INL/EXT-11-21545

Summary of Planned Implementation for the HTGR Lessons Learned Applicable to the NGNP

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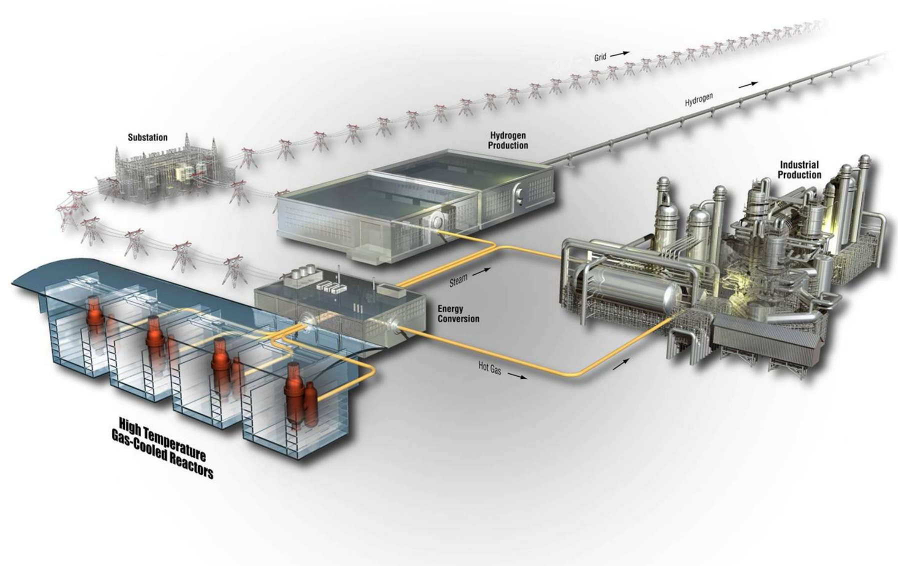

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Next Generation Nuclear Plant Project

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ABSTRACT

This document presents a reconciliation of the lessons learned during a 2010 comprehensive evaluation of pertinent lessons learned from past and present high temperature gas-cooled reactors that apply to the Next Generation Nuclear Plant Project along with current and planned activities. The data used are from the latest Idaho National Laboratory research and development plans, the conceptual design report from General Atomics, and the pebble bed reactor technology readiness study from AREVA. Only those lessons related to the structures, systems, and components of the Next Generation Nuclear Plant (NGNP), as documented in the recently updated lessons learned report are addressed. These reconciliations are ordered according to plant area, followed by the affected system, subsystem, or component; lesson learned; and finally an NGNP implementation statement. This report (1) provides cross references to the original lessons learned document, (2) describes the lesson learned, (3) provides the current NGNP implementation status with design data needs associated with the lesson learned, (4) identifies the research and development being performed related to the lesson learned, and (5) summarizes with a status of how the lesson learned has been addressed by the NGNP Project.

CONTENTS

ACRONYMS

AGC	Advanced Graphite Creep
AGR	Advanced Gas Reactor (INL)
AGR	Advanced Gas-cooled Reactor (United Kingdom)
AMB	active magnetic bearing
ASME	American Society of Mechanical Engineers
AVR	Arbeitsgemeinschaft Versuchsreaktor (Germany)
BPV	Boiler and Pressure Vessel
CRD	Control Rod Drive
DDN	Design Data Need
DTF	Design To Fail
FSV	Fort St. Vrain
GA	General Atomics
HTGR	high temperature gas-cooled reactor
HTR	high temperature reactor
HTR-10	High Temperature Reactor (People's Republic of China)
HTTR	High Temperature Engineering Test Reactor (Japan)
IHX	intermediate heat exchanger
INL	Idaho National Laboratory
JAEA	Japan Atomic Energy Agency
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
PBR	pebble bed reactor
PIE	post-irradiation examination
R&D	research and development
RCCS	reactor cavity cooling system
RPV	reactor pressure vessel
RSS	reserve shutdown system
SSC	structures, systems, and components
THTR	Thorium Hochtemperatur Reaktor (Germany)
TRISO	tristructural isotropic

Summary of Planned Implementation for the HTGR Lessons Learned Applicable to the NGNP

1. INTRODUCTION

In FY 2010, the Next Generation Nuclear Plant (NGNP) Project developed a comprehensive evaluation of the pertinent lessons learned from past and present high temperature gas-cooled reactors (HTGRs) that apply to the NGNP Project and captured those lessons learned in the report *High Temperature Gas-cooled Reactors Lessons Learned Applicable to the Next Generation Nuclear Plant*[1]. A subsequent effort was undertaken to evaluate the current and planned NGNP Project activities that would apply those lessons learned that were documented for projected plant structures, systems, and components (SSC). The results of that evaluation are documented in this report. The data used are from the latest Idaho National Laboratory (INL) research and development (R&D) plans, the conceptual design report from General Atomics (GA), the pebble bed reactor (PBR) technology readiness study from AREVA, and other NGNP Project design studies and assessments.

The formatting of this report follows the general layout of the original lessons learned document. The subsections of this report (1) provide a cross reference to the lessons learned document[1], (2) describe the lesson learned, (3) provide the current NGNP Project implementation status associated with the lesson learned, including the identification of associated Design Data Needs (DDN), if any, (4) identify the R&D being performed related to the lesson learned and (5) provides a summary with a status of how the lesson learned has been addressed by the NGNP project. Final conclusions and references are provided at the end of the report.

2. NUCLEAR HEAT SUPPLY SYSTEM

2.1 Reactor Pressure Vessel

2.1.1 Insulation and Moisture Issues (Fort St. Vrain)

Cross reference to Section 2.1.1 of HTGR Lessons Learned NGNP[2].

2.1.1.1 Lesson Learned

Water was inadvertently injected into the primary system of the Fort St. Vrain (FSV) reactor through the circulator bearings. This water in the reactor pressure vessel (RPV) became one of the main issues because it was difficult to remove, which delayed the resumption of operations. The FSV reactor design did not include a drain at the base of the RPV to assist with the removal of water.

2.1.1.2 NGNP Project Implementation

Implementation Description

Current NGNP Project design concepts for circulators do not use water-cooled bearings. Any future NGNP Project designs of interfacing energy conversion systems (Heat Transfer System) and support systems containing water will need to be designed to minimize the potential for water intrusion into the core (e.g., water containing heat exchangers and steam generators (SGs) will be designed to meet the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code and positioned below the core). The NGNP design concept (GA Vessel System SDD[3]) has a drain installed at the base of the RPV to facilitate water removal, if necessary.

No DDNs related to the lessons learned.

Research and Development

No R&D activities are currently identified specifically related to circulator bearing integrity. A panel of experts in areas related to the U.S. NGNP design assessed modular moisture ingress events for an HTGR using a Phenomena Identification and Ranking Table (PIRT) process. This assessment is documented in the report, Assessment of NGNP Moisture Ingress Events[4]. One of the report's conclusions was: "Considering resource limits and the lack of more detailed NGNP design information available for this assessment, many of the possible sequence options and design variations were not covered. As the design progresses, the assumptions should be revisited in any subsequent PIRT-like activities."

Summary

Lessons Leaned	Status
Water cooled circulator bearings leaking.	The design concepts and assessments do not have water cooled bearings.
No drain at base of RPV.	The NGNP design concepts have a drain at the base of the RPV.

2.1.2 The Sealing and Flanges of the RPV (HTR-10)

Cross reference to Section 2.1.2 of HTGR Lessons Learned NGNP[5].

2.1.2.1 Lessons Learned

The designers of the High Temperature Reactor (HTR-10) in the People's Republic of China had identified the mechanical O-ring seal on the RPV closure flanges as a potential source of helium leakage, and included a provision to utilize a welded Ω-ring to seal the pressure vessel as well as the metallic O-ring. If NGNP reactor designs are to require removing reactor pressure vessel heads for routine refueling or other evolutions, this type of seal welding should be considered.

2.1.2.2 NGNP Implementation

Implementation Description

With regard to maintaining a helium pressure boundary the vessel head seals are an area of interest for the NGNP designs under consideration.

GA identifies DDN D.12.01.05, "Helium Seal Data for Bolted Closures," and Test 12.05.05-1, "Demonstrate the He leakage rate requirement for Vessel System flanged joints can be achieved for Normal Operating Condition," [6] as a need to address the RPV sealing and flange issue.

Research and Development

No R&D activities are currently identified related to RPV sealing and flanges.

Summary

Lesson Learned	Status
Helium leakage from RPV via O-rings.	Reactor designers will design the RPV to be 'helium tight' following design rules in accordance with the ASME BPV Code. Consideration of in-service refueling and other evolutions will be addressed during the design process.

2.1.3 RPV Cooling Design (Dragon)

Cross reference to Section 2.1.3 of HTGR Lessons Learned NGNP[7] .

2.1.3.1 Lessons Learned

The Dragon reactor was designed such that the pressure-bearing walls of the RPV were not to come into contact with the higher temperature outlet helium. Rather, the lower temperature inlet helium from the heat exchangers was used to keep the RPV within allowable temperature limits. Dragon was then designed so that the cool helium from the heat exchangers is used to keep the RPV from overheating. In the case of a complete power failure the reactor would trip and only the latent heat and decay heat would have to be removed.

2.1.3.2 NGNP Implementation

Implementation Description

This strategy has already been incorporated into the HTGR design concepts under consideration for the NGNP. For the NGNP design concepts under consideration, the RPV is generally exposed to primary helium at the reactor core inlet temperature (250 to 288°C). The inlet helium is directed between the RPV walls and the core barrel to maintain allowable RPV temperatures. Several alternatives (active vessel cooling and alternative materials) have been proposed to ensure that the RPV temperature does not exceed long-term operating temperature limits for the RPV material during normal operation and accident scenarios, primarily for HTGRs with the higher reactor outlet temperatures.

No DDNs are currently identified for the RPV cooling issue.

Research and Development

No R&D activities are currently identified related to RPV cooling design.

Summary

Lessons Learned	Status
Reactor should be designed so that the lower temperature helium from the heat exchangers is used to keep the RPV from overheating.	The lesson learned is understood by reactor supplierss and will be further addressed in the design phase.

2.2 Reactor Vessel Internals

2.2.1 Temperature Rise in Core Support Plate (High Temperature Engineering Test Reactor)

Cross reference to Section 2.2.1 of HTGR Lessons Learned NGNP[8].

2.2.1.1 Lessons Learned

During a power rise test, the core support plate of the Japanese High Temperature Engineering Test Reactor (HTTR) showed an unexpected temperature rise. Temperature analyses were carried out considering the anticipated bypass flow in the core support structures. The rise in temperature of the core support plate was attributed to bypass flow. To reduce the helium leakage flow in the spaces between blocks (called bypass flow), seal elements are placed in the spaces.

Implementation Description

In general, bypass flow phenomenon is understood by HTGR vendors, but bypass flow needs to be modeled and analyzed based on the reactor core design.

AREVA identified a DDN 3.4.1.0, “Reactor Core, Test Block Interface Flow Characteristics.” The helium bypass flow is a key parameter and a complex value to determine. Bypass at the interfaces between blocks (tiny gaps and misalignments) and the effects of potential cross flows and axial flows must be characterized under various conditions. Tests should be performed in a helium facility to determine bypass flow at the interfaces between blocks and assess the effects of potential cross flows and axial flows.

Research and Development

A series of bypass flow experiments within the Methods technical program (Next Generation Nuclear Plant Methods Technical Program Plan[9]) will test theories regarding factors that affect the quantity of bypass flow for gas-cooled reactors. Some influencing factors would be manufacturing tolerances and core configuration changes from irradiation or thermal expansion.

Summary

Lessons Learned	Status
Bypass flow for HTGRs should be studied.	The Methods program will test theories regarding factors that affect bypass flow. In addition, bypass flow experiments at the matched-index-of-refraction flow loop at INL will be collecting bypass flow data by the end of 2011.

2.2.2 Flow-Induced Vibrations (AGR)

Cross reference to Section 2.2.2 of HTGR Lessons Learned NGNP[10].

2.2.2.1 Lessons Learned

The CO_2 coolant in the Advanced Gas Reactor (AGR) created severe problems that are difficult to detect with out-of-pile loops. The problems are from flow induced vibrations driven by highly energized gas flow that contacts a relatively flexible structure, such as reactor internals or heat exchanger tubes. These problems may also occur in cross flows of closely spaced arrays of tubes, leaving them damaged. If not designed correctly, flow induced vibrations can cause the structures to become more flexible and possibly fail.

