctuation test, a bottom reflector/core support pressure drop and flow mixing test, and a metallic plenum element and top reflector pressure drop and flow distribution test have been defined for the reactor core in the Reactor Core Assembly Test Plan (GA Report 911135). The RPC/VCS design verification test might best be conducted as an adaptation of the proposed 102-column core fluctuation test that is described as follows in GA Report 911135.

“The flow and thermal conditions in the reactor core will be simulated in the models using air at ambient conditions at the inlet and electrically heated fuel elements to simulate nuclear heating in the core. All the models proposed will dimensionally represent the reactor fuel column at ¼ scale. This scale is small enough to permit easy and economical fabrication, assembly, and handling of the models, yet it is large enough so that test results which are greatly affected by gap dimensions and block distortions are not being influenced by manufacturing tolerances. The models will be made of materials having properties which can simulate the thermal distortion of the columns with small temperature variations.”

Clearly this will be a very complex test, and designing the test is far beyond the scope of the current technology roadmapping effort. The first part of the engineering-scale test activity will be to design the test and to prepare the Test Specification.

4.3.5 Measurements

The engineering-scale model will include the necessary instrumentation to record vessel temperatures and VCS flow rates during the test.

4.3.6 Test Location

The engineering-scale test of the RPV/VCS could be performed in the Component Test Facility (CTF) at the INL, assuming that it is built. Alternately, the test could potentially be performed at GA or at either the INL or ORNL. Another possible test location is Wyle Laboratories. Information for Wyle Laboratories is as follows.

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El Segundo, Ca 90245
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Wyle Laboratories is headquartered in El Segundo, Calif. and employs approximately 4,200 employees at more than 40 facilities nationwide. Wyle is one of the nation's leading providers of specialized engineering, scientific, and technical services to the Department of Defense, NASA, and a variety of commercial customers. Wyle has been designing and building unique test fixtures, equipment and entire test facilities for industry and government use for more than 50 years. These facilities include centrifugal and linear accelerators, vibration systems with up to six axes of motion, high intensity acoustic chambers, dynamic shock devices like crash barriers, plus rail dynamics test facilities and numerous combined-environment test systems. In the nuclear sector, Wyle has qualified more equipment than anyone else in the industry.

4.3.7 Data Requirements

All work performed to support the NGNP R&D Program must be in accordance with The Next Generation Nuclear Plan (NGNP) Quality Assurance Program, INEEL/EXT-04-01776. This program invokes the national consensus standard ASME NQA 1997, "QA Program Requirements for Nuclear Facilities Applications," and Subpart 4.2 of ASME NQA 2000, "Guidance on Graded Application of Quality Assurance (QA) for Nuclear-Related Research and Development."

4.3.8 Test Evaluation Criteria

The conditions for successful completion of the engineering-scale RPV/VCS are: (1) the required measurements as identified in Section 4.3.5 have been completed, (2) the data satisfies the data quality requirements as defined in Section 4.3.7, and (3) the data confirm that the VCS is adequate to maintain RPV temperatures within the ASME code limits for SA508/533.

4.3.9 Test Deliverables

A final Test Report shall be provided which includes:

- Detailed discussion of test method
- Equipment employed
- Equipment calibration verification
- Detailed test procedures
- Original test data
- Summarized and reduced test data
- A detailed discussion of test results, observations, and calculations that were completed throughout the course of testing.

4.3.10 Schedule, Cost, and Risk

4.3.10.1 Schedule

The estimated duration of the test is two years, including preparation of the test specification, fabrication of the engineering-scale model, conduct of the test, evaluation of the test data, and preparation of the test report. The test data are needed about two years before the start of NGNP startup testing.

4.3.10.2 Cost

The estimated cost of the core fluctuation test is \$5M. Expansion of the test to include design verification testing of the RPV/VHS should have a moderate impact on the overall cost of the test. The incremental cost is estimated to be about \$1M.

4.3.10.3 Risk

Failure to conduct the testing to verify the design of the RPV/VHS could result in higher-than-expected RPV temperatures during reactor operation. Should this occur, limitations on reactor operation might be necessary, which could significantly impact the ability of the NGNP to satisfy mission requirements. A potential alternative would be to conservatively design the VCS to provide more RPV cooling in the event that RPV temperatures are higher than expected.

5 TEST PLAN TO ADVANCE FROM TRL 7 TO TRL 8

A TRL of 8 is achieved by demonstrating an integrated prototype of the system in its operational environment with the appropriate number and duration of tests and at the required levels of test rigor and quality assurance. All NGNP systems, structures, and components must have a TRL of 8 as a prerequisite for hot startup of the NGNP. TRL 8 will be achieved for the RPV/VCS system by performing tests of the vessel cooling system in the NGNP during NGNP start-up testing.

5.1 Test Objective

Conduct the appropriate number and duration of tests of the vessel cooling system in the actual operating environment (i.e., in the NGNP) to verify that the system meets reliability requirements and maintains vessel temperatures within the limits defined in the NGNP technical specifications.

5.2 Test Description

TRL 8 will be achieved for the RPV/VCS system by performing tests of the vessel cooling system in the NGNP during NGNP start-up testing. This can presumably be accomplished before hot startup of the reactor by heating the primary coolant and RPV by heat input from the primary helium circulators as was done in the HTTR. The first part of this activity will be to prepare the Test Specification (or alternately to define the test in the NGNP startup plan).

5.3 Test Conditions

The test conditions will be defined in the Test Specification or the NGNP startup plan.

5.4 Test Configuration

The tests will be performed on the as-installed RPV/VCS system in the NGNP.

5.5 Measurements

The NGNP will include the necessary instrumentation to record vessel temperatures and VCS flow rates the tests and during reactor operation.

5.6 Test Location

The tests will ne performed in the NGNP.

5.7 Test Evaluation Criteria

The test evaluation and acceptance criteria will be defined in the Test Specification or the NGNP startup plan.

5.8 Test Deliverables

It is assumed that the test results will be included in the NGNP startup testing report. The information to be provided for the test includes:

- Detailed discussion of test method
- Equipment employed
- Equipment calibration verification
- Detailed test procedures
- Original test data
- Summarized and reduced test data
- A detailed discussion of test results, observations, and calculations that were completed throughout the course of testing

5.9 Schedule, Cost, and Risks

5.9.1 Schedule

This test will be performed as part of NGNP startup testing.

5.9.2 Cost

The incremental cost of this test on the overall cost of the NGNP startup program should be negligible.

5.9.3 Risks

Not applicable.

6 COST AND SCHEDULE SUMMARY

Figure 2 provides an integrated schedule for RPV/VCS system technology development and design, and a summary of the estimated costs.



GENERAL ATOMICS

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