2.2.2.2 NGNP Implementation

Implementation Description

Flow-induced vibration is a fundamental area of investigation (Methods and Heat Transport R&D programs) in the design of NGNP structures exposed to fluid flows, as when primary helium flows across or along core internals and heat exchanger surfaces.

Westinghouse identified DDN PCS-01-12, “Flow Induced Vibration Testing of Helical Bundle.” Data are needed in order to accurately determine the flow-induced vibration characteristics of the NGNP SG-specific helical tube bundle, the lead-out tubes, and the transition tubes. Flow induced vibration is a widely recognized concern in the design of modern tube-and-shell type heat exchangers. Fluid flowing across a tube array can cause dynamic instability.

Research and Development

The NGNP Heat Transport System Components Engineering Test Plan[11] identifies that methods should be developed to detect and evaluate vibration that could damage NGNP components. Components of concern in the heat transport portion of the reactor coolant system include the insulation within the hot duct, SG tubes, internals of the heat exchangers, temperature sensor thermal wells, and the circulator(s). Current methods involve accelerometers to listen for vibration and ultrasonic or other devices to transmit a signal and assess the returning and/or transmitted sonic wave.

The Heat Transport Engineering Test Plan identified above is not part of any R&D programs currently progressing NGNP technology. Unless the test plans and activities identified in this document are adopted by R&D, there will be a gap in the technological advancement of several key components.

Summary

Lessons Learned	Status
Problems from flow induced vibrations driven by highly energized gas flow.	This lesson learned will be captured by future R&D and DDNs when the design of the components is more mature. This will also be addressed in design.

2.3 Reactor Core and Core Structures

2.3.1 High Moisture Content Issues (FSV)

Cross reference to Section 2.3.1 of HTGR Lessons Learned NGNP[12].

2.3.1.1 Lessons Learned

Moisture can cause hydrolysis of the fuel particle with exposed kernels resulting in a fuel failure and subsequent release of fission products. One source of moisture especially prevalent at FSV was the moisture out-gassing that occurs when the graphite is heated up and a so-called "drying out" of the graphite takes place.

2.3.1.2 NGNP Implementation

Implementation Description

Startup procedures for the NGNP plant may assess the state of moisture in the graphite as part of initiating power operation and ascent to full power. Moisture meters may also be positioned to measure moisture in the primary coolant and to detect potential moisture conditions during operation. These detectors can be used to ensure the moisture levels are low enough to start the plant and ascend in power.

No DDNs are currently identified that relate to core structure graphite moisture out-gassing as the graphite is heated up.

Research and Development

No R&D activities are currently identified related to core structure graphite moisture out-gassing as the graphite is heated up.

Summary

Lessons Learned	Status
High moisture content prevalent at FSV is the moisture out-gassing that occurs when the graphite is heated up.	The problem of moisture content in graphite can be alleviated by reactor startup procedures and controlling the amount of moisture absorbed by graphite through shipping and storage conditions.

2.3.2 Helium Pressurization Line (FSV)

Cross reference to Section 2.3.2 of HTGR Lessons Learned NGNP [13].

2.3.2.1 Lessons Learned

Several helium pressurizing lines at FSV became plugged because of corrosion. The helium pressurizing lines were connected to the refueling interspace and to the control rod drive mechanisms located in refueling penetrations. Corrosion was caused by moisture in contact with the carbon steel piping and collected at the interface between the three-quarter-inch supply line and one-eighth-inch inlet line.

2.3.2.2 NGNP Implementation

Implementation Description

NGNP will evaluate all components susceptible to moisture, and design with materials that have corrosion resistance in a relevant environment. Also, compliance with ASME BPV Code will require use of appropriate materials for the NGNP design. The Heat Transport Test Plan[14] identifies a facility called the high temperature helium loop, which is intended for corrosion and irradiation tests of reactor component materials as well as RPV internals at high temperatures and pressures in a helium environment. Currently, the Heat Transport Test Plan has not been adopted by the INL R&D program.

No DDNs are currently identified related to helium pressurizing lines plugged because of corrosion.

Research and Development

The INL materials program test plans for RPV[15] and intermediate heat exchanger (IHX)/SG[16] identify several tests to investigate the corrosion properties of candidate reactor materials. A panel of experts in areas related to the U.S. NGNP design assessed modular moisture ingress events for an HTGR using a Phenomena Identification and Ranking Table (PIRT) process. This assessment is documented in the report, Assessment of NGNP Moisture Ingress Events[17]. One of the report's conclusions was: "Considering resource limits and the lack of more detailed NGNP design information available for this assessment, many of the possible sequence options and design variations were not covered. As the design progresses, the assumptions should be revisited in any subsequent PIRT-like activities."

Summary

Lessons Learned	Status
Helium pressurizing lines became plugged because of corrosion.	The design concepts for reactors do not specifically address plugging of helium pressurizing lines because of corrosion. However, under the INL materials program, there are test plans to investigate the corrosion properties of candidate reactor metals. This lesson learned will be addressed in the design phase.

2.3.3 Core Temperature Fluctuations (FSV)

Cross reference to Section 2.3.3 of HTGR Lessons Learned NGNP [18].

2.3.3.1 Lessons Learned

In November 1977, FSV experienced some dynamics to the power levels and small fluctuations of temperature. The most probable explanation for the temperature fluctuations was small movements of reactor components, such as fuel elements and reflector columns. The motion of the reactor components was induced by pressure differences between gaps and thermal gradients in the core components, which caused component deformations and bowing. The pressure differences between gaps can result in bypass flow, which varies coolant flow between gaps within regions and/or blocks. The fluctuations were sustained by the interplay of these two phenomena (pressure differences and thermal gradients). Region constraint devices were designed and installed to stabilize the gap size between refueling regions at the top of the core, thus preventing, or at least reducing, the extent of core component motion and eliminating the core fluctuations.

2.3.3.2 NGNP Implementation

Implementation Description

The proposed NGNP design concepts have reflectors that include features such as keys and consolidation bands to minimize movement of the core reflector blocks, thus stabilizing the bypass flow size and minimizing the potential for temperature fluctuations related specifically to this phenomenon.

AREVA identified a prismatic reactor DDN 3.4.1.0, “Reactor Core—Test Block Interface Flow Characteristics.” The helium bypass flow is a key parameter and a complex value to determine. Bypass at the interfaces between blocks (tiny gaps and misalignments) and the effects of potential cross flows and axial flows must be characterized under various conditions. Tests should be performed in a helium facility to determine bypass flow at the interfaces between blocks and assess the effects of potential cross flows and axial flows.

Research and Development

The NGNP Methods Technical Program Plan[19] identifies bypass experiments with associated computational fluid dynamics model validation as one of its highest priority methods activities. In addition, bypass flow experiments at the matched-index-of-refraction flow loop at INL will be collecting bypass flow data by the end of 2011.

Summary

Lessons Learned	Status
Core temperature fluctuations because of pressure differences between gaps and thermal gradients in the core components.	The general mechanisms that cause core temperature fluctuations are understood by reactor designers. Further modeling and bypass flow experiments specific to an NGNP design concepts are being performed at INL. Bypass flow modeling and experiments will be adjusted during the design phase.

2.3.4 Inner Reflector (AVR)

Cross reference to Section 2.3.4 of HTGR Lessons Learned NGNP[20].

2.3.4.1 Lessons Learned

Inner reflector inspection was difficult due to unusual design. At the Arbeitsgemeinschaft Versuchsreaktor (AVR), the cylindrical core structure that held the spherical fuel elements was made of graphite bricks, and the graphite brick structure was enclosed in an envelope of carbon brick, which provided shielding and isolation. The whole composition was surrounded by a dual-walled steel liner. Because of AVRs unique core design and accessibility, special lighting and camera methods were developed to visually inspect the inner reflectors.

2.3.4.2 NGNP Implementation

Implementation Description

The NGNP will follow ASME BPV Code, Section XI, Division 2 for in-service inspections (still being developed and not yet issued).

No DDNs were identified for inner reflector inspections.

Research and Development

No R&D activities are currently identified for the inner reflector with regard to inspection capabilities.

Summary

Lessons Learned	Status
Consider reactor core design for Inner Reflector inspection.	There are no DDNs. R&D efforts or any design references for inner reflector inspection capabilities in the design concepts. This issue will be addressed in the design phase.

2.3.5 Graphite and Graphite Dust (HTR-10)

Cross reference to Section 2.3.5 of HTGR Lessons Learned NGNP[21].

2.3.5.1 Lessons Learned

At HTR-10 the reflectors are made from IG-11. The fuel and moderator pebbles comprising the core are made from A3-3 matrix material, a partially graphitized carbon that has different thermo-mechanical properties than IG-11.As the fuel elements move through the pebble bed of HTR-10, they come into contact with the side reflector, steel loading pipes, and other fuel elements. Also, because of the non-uniform temperature distribution and stress and deformation from irradiation, there is movement between the graphite reflector blocks. This movement and contact can wear the graphite, creating graphite dust and small particles. These particles can collect at the bottom of the core or be carried off and collect onto surfaces in the primary circuit, including the heat exchanger, thus decreasing its efficiency.

2.3.5.2 NGNP Implementation

Implementation Description

The generation, distribution, and cleanup of graphite dust in the primary helium circuit have been identified as necessary considerations in the design concepts of the primary helium circuit, helium purification system, and thermal and hydraulic analyses of the core under normal, abnormal, and accident conditions. A cleanup of graphite dust can be performed during a planned reactor core maintenance outage (approximately every 5 years).

No DDNs are identified related to the generation of graphite dust.

Research and Development

The INL Advanced Graphite Creep (AGC) program[22] plans to examine graphite wear using standard pin-on-wheel wear testing procedures that will be used to determine wear, friction, and dust generation values for selected grades of graphite. Previously irradiated and oxidized graphite will be subjected to similar tests to determine any changes. These will be limited studies focused on those graphite types of interest to pebble-bed and prismatic designs. The AGC program is focusing on all the characteristics necessary to qualify the graphite for use in the NGNP per the Nuclear Regulatory Commission (NRC) requirements, including the ASTM International standards and ASME BPV Code.

Summary

Lessons Learned	Status
Graphite dust generation in a reactor core.	Graphite dust will be addressed by the reactor suppliers and R&D program tests. A comprehensive report on graphite dust written by INL (HTGR Dust Safety Issues and Needs for Research and Development[23]) evaluates dust generation, distribution, characterization, explosion and many other dust properties.

2.3.6 Outer Reflectors Bowing (Peach Bottom Unit 1)

Cross reference to Section 2.3.6 of HTGR Lessons Learned NGNP [24].

2.3.6.1 Lessons Learned

The Peach Bottom reflector element B16-01 was made by the National Carbon Company's (Union Carbide Corporation, UCC) AGOT grade graphite. These outer reflectors were shown to work well within the reactor. Changes in length and bowing were measured by holes drilled into the graphite reflectors. These changes were within established limits and within predictions. The length changes and bowing are caused by neutron fluence at high temperatures.

2.3.6.2 NGNP Implementation

Implementation Description

The outer reflectors are subjected to large flux, and possibly large temperature gradients, across their width and length. It is therefore possible that they will become significantly distorted and bowed after several years of operation. In the GA design concepts, the outer reflectors are replaceable and have estimated lives of 3 to 10 years. Approximately one-third of these replaceable reflectors are exchanged for a new reflector block during each refueling.

No DDNs are identified related to outer reflector bowing.

Research and Development

The NGNP Graphite Technology Development Plan[25] identifies an AGC experiment that is designed to provide irradiation creep rates for moderate doses and higher temperatures of leading graphite types that will be used in the NGNP reactor design. The AGC program consists of six experiments, AGC-1 through AGC-6 that will identify irradiation, thermal, mechanical, and physical properties for inclusion in the graphite material property database.

The R&D program is developing and will validate (through irradiation and post-irradiation examination [PIE]), analytical tools to predict graphite block distortion, including bowing, as a function of temperature, stress, and neutron exposure. Extensive research is being done on numerous block reflector qualification parameters. The requirements and tests being performed are outlined in the above mentioned Graphite Technology Development Plan.

Summary

Lessons Learned	Status
Outer reflectors worked well. Selected material showed little bowing.	The R&D program addresses this lesson learned. The AGC program addresses the issue of graphite bowing. Through the various AGC experiments, researchers are investigating this behavior to characterize graphite in the high temperature and helium environments of the NGNP. The design concepts also identifies the need to be able to change out the reflectors as required.

2.4 Fuel Elements

2.4.1 Fuel Cracks Caused by Tensile Stress (FSV)

Cross reference to Section 2.4.1 of HTGR Lessons Learned NGNP [26].

2.4.1.1 Lessons Learned

Sometime before October 1981, the FSV licensee discovered that a crack had propagated through two stacked fuel elements. Based on calculation models, the licensee and the reactor suppliers concluded that the cracks were caused by induced high tensile and irradiation stresses resulted from incompatible peak factors in high stresses on the inter-regional faces of the two cracked fuel elements.

Implementation Description

Irradiation can cause high tensile stress in the fuel elements. The NGNP Advanced Gas Reactor (AGR) Graphite Development Program[27] is performing R&D to better understand the behavior of graphite fuel elements.

AREVA identified a prismatic reactor DDN 2.4.1.0a, “Graphite Thermal-Physical Properties.” This DDN was established to determine the thermal-physical properties of nuclear grade graphite anticipated to be used for NGNP fuel elements and core structures. The properties data will be used to qualify such graphite grades for use under NGNP operation.

Research and Development

The NGNP graphite program (of which the fuel element is made) identifies irradiation experiments[28] on graphite to see the effects resulting from tensile stress. Tensile stress either promotes or, at the very least, allows unhindered strain relief during irradiation, providing a worst-case creep rate for the graphite types exposed to higher doses. Compressive loads, after turnaround, will tend to retard the creep rate and effectively delay the tertiary creep regime. Therefore, once turnaround has been achieved, graphite samples should be in a tensile stress state to determine the fastest rate of irradiation creep possible within the graphite.

The INL graphite program[29] includes whole graphite core and component behavior models. Core and component-scale models will allow designers to predict core and core block (e.g., reflector or fuel element) dimensional distortion, component stresses, residual strength, and probability of failure during normal or off-normal conditions.

Summary

Lessons Learned	Status
Fuel element cracks because of tensile stress.	The R&D program will address this lesson learned. The AGC program addresses the issue of tensile stress on graphite.

2.4.2 Prismatic Fuel Performance (FSV)

Cross reference to Section 2.4.2 of HTGR Lessons Learned NGNP [30].

2.4.2.1 Lessons Learned

Experience with fuel design, development, and manufacture for FSV provided the basis for the fuel technology used for subsequent fuel quality and performance improvements. FSV provided invaluable fuel performance, fission product release, and plateout data that have been used for validation of GA design methods.

2.4.2.2 NGNP Implementation

Implementation Description

Fission product release from the fuel and plateout on the primary circuit were experienced at FSV. Data obtained from FSV can be used to validate HTFR fuel performance codes. Both GA and NGNP R&D are investigating these concerns to better understand and provide a path forward to limit fission product release and plateout. GA identified several DDNs related to fuel performance:

DDN 7.02.01, Coating Material Property Data

DDN 7.02.02, Defective Particle Data

DDN 7.02.03, Thermochemical Performance Data for Fuel

DDN 7.02.04, Fuel Compact Thermophysical Properties

DDN 7.02.05, Normal Operation Fuel Performance Validation Data

DDN 7.02.06, Accident Fuel Performance Validation Data

DDN 7.02.07, Fuel Proof Test Data.

Research and Development

The INL AGR fuels R&D program is based upon demonstration of fuel performance and qualification for service conditions enveloping normal operations and accidents.

Summary

Lessons Learned	Status
Prismatic fuel performance.	The fuel R&D program activities will address this lesson learned (and subsequently identified DDNs).

2.4.3 Pebble Bed Fuel Performance (AVR)

Cross reference to Section 2.4.3 of HTGR Lessons Learned NGNP[31] .

2.4.3.1 Fission Products at Elevated Temperatures

Cross reference to Section 2.4.3.1 of HTGR Lessons Learned NGNP[32].

2.4.3.1.1 Lessons Learned

In September 1988, the Jülich Institute für Reaktorwerkstoffe GmbH published a report[33] documenting experiments on fission products released for pebble-bed fuel elements containing tristructural isotropic (TRISO) coated particles. The experiments confirmed that TRISO fuel would perform correctly at accidental temperature scenarios without degradation and with minimal fission product release.

2.4.3.1.2 NGNP Implementation

Implementation Description

The NGNP Fuel Development and Qualification Program is focused on the prismatic reactor fuel design. The fuel elements are in the form of compacts and contain fuel particles with the prismatic UCO fuel kernels (with the exception of AGR-2, which has both UCO and UO_2 particle kernels).

No DDNs related to pebble bed fuel performance were identified.

Research and Development

The technical program for the NGNP/AGR Fuel Development and Qualification Program[34] utilizes data from the AVR program as well as other reactors from around the world such as Peach Bottom, Dragon, and HTTR.

Summary

Lessons Learned	Status
Pebble bed fuel performance.	The Fuel R&D program activities will address this lesson learned. The AGR fuel development program is focused around the prismatic particles with a UCO fuel kernel. If a pebble bed design is selected for the NGNP then the program may need to switch focus to the pebble particle with a UO_2 kernel. The prismatic particle can be used in a pebble design, leaving the option to continue with UCO R&D.

2.4.3.2 Immediate Post-AVR Perspective

Cross reference to Section 2.4.3.2 of HTGR Lessons Learned NGNP[35].

2.4.3.2.1 Lessons Learned

A report was generated immediately following shutdown of the AVR[36]. One point noted in the report was that the deposition of solid fission products in the primary loop is determined to a significant extent by dust; a high fine dust fraction with grain sizes in the range of 1 μm has been ascertained; this is the principal carrier of mobilized activity.

2.4.3.2.2 NGNP Implementation

Implementation Description

AVR experience points out the importance of accounting for dust in proposed NGNP operations and process heat equipment design. The generation, distribution, and cleanup of graphite dust in the primary helium circuit have been identified as necessary considerations in the design concepts of the primary helium circuit and helium purification system.

No DDNs are identified related to the generation of graphite dust.

Research and Development

The NGNP/AGR Fuel Development and Qualification Program[37] identifies that the available data on the effects of dust on radionuclide transport in the primary coolant circuit are largely from reactor surveillance measurements from Peach Bottom, Dragon, AVR, and HTTR. Samples of deposited particulate matter were obtained from an FSV circulator and have been partially characterized at Oak Ridge National Laboratory.

NGNP AGR R&D activities are identified to address dust and plateout of radionuclides, such as that experienced at AVR. The NGNP Graphite Technology Develop Program also addresses graphite dust and the impacts it may have on the reactor.

Summary

Lessons Learned	Status
Graphite dust generation in a reactor core.	Graphite dust will be addressed by the reactor suppliers and R&D program tests. A comprehensive report on graphite dust written by INL (HTGR Dust Safety Issues and Needs for Research and Development[38]) evaluates dust generation, distribution, characterization, explosion and many other dust properties.

2.4.3.3 Direct Operational AVR Experience

Cross reference to Section 2.4.3.3 of HTGR Lessons Learned NGNP[39].

2.4.3.3.1 Lessons Learned

When the AVR first started up, it experienced a higher than expected damage rate of the fuel elements. This was determined to be the result of overly dense packing on the initial fuel load. Through continuous cycling of the initial fuel, the damaged fuel was removed, loosening the fuel bed to a lower density, therefore decreasing the damage rate to expected levels.

2.4.3.3.2 NGNP Implementation

Implementation Description

NGNP would benefit by considering AVR issues such as the pebble elements becoming too tightly packed for initial operations. The AVR program provided a large amount of fuel performance data that the NGNP Project can use in the future design phase for fuel.

One key difference in post-AVR reactors is placement of the control rods in the graphite reflectors (which resulted in no contact between the control rods and fuel) compared to direct insertion into the pebble beds (as the AVR did), giving rise to the potential to damage the fuel.

No DDNs are identified related to the packing density of pebble fuel.

Research and Development

No R&D activities are currently identified related to the packing density of pebble fuel.

Summary

Lessons Learned	Status
Packing density of pebble fuel.	AREVA has addressed initial first core loading in their PBR assessment. No R&D activities or DDNs identified with this lessons learned. This will be addressed during design and initial operation.

2.4.4 Fuel Summary (HTR-10)

Cross reference to Section 2.4.4 of HTGR Lessons Learned NGNP[40] .

2.4.4.1 Lessons Learned

Testing of HTR-10 fuel provides useful information for the NGNP fuel design program. From this testing of HTR-10 fuel, fuel failures were experienced above 1600°C. Fuel failure was also experienced as a result of impurities and manufacturing defects. The HTR-10 experience provides examples of fuel failure mechanisms and provides additional justification for NGNP using quality control of the fuel manufacturing process.

2.4.4.2 NGNP Implementation

Implementation Description

Fuel failure is a concern across all stakeholders of the NGNP. The INL fuels program is to provide a fuel qualification data set in support of the licensing and operation of the NGNP. The qualification data set includes accident and fuel failure conditions.

Two DDNs are related to fuel failure experiments and modeling:

- AREVA DDN 4.1.3.1d, “Fuel—Heat-Up Experiment Modeling of Irradiated Fuel Particles in ATLAS.” ATLAS can model the behavior of HTGR fuel in heat-up accident conditions. It calculates the fission gas release and the failure rate in such conditions as well as in normal operating conditions.

- GA DDN C.11.03.04, “Core Element Failure Mode Data.”

Research and Development

The INL AGR fuel qualification program has a number of irradiation experiments (AGR-1 to AGR-8)[41] identified in its program. One of the irradiation experiments is AGR-3/4 Fission Product Transport Data, which has design-to-fail fuel particles (DTF) which will fail early in the irradiation and provide a known source of fission products. This will allow an assessment of the effect of impurities on intact and design-to-fail fuel performance and subsequent fission product transport.

Summary

Lessons Learned	Status
Experience learned from fuel failures.	The INL AGR fuel qualification program will address fuel particle failure testing and PIE. Two DDNs are related to fuel failure.

2.4.5 Pebble Bed Graphite Dust (AVR)

Cross reference to Section 2.4.5 of HTGR Lessons Learned NGNP[42].

2.4.5.1 Lessons Learned

Several sources of graphite dust have been identified at AVR: notching, spalling, pitting, fuel-sphere fracture, and peeling of the fuel spheres. The dust from these sources has contributed to activity concentration in the primary coolant.

2.4.5.2 NGNP Implementation

Implementation Description

Dust generation in pebble bed reactors is presumed to result primarily from friction between pebbles within the core and between pebbles and the fuel handling system. The latter may contribute a significant portion of the total dust to be found in such a reactor.[43]

Since pebble bed fuel pebbles are susceptible to graphite deterioration, much attention is given to identifying and preventing pebble bed graphite dust generation. A comprehensive report on graphite dust written by INL (HTGR Dust Safety Issues and Needs for Research and Development[44]) evaluates dust generation, distribution, characterization, explosion and many other dust properties.

No DDNs are identified related to the generation of graphite dust from fuel pebbles.

Research and Development

The NGNP Graphite Technology Development Plan[45] includes examining graphite wear using standard pin-on-wheel wear testing procedures that will be used to determine wear, friction, and dust generation values for selected grades of graphite. Previously irradiated and oxidized graphite will be subjected to similar tests to determine any changes. These will be limited studies focused on those graphite types of interest to pebble-bed designs.

Summary

Lessons Learned	Status
Dust generation because of handling fuel pebbles.	The INL R&D graphite program will examine graphite wear and dust generation for selected grades of graphite.

2.4.6 Prismatic Graphite Dust (HTTR)

Cross reference to Section 2.4.6 of HTGR Lessons Learned NGNP[46].

2.4.6.1 Lessons Learned

In November 2009, the Paul Scherer Institute in Villingen, Switzerland sponsored a meeting of researchers and subject matter experts on high temperature reactor (HTR) graphite dust. A report[47] spawned from this meeting discusses the key issues with graphite dust. Two phenomena are identified in the report: tribology of graphite in impure helium environments, and graphite dust generation. HTTR has experienced carbonaceous dust deposits (as discussed in the report) in the mesh filter of the primary helium circulator.

2.4.6.2 NGNP Implementation

Implementation Description

Significantly less dust is expected to be present in a prismatic reactor [48] than a pebble-bed type reactor, since the primary source of dust generation is from pebble friction. Nevertheless, small quantities of dust may be present in a prismatic reactor for a variety of reasons: (1) foreign material introduced during construction or refueling, (2) friction or erosion of prismatic fuel and reflector surfaces exposed to helium, (3) foreign material from interfacing systems, (4) corrosion or erosion of metallic surfaces in the primary coolant system, and (5) CO decomposition.

A comprehensive report on graphite dust written by INL (HTGR Dust Safety Issues and Needs for Research and Development[49]) evaluates dust generation, distribution, characterization, explosion and many other dust properties.

No DDNs are identified related to prismatic dust generation.

Research and Development

Tribological properties for graphite materials being considered for the NGNP are being evaluated by the AGC program[50]. Research is being conducted with a focus on any dust-oriented PIRTs defined by the NRC.

The basic feasibility of graphite planned for the NGNP has previously been demonstrated in former HTGR plants (e.g., Dragon, Peach Bottom, AVR, and FSV reactor). The Japanese HTTR is demonstrating the feasibility of the reactor components and materials needed for NGNP. This experience has, in large part, formed the current understanding of graphite response within an HTGR nuclear environment.[51]

Summary

Lessons Learned	Status
Dust generation from prismatic fuel.	The INL R&D graphite program will examine graphite wear and dust generation for selected grades of graphite. Past experience will serve as a basis for dust generation and particle size information for a future prismatic reactor.

2.4.7 Fission Product Release (HTTR)

Cross reference to Section 2.4.7 of HTGR Lessons Learned NGNP[52].

2.4.7.1 Lessons Learned

HTTR experience shows a positive example that an HTGR can be run at full power (at 950°C) without releasing any detectable amounts of fission products. Over the course of numerous years, HTTR had a goal to achieve an operational state with a reactor outlet temperature of 950°C. From January 5 to March 21, 2010, JAEA operated the HTTR at full power and temperature, recording plant conditions and collecting tritium data to complete a mass balance of the entire plant. Tritium concentrations were measured in the primary helium cooling system, secondary helium cooling system, pressurized water cooling system, auxiliary water cooling system, and containment vessel.

2.4.7.2 NGNP Implementation

Implementation Description

Under cooperation and agreements with members of the Generation IV International Forum, such as the Japan Atomic Energy Agency (JAEA), NGNP will be able to use and benefit from the parallel research of other HTGRs currently operating as research reactors. Battelle Energy Alliance, LLC (as the INL contractor) licensed the data from JAEA, configured TPAC as the HTTR and duplicated the operating conditions in the code. TPAC predictions of tritium concentrations at multiple locations in the system were then compared to the actual data.

There are a number of DDNs related to fission product release:

- AREVA DDN 1.3.1.0b, “Fuel Compact—Fission Product Interactions. Determine the interactions between the fuel matrix material and key radionuclides that impact fission product transport through the matrix material.”
- AREVA DDN 1.3.1.0, “Fission Product Speciation During Mass Transfer. Determine chemical speciation of fission products within the primary system and the confinement for differing potential atmospheres, including those encountered during water or air ingress events.”
- GA DDN C.07.03.01, “Fission Gas Release from Core Materials.”
- GA DDN C.07.03.03, “Fission Product Effective Diffusivities in Particle Coating.”

Research and Development

One key area of the fuels qualification program is fission product transport and source term. The AGR-3/4 Fission Product Transport Data and AGR-7/8 Fuel Performance and Fission Product Transport Verification and Validation irradiation experiments will explore the fission product transport. There is also a Fuels Performance Modeling program (part of the AGR program), whose main purpose is to develop validated fuel performance models. Fuel performance modeling addresses the structural, thermal, and chemical processes that can lead to coated-particle failures. It considers the effect of fission product chemical interactions with the coatings, which can lead to degradation of the coated-particle properties. Fission product release from the particles and transport within the fuel compact matrix and fuel element graphite will also be modeled.

Summary

Lessons Learned	Status
HTTR experience shows a positive example that an HTGR can be run at full power (at 950°C) without releasing any detectable amounts of fission products.	The INL R&D Fuels program will address fission product release over a number of scenarios based around normal and accident conditions. There are several DDNs related to fission product release.

2.4.8 Fission Product Trapping (Peach Bottom Unit 1)

Cross reference to Section 2.4.8 of HTGR Lessons Learned NGNP[53].

2.4.8.1 Lessons Learned

A fission product trapping system was used at Peach Bottom to purify the primary coolant. Helium would enter the fission product trapping system after going through the fuel purge line in the reactor core. The fission product trapping system and the purge system worked efficiently. It was observed that the primary circuit at the end of life was remarkably clean and the activity was never greater than 1 Ci.

2.4.8.2 NGNP Implementation

Implementation Description

To remove fission products from the primary circuit/coolant, trapping mechanisms must be deployed to capture fission products. Trapping the fission products may include the use of a helium purification system. It has the capacity to remove both chemical and fission products.[54]. There is an NGNP Reactor Coolant Chemistry Control Study[55] written by the INL that reviews helium purification techniques.

No DDNs are identified related to fission product trapping.

Research and Development

No R&D activities are currently identified related to trapping fission products.

Summary

Lessons Learned	Status
The fission product trapping system on Peach Bottom was successful.	There are no INL R&D activities or DDNs associated with this lesson learned. The reactor suppliers will address this lessons learned in design.

2.4.9 Amoeba Effect (Peach Bottom Unit 1)

Cross reference to Section 2.4.9 of HTGR Lessons Learned NGNP[56].

2.4.9.1 Lessons Learned

Fuel experience from Peach Bottom was useful for the development of TRISO fuel, which was the next step and solution to many of the problems with the earlier fuels. However, the amoeba effect—the migration of the nuclear fuel kernel across the fuel particle driven by high temperature thermal gradients through the fuel element cross section—may still have some impact on TRISO fuel performance.

2.4.9.2 NGNP Implementation

Implementation Description

The AGR Fuel Development and Qualification Program is currently qualifying TRISO fuel for the NGNP. Failure of fuel particle coatings because of amoeba effect is under investigation by the Fuel Qualification R&D Program.

No DDNs are identified related to the Amoeba effect.

Research and Development

Kernel migration at high burnups, power densities, temperatures, and temperature gradients are under investigation within the NGNP AGR program.

NGNP has selected UCO fuel because the kernel composition is engineered to prevent carbon monoxide formation and kernel migration—the "…key threats to fuel integrity of higher burnups, temperatures, and temperature gradients."[57] The AGR-7 test train will be designed so that a measurable level of fuel failure and/or fission product release is expected to occur in support of fuel performance model validation and operating margins.[58]

Summary

Lessons Learned	Status
Experience gained at Peach Bottom on the amoeba effect is relevant to TRISO fuel development.	The INL R&D fuels program will address the kernel migration at high burnups. No DDNs are or reactor designer programs related to this lesson learned.

2.5 Reactivity Control Systems

2.5.1 Moisture Effects on Reactivity Control System (FSV)

Cross reference to Section 2.5.1 of HTGR Lessons Learned NGNP[59].

2.5.1.1 Lessons Learned

A reactor scram occurred at FSV on June 23, 1984. The operators first verified the reactor was subcritical; however, they also noted that six control rod pairs had failed to fully insert in response to the scram signal. After several attempts the control rods were finally inserted 20 minutes after the initial automatic scram signal. That event was ultimately attributed to moisture collecting on the control rods causing corrosion and preventing the control rods from being released into the core.

2.5.1.2 NGNP Implementation

Implementation Description

The proposed NGNP control rod design concepts are based around differing neutron absorber compacts, but they identify Alloy 800H as the preferred candidate material for the cladding. Alloy 800H has an excellent resistance to corrosion at high temperatures. The 800 series of alloys were developed for high temperature strength and resistance to oxidation, carburization, and other forms of high temperature corrosion.[60]

No DDNs are related to moisture affects on reactor control rods.

Research and Development

NGNP and the Materials R&D program are evaluating materials that are suitable for the control rod environment. A panel of experts in areas related to the U.S. NGNP design assessed modular moisture ingress events for an HTGR using a Phenomena Identification and Ranking Table (PIRT) process. This assessment is documented in the report, Assessment of NGNP Moisture Ingress Events[61]. One of the report's conclusions was: "Considering resource limits and the lack of more detailed NGNP design information available for this assessment, many of the possible sequence options and design variations were not covered. As the design progresses, the assumptions should be revisited in any subsequent PIRT-like activities."

Summary

Lessons Learned	Status
Moisture effects on reactor control rods from FSV.	The effects of moisture on reactor control rods are understood by reactor designers. The design concepts of control rod cladding materials are based around alloys with high corrosion resistance in an HTGR environment.

2.5.2 Control Rod Cable Failure (FSV)

Cross reference to Section 2.5.2 of HTGR Lessons Learned NGNP[62].

2.5.2.1 Lessons Learned

In August 1984 a cable for a control rod pair at FSV broke and jammed in its guide tube during a test of the control rod drive (CRD). This was attributed to moisture induced leaching of volatile chlorides (from various sources within the reactor). The stainless steel control rod was subsequently found to have chloride-induced stress corrosion cracking. The steel cables were replaced with corrosion resistant Inconel cable.

2.5.2.2 NGNP Implementation

Implementation Description

The selection of reactor control rod cable is a design issue. The reactor designers are aware that materials for the control rod cables have to withstand the environment of an HTGR. The INL R&D materials program are investigating candidate reactor component materials such as Alloy 800H. The extension of Alloy 800H material properties in ASME BPV Code, Section III, Division 1, Subsection NH is currently underway for a service life up to 500,000 hours and up to an operating temperature range between 850 - 900°C.

AREVA identifies DDN Comp-01-02, "RCS Materials Characterization," to address the evaluation of materials corrosion concerns.[63] This DDN states the need to evaluate Alloy 800H to address the need to extend the code for materials to handle the high temperatures over time currently allowed by ASME BPV Code Case N-201-5 and Section III.

Research and Development

No specific R&D activities are based around the effects of corrosion on reactor control cables. However, there are R&D activities in the materials program based on the candidate materials for use in reactor internals (such as Alloy 800H). A panel of experts in areas related to the U.S. NGNP design assessed modular moisture ingress events for an HTGR using a Phenomena Identification and Ranking Table (PIRT) process. This assessment is documented in the report, Assessment of NGNP Moisture Ingress Events[64]. One of the report's conclusions was: "Considering resource limits and the lack of more detailed NGNP design information available for this assessment, many of the possible sequence options and design variations were not covered. As the design progresses, the assumptions should be revisited in any subsequent PIRT-like activities."

Summary

Lessons Learned	Status
Corrosion effects on reactor control rods cable due to moisture induced leaching of volatile chlorides.	The effects of corrosion on reactor control rods cable are understood by reactor designers. Control rod cable selection will be based on approved materials according to ASME BPV Code specifications.

2.5.3 Control Rod Temperature Anomalies (HTTR)

Cross reference to Section 2.5.3 of HTGR Lessons Learned NGNP[65].

2.5.3.1 Lessons Learned

In 1997, nonnuclear heat-up tests were carried out at HTTR. When the primary coolant temperature reached 110°C by heat input from the gas circulators, the helium gas temperature around the control rod drive mechanisms inside the standpipes reached the alarm point of 60°C. At the same time, the temperature of the primary upper shielding reached about 75°C, which was higher than anticipated. The cause of the temperature rise of the primary upper shielding and the helium gas inside the stand- pipes was investigated and found to be unanticipated bypass flow.

2.5.3.2 NGNP Implementation

Implementation Description

Hot spots can occur in the reactor because of unanticipated bypass flow of the primary coolant. Future HTGR designs are anticipated to benefit by carefully analyzing reactor coolant flows and incorporating proper cooling for all operational scenarios.

There are DDNs specific to bypass flow around the control rod drive mechanisms and primary upper shielding.

Research and Development

Within the Methods technical program, a series of bypass flow experiments[66] will test theories regarding factors that affect the quantity of bypass flow for both prismatic and pebble bed reactors. The bypass flow experiments are focused around the flow throughout the reactor core. Another source of elevated control rod temperatures is an extended loss of forced cooling. Generation of decay heat without active removal may lead to very high temperatures in the reflector regions in which the control rod guide tubes are located. While the fuel is expected to remain intact under such conditions, metallic control rod guide tubes achieve failure temperatures. The development of the Evaluation Model under the NGNP Methods plan will provide the tools for estimating control rod region temperatures and the likelihood of failure.

Summary

Lessons Learned	Status
Bypass flow effects on the primary upper shielding at HTTR.	Bypass flow is an issue being studied by the INL R&D methods program for both prismatic and pebble bed reactors. This lesson learned will be addressed in the design phase.

2.5.4 Control Rods in Side Reflectors and Lubricants in a Helium Environment (HTR-10)

Cross reference to Section 2.5.4 of HTGR Lessons Learned NGNP[67].

2.5.4.1 Lessons Learned

Lubrication for the control rods at HTR-10 was a concern because of the requirement to maintain a helium purity in a high temperature and high radiation environment. Oils could not be used because they would evaporate and pollute the helium. Molybdenum disulfide was found to be a good solid lubricant and demonstrated to have an effective duty life during radiation tests and friction tests, within the operating environment of helium and at operating temperatures.

2.5.4.2 NGNP Implementation

Implementation Description

The selections of control rod lubricants that are effective in an HTGR environment is a reactor suppliers design issue.

No DDNs are identified related to control rod lubricants.

Research and Development

No R&D activities are currently identified related to control rod lubricants.

Summary

Lessons Learned	Status
HTR-10 Control rod lubricant successful selection.	No DDNs are nor INL R&D activities related to control rod lubricants. There is nothing in the design concepts specifically related to control rod lubrication. This lesson learned will be addressed in the design phase.

2.5.5 Control Rods Placements (Peach Bottom Unit 1)

Cross reference to Section 2.5.5 of HTGR Lessons Learned NGNP[68].

2.5.5.1 Lessons Learned

Peach Bottom control rods were placed and operated at the bottom of the reactor to avoid the severe operating conditions imposed by the high temperature and high radiation environments at the top of the reactor. The location of the control rod assembly caused some licensing issues in that the reactor would not have “an inherent fail-safe shutdown geometry that would insert control rods by gravity in case of loss of actuation power.” Demonstrated proof of reliability was required prior to licensing.

2.5.5.2 NGNP Implementation

Implementation Description

All current HTGR proposed designs have control rods located on top of the reactor in order to have an inherent fail-safe shutdown in case of power loss (control rods inserted by gravity).

No DDNs are identified related to positioning of the control rods at the bottom of the reactor.

Research and Development

No R&D activities are currently identified related to control rod placement.

Summary

Lessons Learned	Status
Peach bottom positioning of control rods at the bottom of the reactor caused licensing issues.	This is a lesson that has been well learned, there are no proposals by any reactor designers to position control rods at the bottom of the reactor.

2.5.6 Control Rod Lubrication (Peach Bottom Unit 1)

Cross reference to Section 2.5.6 of HTGR Lessons Learned NGNP[69].

2.5.6.1 Lessons Learned

At Peach Bottom Unit 1 conventional lubrication (such as oils) could not be used because of the high temperature, high radiation environment and because of the potential impurities introduced into the helium coolant. Testing of new lubricants indicated that material combinations possessing extreme surface hardness of nearly equal value allowed continued functioning, although with a noticeable increase in friction. Nitride surface hardened materials were the most outstanding in this respect and were universally chosen throughout the design where this requirement was a factor.

2.5.6.2 NGNP Implementation

Implementation Description

The selection of control rod lubricants effective in an HTGR environment is a reactor suppliers design issue.

No DDNs are identified related to control rod lubricants.

Research and Development

No R&D activities are currently identified related to control rod lubrication.

Summary

Lessons Learned	Status
Peach Bottom control rod lubricants selection success.	No DDNs are nor INL R&D activities related to control rod lubricants. There is nothing in the design concepts related to control rod lubrication. This lesson learned will be addressed in the design phase.

2.5.7 Oil Leaks (Peach Bottom Unit 1)

Cross reference to Section 2.5.7 of HTGR Lessons Learned NGNP[70].

2.5.7.1 Lessons Learned

At Peach Bottom Unit 1 several occurrences of oil leaks were found in the hydraulic components at static seal connections and piston seals associated with accumulators that provide the stored energy source for a reactor trip insertion. The piston seal leaks were determined to be caused by defects in the cylinder wall surface machining.

2.5.7.2 NGNP Implementation

Implementation Description

There is nothing in NGNP design concepts related to hydraulic components and seal integrity. This is a design issue that will be addressed by the reactor suppliers in the design phase.

No DDNs are identified related to oil leaks.

Research and Development

No R&D are activities related to hydraulic components and oil leaks concerns.

Summary

Lessons Learned	Status
Peach Bottom Unit 1s hydraulic component oil leaks.	No DDNs are nor INL R&D activities related to hydraulic component oil leaks. There is nothing in the design concepts related to hydraulic component oil leaks. This lesson learned will be addressed in the design phase.

2.5.8 Fatigue in Control Rods (Peach Bottom Unit 1)

Cross reference to Section 2.5.8 of HTGR Lessons Learned NGNP[71].

2.5.8.1 Lessons Learned

During low power testing at Peach Bottom Unit 1, two control rods began to show symptoms of erratic motion during the regulating mode of operation. After investigation it was found that several balls in the ball screw assemblies had broken. It was determined that the design was satisfactory but the balls had imperfections. All the balls were replaced with new ones of load carrying size and with a higher grade of precision. After this change, the ball screw components showed no problems.

2.5.8.2 NGNP Implementation

Implementation Description

This lesson learned is a manufacturing quality issue for reactor components. All reactor components will be designed, manufactured, and installed in accordance to ASME BPV Code, ASTM International standards.

No DDNs are identified related to ball screw component quality.

Research and Development

No R&D are activities associated with component imperfections.

Summary

Lessons Learned	Status
Imperfections in the balls of the ball screw assemblies at Peach Bottom Unit 1.	This is a quality issue that will be addressed by NQA-1 quality procedures.

2.6 Reactivity Cavity Cooling System

2.6.1 Reactivity Cavity Cooling System

Cross reference to Section 2.6 of HTGR Lessons Learned NGNP[72].

2.6.1.1 Lessons Learned

The HTTR's reactor cavity cooling system (RCCS) is cooled by forced circulation of water. A water-cooled RCCS is a three-dimensional structure with many parallel channels. There is considerable operating experience for a water-cooled RCCS, but the Japanese experience from HTTR was that it can be difficult to operate properly.

2.6.1.2 NGNP Implementation

Implementation Description

The RCCS is a standard component across all HTGR design concepts. Its primary function is to remove normal operating waste heat from the reactor cavity. The NGNP design concepts include both air and water cooled designs.

There are several DDNs identified related to the characteristics of an RCCS in operation:

- AREVA DDN 3.3.4.0, "RCCS—Characterization of the heat transfer characteristics of the anticipated or proposed surface treatments for the reactor vessel and the panel heat exchanger will need to be accomplished." A large scale (e.g., representative height) demonstration of the capability of an RCCS is required.
- AREVA DDN 3.3.4.0a RCCS Characterization of the Heat Transfer of the Surface Treatments for the RV & PHX. The primary function of the Reactor Cavity Cooling System (RCCS) is to protect the reactor cavity concrete from overheating during normal operation. It provides an alternate means of heat removal from the Reactor System to the environment.
- AREVA DDN 3.3.4.0b, "RCCS—Optional Large Scale Test—Data Range/Service Conditions: Under irradiation, not under irradiation, under normal operating conditions, under accident conditions."
- AREVA DDN 3.3.4.0c, "RCCS—Characteristic Effects of Particulate and Plateout on Radiation Heat Transfer." Effects of plateout and particulates on radiation heat transfer. More examination is required during conceptual design to determine which tests will be required.
- GA DDN C.16.00.02, "Wind Tunnel Test of RCCS I/O Structure."
- GA DDN C.16.00.03, "Integrated RCCS Performance."

Research and Development

The NGNP Methods technical program has identified a task for ex-core heat transfer. The objective of this task is to acquire model/code validation data for natural convection and radiation heat transfer in the RCCS by performing experiments in the Argonne National Laboratory Natural Convection Shutdown Heat Removal Test Facility.[73]

Summary

Lessons Learned	Status
From HTTR experience it can be difficult to operate a water cooled RCCS.	All the NGNP reactor designers have a water cooled reactivity control system as an option. There are several DDNs requesting operational and performance data on a water-cooled RCCS. Apart from INL R&D modeling by the Methods program there are no other R&D activities related to the RCCS. This lesson learned will be addressed in the design phase.

2.7 Reserve Shutdown System

2.7.1 Inadvertent Actuation of Reserve Shutdown System (FSV)

Cross reference to Section 2.7.1 of HTGR Lessons Learned NGNP[74].

2.7.1.1 Lessons Learned

At FSV the reserve shutdown system (RSS) was inadvertently actuated, which injected the Region 27 RSS boron balls (also denoted as boronated graphite balls) into the core. The accidental injection of the boronated graphite balls went undetected for almost a month, since there was no indication that the hopper door had failed or was open.

2.7.1.2 NGNP Implementation

Implementation Description

This lesson learned is specifically related to the RSS releasing boron balls that went undetected for a considerable period of time. There is nothing in the NGNP design concepts that addresses this issue.

No DDNs are identified related to the detection of boronated graphite balls being released into a core.

Research and Development

No R&D activities are currently identified related to detecting the release of boronated graphite ball.

Summary

Lessons Learned	Status
FSV failure to detect release of boronated graphite balls into the reactor core.	This is a design activity that will be addressed by the selected reactor supplier in the design phase.

2.7.2 RSS Failure to Deploy Boronated Graphite Balls (FSV)

Cross reference to Section of HTGR Lessons Learned NGNP[75].

2.7.2.1 Lessons Learned

One RSS hopper only discharged about half of the full amount of the poison material (boronated graphite balls) during a test. The highly boronated material was stuck together, apparently because boric acid crystals had formed. A source of water in the purified helium train was suspected of causing the leaching of the boronated material that formed the acid crystals. The reserve shutdown materials with high boric acid concentrations were located in 18 of the system's 37 hoppers.

2.7.2.2 NGNP Implementation

Implementation Description

This lesson learned is specifically related to the RSS not discharging all the boron balls from the RSS hopper. Nothing in the NGNP design concepts addresses this issue. This is a design issue related to the RSSs inability to detect full poison material deployment (boron balls) and the ability to prevent moisture ingress into the RSS hopper. A panel of experts in areas related to the U.S. NGNP design assessed modular moisture ingress events for an HTGR using a Phenomena Identification and Ranking Table (PIRT) process. This assessment is documented in the report, Assessment of NGNP Moisture Ingress Events[76]. One of the report's conclusions was: "Considering resource limits and the lack of more detailed NGNP design information available for this assessment, many of the possible sequence options and design variations were not covered. As the design progresses, the assumptions should be revisited in any subsequent PIRT-like activities."

No DDNs are identified related to the failure to discharge all of its boron balls from the RSS hopper.

Research and Development

No R&D activities are related to RSSs ability to deploy boronated graphite balls.

Summary

Lessons Learned	Status
FSV failure to deploy all of the boronated graphite balls from the systems hopper.	This is a design activity that will be addressed by the selected reactor supplier in the design phase.

3. HEAT TRANSPORT SYSTEM

3.1 Circulators

3.1.1 Bolt Shearing (FSV)

Cross reference to Section 3.1.1 of HTGR Lessons Learned *NGNP*[77].

3.1.1.1 Lessons Learned

A detailed investigation was conducted at FSV following replacement of the circulator in 1987, which revealed that the bolts holding the insulation shroud and steam seal in place failed because of stress corrosion cracking brought about by caustic embrittlement.

3.1.1.2 NGNP Implementation

Implementation Description

The lesson learned is specific to bolt failure on a helium circulator holding the insulation shroud and steam seal in place. There is nothing in the NGNP design concepts that addresses bolt failure because of caustic embrittlement.

This is a design issue that will be addressed by the selected reactor supplier. All reactor component materials must adhere to the ASME BPV Code requirements.

No DDNs are identified related to bolt failure because of caustic embrittlement (or any other type of failure).

Research and Development

No R&D are identified for activities addressing bolt failure.

Summary

Lessons Learned	Status
Circulator bolt failure at FSV because of caustic embrittlement.	This is a design activity that will be addressed by the selected reactor supplier in the design phase.

3.1.2 Circulator Seals and Stress Corrosion Cracking (FSV)

Cross reference to Section 3.1.2 of HTGR Lessons Learned NGNP[78].

3.1.2.1 Lessons Learned

At FSV it was discovered that the D helium circulator was leaking helium through its seal above the technical specification limit. The D circulator had sustained damage and had to be replaced. Metallurgical observations confirmed preexisting cracks in the labyrinth seal mounting bolts, the steam ducting-to-bearing assembly bolts, and the spring plunger. The cracks were likely caused by stress corrosion cracking.

3.1.2.2 NGNP Implementation

Implementation Description

This lesson learned is specific to a circulator leaking because of stress corrosion cracking of the seal mounting bolts. There is nothing identified in the NGNP design concepts related to stress corrosion cracking of components in the circulator.

No DDNs are identified that are related to stress corrosion cracking of bolts.

Research and Development

Although not specifically identified with seal mounting bolts, stress corrosion cracking is a well known phenomenon and has been identified as a potential issue for a number of components subjected to an HTGR environment[79]. The INL R&D materials program identifies stress corrosion as an item that requires expansion of appropriate databases to support ASME BPV Code and Code Cases to cover unresolved regulatory issues for very high temperature service[80].

Summary

Lessons Learned	Status
At FSV circulator leaking because of stress corrosion cracking of the seal mounting bolts.	This is a design activity that will be addressed by the selected reactor supplier in the design phase. The INL R&D materials program will expand stress corrosion databases to support ASME BPV Code and Code Cases.

3.1.3 Use of Active Magnetic Bearings (HTR-10)

Cross reference to Section 313 of HTGR Lessons Learned NGNP[81].

3.1.3.1 Lessons Learned

HTR-10 currently uses a single-stage centrifugal compressor to circulate the helium through the primary loop. High-performance grease is used for the bearings supporting the compressor. In a subsequent HTR project (HTR-PM), active magnetic bearings (AMB) will be tested and used for the circulator. AMBs do not require lubrication, eliminating the possibility of the lubricant contaminating the helium coolant. Since there is no mechanical wear on the bearings, an AMB circulator theoretically has a longer duty lifetime and less maintenance.

3.1.3.2 NGNP Implementation

Implementation Description

Circulator designers/vendors, with direction and consultation from reactor suppliers, will determine whether AMBs are a viable option for the NGNP circulators. GA uses AMBs in their design along with a set of catcher bearings for starting and stopping.

GA identified DDN 14.01.01, “SCS Circulator Magnetic and Catcher Bearings Design Verification,” related to AMB bearings.

Research and Development

Close cooperation between the circulator designer, motor designer, and magnetic bearing designer will be required to ensure that the complex interactions between the rotating items and AMBs are addressed. Testing of AMB electrical connections and insulation will also be needed to confirm their operation in the helium environment.

Summary

Lessons Learned	Status
A new project (HTR-PM) will use AMBs.	This is a design activity that will be addressed by the selected reactor supplier (in conjunction with the circulator manufacturers) in the design phase. The Heat Transport Test Plan[82] identifies the need for testing AMBs. Currently the Heat Transport Test Plan has not been adopted by the INL R&D program.

3.1.4 Oil Ingress in Compressor (Peach Bottom Unit 1)

Cross reference to Section 3.1.4 of HTGR Lessons Learned NGNP[83].

3.1.4.1 Lessons Learned

At Peach Bottom near the end of Core 1's life, there was concern about oil ingress. It was shown that the oil ingress originated from the compressor used to circulate helium through the primary system. More specifically, the ingress started from the oil demister/filter, which removes any oil vapor and oil mist from the discharge in the compressor, and the oil lubricant. Approximately 100 kg of oil entered the reactor's primary system.

3.1.4.2 NGNP Implementation

Implementation Description

Oil that is used within components directly connected to the main system coolant loop is at risk of ingress into the primary circuit. Depending on the likelihood and design basis event, oil can impact the performance of the reactor.

No DDNs are identified related to oil ingress.

Research and Development

The Heat Transport Test Plan[84] identifies seals as a potential failure mechanism causing leaks. Testing is needed to determine the flow rates escaping the seal and to verify that seals work as expected under various pressures and temperature loading scenarios. Testing may also be needed to determine leak rates and required sealing gas flow rates. Currently the Heat Transport Test Plan has not been adopted by the INL R&D program.

Summary

Lessons Learned	Status
Oil ingress into primary system from a circulator compressor.	This is a design activity that will be addressed by the selected reactor supplier in the design phase. The INL has identified R&D tests that need to be performed for seals (most likely the seal manufacturers will perform the R&D tests).

3.1.5 Friction Damage (Dragon)

Cross reference to Section 3.1.5 of HTGR Lessons Learned NGNP[85].

3.1.5.1 Lessons Learned

Gas bearings were used in Dragon. It was discovered that at low speeds dry friction in an oxygen-free helium environment in the circulator bearings could lead to damage. Damage was avoided at speeds lower than the minimum by pressing helium as a hydrostatic lubricant during starting and stopping of the circulators.

3.1.5.2 NGNP Implementation

Implementation Description

No currently proposed HTGR designs have gas bearings. This lesson learned is no longer applicable to current HTGR design.

No DDNs are identified related to gas bearings.

Research and Development

There are no R&D identified activities related to gas bearings.

Summary

Lessons Learned	Status
At Dragon gas bearings could be damaged at low speeds.	Gas bearings are not included in current HTGR proposed designs. This lesson learned is no longer applicable.

3.2 Intermediate Heat Transfer

3.2.1 Intermediate Heat Exchanger Materials (HTR)

Cross reference to Section 3.2.1 of HTGR Lessons Learned NGNP[86].

3.2.1.1 Lessons Learned

Recent work associated with the advancement of heat exchangers for HTGR research recommends the use of Alloy 617 (nickel base) for temperatures above 850°C and Alloy 800H (iron-base) for temperatures below 850°C. This German research in the HTR project was able to demonstrate stress rupture behaviors of Alloy 617 and how carburization or decarburization occurs, depending on the materials used and on flow rates.

3.2.1.2 NGNP Implementation

Implementation Description

IHX materials selection requires close collaboration between the reactor suppliers and R&D programs. Not all of the current HTGR proposed designs are recommending Alloy 617 or Alloy 800H for the IHXs.

There are a number of DDNs related to IHX materials.

- AREVA DDN 2.2.2.1, "IHX Materials For both nickel-base alloys, the following issues need to be addressed: - Baseline mechanical property data, including creep-fatigue data."
- WEC DDN HTS-01-01, "Establish Reference Specifications for Alloy 617. Data needed include assessment of the current database and verification that at least three heats obtained for the IHX testing and qualification program conform to the procurement and quality assurance documentation."
- WEC DDN HTS-01-07, "Establish Reference Specifications for Alloy 230. Alloy 230 is a candidate material for heat exchangers and other HTS components used in hydrogen production and/or other very high temperature (850–950°C) applications."
- WEC DDN HTS-01-18, "Data Supporting Design Code Case. Data needed includes all information required to prepare the desired materials code cases for Alloy 800H and to resolve issues that may occur during further discussions with the ASME during the code case approval process. The IHX high-temperature primary to secondary system interface will be designed as an ASME BPV Code, Section III component and the alloys selected for this interface (Alloy 800H)."
- WEC DDN HTS-01-20, "Influence of Section Thickness on Materials Properties of Alloy 617 for heat exchangers. Assumption: Very thin material sections required for compact type IHXs."
- WEC DDN HTS-01-22, "Establish Reference Specification for Alloy 800H. An assessment is required to confirm that the present specifications for Alloy 800H, also considering manufacturing methods that will be employed, will produce thin-section materials suitable for the IHX."
- GA DDN N.13.02.01, "Effects of Primary Helium and Temperature on IHX Materials. Data are needed on the effects of elevated temperature corrosion from primary coolant helium impurities during normal operation and accident conditions on the material properties of metallic IHX candidate material base metal and weldments."

Research and Development

R&D is underway at the INL by the NGNP Materials program[87] to evaluate candidate materials for the IHX. The program is currently focused around Alloy 617 and Alloy 800H.

Summary

Lessons Learned	Status
German HTR project recommending the use of Alloy 617 and Alloy 800H for the IHX.	Final selection of the IHX material is a design activity. R&D is actively pursuing data on Alloy 617 and Alloy 800H in anticipation that these are the most likely materials to be selected for the NGNP.

3.2.2 Successful Operation at High Temperatures (HTTR)

Cross reference to Section 3.2.2 of HTGR Lessons Learned NGNP[88].

3.2.2.1 Lessons Learned

HTTR was able to successfully use and test a 10 MW(t) helical coil IHX. The heat transfer tubes and headers were made out of Hastelloy XR with the shell being made of 2¼ Cr-1Mo. The maximum operating temperature was 955°C for the heat tubes and 430°C for the outer shell. The maximum pressure rating was 4.8 MPa.

3.2.2.2 NGNP Implementation

Implementation Description

The INL is currently in negotiations with JAEA to provide various HTTR data for use by the NGNP Project. Acceptance of the data for use in NRC licensing and/or ASME BPV Code development is under review. JAEA has successfully operated the HTTR at similar temperatures and pressures intended for the NGNP.

No DDNs are identified related to the HTTR helical coil IHX.

Research and Development

The INL R&D program is currently pursuing Alloy 617 as the preferred material for the NGNP IHX, and there is a set of test plans associated with its codification into the ASME BPV Code and certification with NRC.[89]. Alloy 617 is being considered for approved for entry into ASME BPV Code, Section III, Division 5 rules (and specifically Section III, Division 1, Subsection NH for elevated temperatures).

Summary

Lessons Learned	Status
HTTR successfully used and tested an IHX made from Hastelloy XR and 2¼ Cr-1Mo at 955°C.	The INL R&D program is not investigating any of the HTTR IHX test materials (Hastelloy XR and 2¼ Cr-1Mo). The materials to be used for the IHX will be selected in the design phase. R&D is actively pursuing data on the Alloy 617 and Alloy 800H in anticipation that these are the most likely materials to be selected for the NGNP.

3.2.3 Helium Leakage in Secondary Loop (HTTR)

Cross reference to Section of HTGR Lessons Learned NGNP[90].

3.2.3.1 Lessons Learned

It was discovered through discussions with personnel who have toured the HTTR facility that there have been problems with helium leaks in the secondary helium coolant system. The helium leaked through flanged pipe connectors, which later needed to be welded to contain the helium.

3.2.3.2 NGNP Implementation

Implementation Description

Helium leakage can occur at flange connection and/or weldment points within primary and secondary loops. Experiences with the HTTR have shown that weldments were needed to replace the flange pipe connectors, requiring additional work to prevent unplanned leakage and sufficiently retain helium coolant.

One DDN was identified by GA related to helium leakage from flanged joints:

- GA DDN C.12.01.04, "Helium Seal Data for Bolted Closures. Demonstrate He leakage rate requirement for Vessel System flanged joints can be achieved for normal operating conditions."

Research and Development

Coolant leaks in both primary and secondary NGNP loops have been identified as a concern. The Heat Transport System Components Engineering Test Plan[91] identifies seals and their potential failure mechanisms such as excessive shaft vibration, pressure transients, and temperature transients occurring on either side of the dry gas seal. Testing is needed to determine the flow rates escaping the seal and to verify that seals work as expected under various pressure and temperature loading scenarios. Testing may also be needed to determine leak rates and required sealing gas flow rates. It is anticipated that this testing will be large scale. The Heat Transport Test Plan has not yet been adopted by the INL R&D program.

Summary

Lessons Learned	Status
HTTR encountered helium leaks in the secondary helium coolant system (IHX).	Further R&D is required to investigate the problem of helium leakage from flanges and pipe connectors (not just for IHXs). This is also a design activity that will be addressed by the selected reactor supplier in the design phase.

3.2.4 Steam Generator Integration Design (HTR-10)

Cross reference to Section 3.2.4 of HTGR Lessons Learned NGNP[92].

3.2.4.1 Lessons Learned

During in-service inspection of the SG at the HTR-10 facility, tube plates located in the SG tube box are accessible when the tube box header is opened. If the helium leakage rate is not acceptable, helium leakage inspection equipment can be used to find the leaking tube when the reactor is shut down. The leaking tube can then be plugged on the nonradioactive side in the cold leg tube box and in the hot leg tube box.

3.2.4.2 NGNP Implementation

Implementation Description

ASME BPV Codes specify in-service inspection of key components as critical to the longevity and operational performance of an SG within an HTGR operational environment. The NGNP design should be consistent with the applicable requirements of ASME BPV Code, Section XI on in-service inspections.

Westinghouse identified one DDN related to inspection needs for a SG:

- WEC DDN PCS-01-17, “Tubing Inspection Methods and Equipment. Test data are needed first to confirm the capability to inspect the steam generator tubing and, secondly, to determine inspection sensitivity. From this, an inspection program can be developed which is compatible with outage times. The capability to inspect the tubing and tube-to-tube welds in the NGNP steam generator tube circuits is desired to support safe and reliable operation of the units with high availability.”

Research and Development

No R&D are activities are identified related to the in-service inspection of a SG.

Summary

Lessons Learned	Status
Ability to perform in-service inspection of the SG.	This is a design activity that will be addressed by the selected reactor supplier in the design phase. The SG will be designed to the applicable requirements of ASME BPV Code, Section XI on in-service inspections.

3.3 Hot Duct and Cross Vessel

3.3.1 Hot Duct Materials (THTR)

Cross reference to Section 3.3.1 of HTGR Lessons Learned NGNP[93].

3.3.1.1 Lessons Learned

The German HTR project successfully used Alloy 617 as the material for the hot duct. There were concerns that the high cobalt content in Alloy 617 could generate radioactive Cobalt. It was found that Cobalt was not incorporated into the oxide scale so radioactive Cobalt would not therefore enter the hot gas circuit, even if the oxide spalled off.

3.3.1.2 NGNP Implementation

Implementation Description

Alloy 617 has been identified in NGNP design concepts as a potential candidate material for use in the hot duct and cross vessel. Material's property tests are identified in the INL R&D materials test program,[94] although none of the tests relate to the release of cobalt from alloy 617 during oxidation.

No DDNs are identified related to the generation of radioactive cobalt from Alloy 617.

Research and Development

No R&D activities are currently identified related to the generation of radioactive cobalt from Alloy 617, or any documentation that identifies Alloy 617 as a potential source of radioactive cobalt.

Summary

Lessons Learned	Status
On the German HTR project there was no Cobalt found in the oxide scale of the hot duct area.	There are no INL R&D documents that identify the use of Alloy 617 as a source of radioactive Cobalt. This is a design activity that will be addressed by the selected reactor supplier in the design phase.

3.3.2 Successful use of High Temperature Hot Duct (HTTR)

Cross reference to Section 3.3.2 of HTGR Lessons Learned NGNP[95].

3.3.2.1 Lessons Learned

The HTTR was able to successfully use a hot duct constructed with Hastelloy XR as the hot duct and 2¼ Cr-1Mo the cross vessel. The insulation material was a ceramic fiber composed of SiO_2 and Al_2O_3.

3.3.2.2 NGNP Implementation

Implementation Description

NGNP design concepts are not using the materials identified in this lesson learned. The INL R&D programs are not testing the materials identified in this lessons learned.

No DDNs are identified related to the testing of the properties of Hastelloy XR for use in the hot duct or 2¼ Cr-1Mo in the cross vessel.

Research and Development

No INL R&D activities are identified in the Materials program related to the testing of Hastelloy XR and 2¼ Cr-1Mo.

Summary

Lessons Learned	Status
HTTR successfully used Hastelloy XR and 2¼ Cr-1Mo in the hot duct.	The selection of materials for use in the hot duct and cross vessel is a design activity that will be addressed by the selected reactor supplier in the design phase. If the alternate materials are selected, further R&D work may be required for acceptance into the ASME BPV Code.

3.3.3 Loadings on Cross Vessel (HTR-10)

Cross reference to Section 3.3.3 of HTGR Lessons Learned NGNP[96].

3.3.3.1 Lessons Learned

The Cross vessel can come under variable loads. In HTR-10, the cross vessel connects the RPV with the SG pressure vessel. The cross vessel can experience variable loadings that include pressure, bolt forces, and temperature variations.

3.3.3.2 NGNP Implementation

Implementation Description

The cross vessel connects the RPV to SG or IHX pressure vessels. Loading and other mechanical concerns apply to the cross vessel since it is the coolant link between the reactor vessel and the SG. Reactor suppliers and NGNP R&D acknowledge that the material used for the cross vessel will be identical or similar to that used for the RPV and/or SG pressure vessel.

This is a design issue; loadings are calculated during the design phase. Currently, all vendors are recommending SA 508/533 as the material of choice for the NGNP pressure vessels and cross vessel.

No DDNs are identified related the loadings on the cross vessel.

Research and Development

No R&D are identified activities related to hot duct loadings.

Summary

Lessons Learned	Status
The hot duct will encounter various loadings.	This is a design activity that will be addressed by the selected reactor supplier in the design phase.

3.4 High Temperature Valves

3.4.1 Mitigating Air ingress Through Valves (HTR-10)

Cross reference to Section 3.4.1 of HTGR Lessons Learned NGNP[97].

3.4.1.1 Lessons Learned

From HTR-10 experience, high temperature valves can be susceptible to leaks, allowing air ingress. The life of the valves depends on the surfaces of the sealing parts. If any damage occurs or the matching between the disc and the seat is not rigorous, a leak will form. This type of leak could potentially grow and disable the valves.

3.4.1.2 NGNP Implementation

Implementation Description

As a source for air ingress, valves play an important role in HTGR operations. The maturity of higher temperature valves is high enough to not merit R&D beyond integration demonstrations. This is a design issue that will be addressed in the design phase.

Two DDNs are identified by GA as related to the testing of high temperature valves:

- GA DDN N.42.02.01, "High Temperature Steam Isolation Valves. In the event the High Temperature Isolation Valves (HTIVs) in the secondary system are identified to be part of the primary pressure boundary, the HTIVs would need to be classified as ASME BPV Code, Section III Class I components."
- GA DDN N.42.02.02, "High Temperature Isolation Valve Prototype Design Verification. Test data are required to provide validation of Helium pressure relief valve and pressure relief system performance when subjected to simulated environmental operating conditions, including installation conditions."

Research and Development

The Heat Transport Test Plan[98] identifies tests to be conducted for the primary (helium) side valves that will involve material property testing and some limited performance testing for the helium side. Currently the Heat Transport Test Plan has not been adopted by the INL R&D program. Testing will also be performed for flow verification and to determine performance of gaskets, packing, and seals. The importance of valves is identified in the NGNP Technology Development Roadmaps based on the pre-conceptual designs.[99]

Summary

Lessons Learned	Status
High temperature valves can be susceptible to leaks allowing air ingress.	This is a design activity that will be addressed by the selected reactor supplier in the design phase. Specific valve testing and R&D will most likely be conducted by a valve manufacturer.

4. POWER CONVERSION SYSTEM

4.1 Steam Generator

4.1.1 Cracks/Leakage in Steam Generator (FSV)

Cross reference to Section 4.1.1 of HTGR Lessons Learned NGNP[100].

4.1.1.1 Lessons Learned

FSV experience shows that SGs leak for a number of reasons. A leak in the superheated steam section at FSV in one of the SG modules occurred late in November 1977. The presence of the leak was readily detectable because of a gradual rise in moisture content of the primary helium coolant. Several SG tubes leaked purified helium out of the main coolant system.

4.1.1.2 NGNP Implementation

Implementation Description

Reactor designers are aware of the fact that SGs leak. The use of robust weldments is an important factor to reduce the likelihood of leakage in the SG. The design of the SG to facilitate easier in-service inspections would mitigate the problem of late leak detection. The selection and design of the SG is a design issue that the reactor supplier will address in the design phase.

No DDNs are identified related to leakage from SGs.

Research and Development

Weld and component requirements are discussed in the NGNP Steam Generator and IHX Materials R&D Plan[101]. ASME BPV Code requires pressure testing (in some cases helium leak testing) as part of welding procedures. ASME BPV Code Case work is also discussed in the plan, including safety considerations for primary to secondary leakage. The candidate material by the INL R&D Materials program for the NGNP SG is Alloy 800H. A panel of experts in areas related to the U.S. NGNP design assessed modular moisture ingress events for an HTGR using a Phenomena Identification and Ranking Table (PIRT) process. This assessment is documented in the report, Assessment of NGNP Moisture Ingress Events[102]. One of the report’s conclusions was: “Considering resource limits and the lack of more detailed NGNP design information available for this assessment, many of the possible sequence options and design variations were not covered. As the design progresses, the assumptions should be revisited in any subsequent PIRT-like activities.”

Summary

Lessons Learned	Status
FSV experience shows that SGs leak for a number of reasons.	This is a design activity that will be addressed by the selected reactor supplier in the design phase. INL R&D will continue with weldment leak tests per ASME BPV Code requirements.

4.1.2 Tube Rupture (AVR)

Cross reference to Section 4.1.2 of HTGR Lessons Learned NGNP[103].

4.1.2.1 Lessons Learned

The AVR was out of service for long periods because of steam generator tube leaks. In May 1978, the circulator for the AVR flooded with approximately 27.5 tons of water. The direct cause of the flooding was a leak within the SG superheater tube. Water also got into the circulators oil lubricating systems, which required extensive cleaning, rinsing, and moisture removal, resulting in a several-month delay in reactor operation. Because the SG was located directly above the reactor vessel, the core and internals were found to be very wet.

4.1.2.2 NGNP Implementation

Implementation Description

Nothing in the NGNP design concepts specifically deals with the prevention of tube ruptures. This issue of tube integrity is a design issue that will be addressed in the design phase by the selected reactor supplier. The materials used in the tubes will comply with ASME BPV Code requirements.

The problem of water ingress from a tube leak is mitigated in the current design concepts by locating the SG below the RPV to eliminate the siphoning effect. The provision of installing a drain in the base of the RPV will also speed up the removal of water from the RPV after an event. In addition, the circulator is located on top of the SG. ASME BPV Code, Section XI inspections will be part of the implementation.

No DDNs are identified related to tube leaks.

Research and Development

There is a text matrix for Alloy 800H Tube Burst Tests (A15) that are identified in the SG and IHX development test plans.[104] These tests will be performed on a number of specimens (tube) at 760°C in air.

The Heat Transport Test Plan[105] identifies the need for tube testing. Material properties testing will likely need to investigate the performance of the bimetallic weld between the 800H and 2¼ Cr-1 Mo tubes. Experimental scale flow testing of the various SG components, including the tubes and tube supports, is recommended. This testing may be performed in a hot flowing loop or, more likely, in an air loop with proper scaling factors to provide representative flow conditions. The INL R&D program has not currently adopted the Heat Transport Test Plan.

Summary

Lessons Learned	Status
The AVR was out of service for long periods because of tube leaks.	This is a design activity that will be addressed by the selected reactor supplier in the design phase. Some R&D activities will provide information on tube material properties. ASME BPV Code, Section XI inspections will be part of the implementation.

4.1.3 Hot Steam Headers for the Steam Generators (THTR)

Cross reference to Section 4.1.3 of HTGR Lessons Learned NGNP[106].

4.1.3.1 Lessons Learned

The German Thorium Hochtemperatur Reaktor (THTR) project found that Alloy 800 (formerly known as Alloy 800 Grade 1) has a higher specified yield strength than Alloy 800H (formerly known as Alloy 800 Grade 2). The THTR project used Alloy 800 in the steam header of the SG. After the material behavior was evaluated, the specified yield strength of the Alloy 800 material (used in the header) was higher than Alloy 800H material.

4.1.3.2 NGNP Implementation

Implementation Description

This is a design issue that will be addressed by the selected reactor designer during the design phase. The extension of Alloy 800H material properties in ASME BPV Code, Section III, Division 1, subsection NH is currently underway for a service life up to 500,000 hours and up to an operating temperature range between 850 - 900°C.

No DDNs are identified related to the properties of the materials used in a steam header of an SG.

Research and Development

No R&D activities are currently identified related to the properties of the steam header used in an SG.

Summary

Lessons Learned	Status
THTR found Alloy 800 Grade 1 to have a higher specified yield strength than Alloy 800H Grade 2.	This design activity will be addressed by the selected reactor supplier in the design phase. The SG will be designed according to ASME BPV Codes. The extension of Alloy 800H material properties in ASME BPV Code, Section III, Division 1, subsection NH is currently underway for a service life up to 500,000 hours and up to an operating temperature range between 850 - 900°C.

4.1.4 Fatigue Analysis (HTR-10)

Cross reference to Section 4.1.4 of HTGR Lessons Learned NGNP[107].

4.1.4.1 Lessons Learned

A study conducted on HTR-10 showed that shutdowns can cause fatigue in the SG. This fatigue is from the thermal stress of the different temperatures in the primary and the secondary loop. The temperatures are 430 and 100°C for the primary and secondary circuit, respectively, during shutdown and startup conditions. A fast startup is performed after shutdown to save time and quickly reach full power, but does not allow enough time for the primary circuits to cool down.

4.1.4.2 NGNP Implementation

Implementation Description

This lesson learned is specifically about reactor operations and impacts service life. Experience with HTR-10 identified that shutdowns (and fast startups) can cause fatigue in the SG. There is nothing in the NGNP design concepts related to reactor shutdown operations. The structural integrity of the SG is a design issue to be addressed by the selected reactor supplier.

No DDNs are identified related to reactor cycles and their affect on the SG.

Research and Development

No R&D activities are currently identified related to the shutdown and startup of the reactor with regards to its affect on an SG.

Summary

Lessons Learned	Status
Reactor shutdown and quick startups can cause fatigue in the SG.	This is an operational issue that impacts service life and should be addressed in the operational phase of the project. It is also a design issue to be addressed in the design phase.

4.1.5 Materials Used and Migration of Tritium (Peach Bottom Unit 1)

Cross reference to Section 4.1.5 of HTGR Lessons Learned NGNP[108].

4.1.5.1 Lessons Learned

Peach Bottom demonstrated the successful use of Alloy 800H in an SG and gathered tritium permeation data. Tritium permeation was shown to be independent of the operating temperature, which only pertains to certain materials at certain temperature ranges. It was also shown that, if the surface film on the coolant side was removed, the tritium permeation would actually be lowered. NGNP is using this data as part of the High-Temperature Materials Qualification Program.

4.1.5.2 NGNP Implementation

Implementation Description

Alloy 800H is well known for its use in previous HTGRs and its use is considered by GA, AREVA, and researchers regarding its importance for use in the NGNP. Tritium is one of the fission products present with HTGRs, and much research is underway to capture it from the primary and, if necessary, secondary coolant loops. From January 5 to March 21, 2010, JAEA operated the HTTR at full power and temperature, recording plant conditions and collecting tritium data to complete a mass balance of the entire plant. Tritium concentrations were measured in the primary helium cooling system, secondary helium cooling system, pressurized water cooling system, auxiliary water cooling system, and containment vessel.

There were two DDNs related to tritium migration in SGs:

- AREVA DDN 4.1.4.1, "FP Transport. Models for: The assessment of product activation in the primary circuit (in particular tritium and carbon-14). Investigation of tritium migration and control in SG and secondary water loops."
- AREVA DDN 4.1.4.1b, "FP Transport. Modeling of Tritium Migration and Control in SG and Secondary Water Loops. Models for tritium migration and control in SG and secondary water loops."

Research and Development

The migration of tritium is being investigated under the R&D Test Control Plan – High Temperature Hydrogen Permeation through Nickel Alloys.[109] The permeation tests are performed at the Safety and Tritium Applied Research Facility at the Advanced Test Reactor Complex.

Summary

Lessons Learned	Status
Tritium permeation was shown to be independent of the operating temperature, which only pertains to certain materials at certain temperature ranges.	This will be addressed by INL R&D on tritium migration and by the reactor designers in the design phase in the selection of the materials and design for the SG.

4.1.6 Dynamic Stresses and Side Flow Maldistribution (AGR)

Cross reference to Section 4.1.6 of HTGR Lessons Learned NGNP[110].

4.1.6.1 Lessons Learned

The AGR's SG had some trouble with side flow maldistribution, which lowered the power rating to 58% and caused the SG to "…operate outside of the limits imposed by material properties, bimetallic weld and water/steam side stress corrosion concerns." AGR was able to increase the power rating up to 82% after a reorificing effort within the feed water tube sheet.

4.1.6.2 NGNP Implementation

Implementation Description

As part of the design process, NGNP HGTR suppliers will specify the criteria to address various stresses and flow issues for the SGs. Manufacturers of SGs will produce the components per NGNP suppliers' requirements and specifications and ASME BPV Code.

There were no DDNs identified related to side flow maldistribution for an SG.

Research and Development

The Heat Transport Test Plan[111] identifies the SG Model and High Temperature Heat Exchanger Facility (ST-1312) as the place to test thermal-hydraulic and vibration performance, as well as high temperature effects to test steam isolation valves. Currently the Heat Transport Test Plan has not been adopted by the INL R&D program.

Summary

Lessons Learned	Status
Side flow maldistribution lowered the power rating in the SG.	This is an SG design issue that will be addressed by the selected reactor supplier and equipment vendor in the design phase.

5. BALANCE OF PLANT (BOP)

5.1 Fuel Handling System

5.1.1 Design Issues (FSV)

Cross reference to Section 5.1.1 of HTGR Lessons Learned NGNP[112].

5.1.1.1 Lessons Learned

There was a potential for fuel damage during fuel handling maneuvers at FSV. On November 24, 1981, a grappling device was used to remove a core restraint device called a "Lucy Lock" from the top of the core, and the device was dropped onto the top of the core (without damage to the fuel). GA redesigned the fuel handling system to improve efficiency and reliability in future refueling operations.

5.1.1.2 NGNP Implementation

Implementation Description

Fuel handling systems for the NGNP will be designed during the design process, and regulatory agencies will have input on and oversight of those systems.

AREVA DDN 3.3.3.0, "Fuel Handling System – Material/Subcomponent Testing," was identified as being related to the fuel handling system.

Research and Development

No R&D activities were identified related to fuel handling.

Summary

Lessons Learned	Status
The grappling device at FSV in the fuel handling system failed and had to be redesigned.	This is a design issue that will be addressed by the selected reactor supplier in the design phase.

5.2 Instrumentation and Control

5.2.1 Instrumentation Failure (FSV)

Cross reference to Section 5.2.1 of HTGR Lessons Learned NGNP[113].

5.2.1.1 Lessons Learned

FSV experienced instrument failure, which might have an effect on operations. There were numerous events during the operation of FSV that were related to either a moisture incursion or a failure of a moisture detection system. The distribution of the instrumentation and controls events were categorized into four general areas:

- Inoperable instruments that were out of calibration or had drifted from their correct set points
- Instruments were moved, disturbed, or otherwise subjected to physical motion that produced an erroneous or false signal from the instrument
- Instruments failed, sent a false signal, or tripped because of a short between contacts or because the instrument had dirty contacts
- Instrument "noise" or a spike on an instrument's output signal.

5.2.1.2 NGNP Implementation

Implementation Description

Instrumentation is a vital and integrated component of all NGNP SSC installations and operations. New instrumentation and controls using digital technology still need to be developed and receive regulatory approval. NGNP suppliers will include thorough instrumentation in all areas for the plant to provide the necessary information feeds to control operating systems.

There are a number of DDNs related to instrumentation.

- AREVA DDN 3.3.5.0, "Instrumentation. Examples of R&D which might be envisioned: Neutron flux detectors: Some R&D and qualification efforts may be desirable to select detector technology and verify adequate sensitivity and lifetime. Temperature Measurements."
- AREVA DDN 3.3.5.0a, "Instrumentation. R&D and Qualification of Pt-Rh Thermocouples. Platinum-Rhodium (Pt-Rh) thermocouples are used for high-temperature measurements as they can perform in inert or lightly oxidized environments (as the NGNP environment) with temperatures as high as 1300°C."
- AREVA DDN 3.3.5.0b, "Instrumentation. Qualification Testing in Helium. The Primary Loop Instrumentation required for plant protection and Plant Control System measurements are commercially available but have not been used at NGNP operating pressures and temperatures."
- GA DDN C.11.01.01. "Control Rod Instrumentation and Control Verification."
- GA DDN C.14.04.03. "SHE Instrumentation Attachment Test."
- GA DDN C21.01.04. "Verify Fuel Handling System Instrumentation and Control."
- GA DDN C31.01.01. "Verify Helium Mass Flow Measurement Instrumentation."
- GA DDN C31.01.02. "Verify Conduction Cooldown Temperature Monitoring Instrumentation."
- GA DDN C34.01.01. "Verify Core Inlet and Outlet Helium Temperature Measurement Instrumentation."
- GA DDN C.34.01.02. "Verify Plateout Probe Operation."
- GA DDN N.33.01.01. "Verify Moisture Monitor Instrumentation and SG Isolation and Dump."
- GA DDN N34.01.04. "Verify Reliability and Availability of Digital Hardware Based Plant Control System."
- WEC DDN PCS-01-14. "Instrumentation Attachment Test."

Research and Development

There are no identified INL R&D activities under the current programs related to instrumentation testing.

Summary

Lessons Learned	Status
FSV experienced instrument failure, which can have an effect on operations.	The reactor suppliers will address this lesson learned through design. There is a Heat Transport Test Plan[114] prepared by NGNP engineering that identifies the need for a test loop to assess the operation of instruments that are likely to be used in the NGNP. Currently the Heat Transport Test Plan has not been adopted by the INL R&D program. The INL R&D programs need to incorporate and build upon the Heat Transport Test Plan and DDNs to develop an instrumentation testing program.

5.2.2 Core Temperature Instrumentation (AVR)

Cross reference to Section 5.2.2 of HTGR Lessons Learned NGNP[115].

5.2.2.1 Lessons Learned

One concern with the AVR is that there was not enough instrumentation to monitor the core temperature of the fuel. It is theorized that the fuel exceeded 1600°C, which may have generated the graphite dust, caused the TRISO layers to fail, thus releasing fission product gases.

5.2.2.2 NGNP Implementation

Implementation Description

The current first-of-a-kind NGNP design concepts do not address temperature monitoring capability in the core. The NGNP design may benefit by evaluating this capability in the core. This is very challenging to implement, but determining new methods of core temperature monitoring may be an area for future research.

A number of DDNs related to core temperature instrumentation were found:

- AREVA DDN 3.3.5.0. "Instrumentation. Examples of R&D which might be envisioned: Neutron flux detectors: Some R&D and qualification efforts may be desirable to select detector technology and verify adequate sensitivity and lifetime. Temperature Measurements."
- AREVA DDN 3.3.5.0a. "Instrumentation. R&D and Qualification of Pt-Rh Thermocouples. Platinum-Rhodium (Pt-Rh) thermocouples are used for high-temperature measurements as they can perform in inert or lightly oxidized environments (as the NGNP environment) with temperatures as high as 1300°C."
- GA DDN C34.01.01. "Verify Core Inlet and Outlet Helium Temperature Measurement Instrumentation."

Research and Development

No R&D activities are currently identified related to core temperature instrumentation.

Summary

Lessons Learned	Status
Not enough instrumentation at AVR to measure core temperature.	The reactor suppliers will address this lesson learned through design. Future R&D research into high temperature monitoring sensors that are capable of handling temperatures in excess of 1600°C would benefit reactor suppliers.

5.3 Other

5.3.1 Helium Purification System Issues (FSV)

Cross reference to Section 5.3.1 of HTGR Lessons Learned NGNP[116].

5.3.1.1 Lessons Learned

At FSV, moisture had bypassed the chiller that was used to precipitate water from the helium purification system. Excessive moisture caused the operating helium purification train at FSV to ice up, causing problems with the purification system.

5.3.1.2 NGNP Implementation

Implementation Description

This lesson learned is specific to excessive moisture entering the helium purification train and causing ice-up. There is nothing in the NGNP design concepts related to this lesson learned. This is a issue to be addressed in the design phase.

No DDNs are identified related to helium purification systems reliability or excessive moisture.

Research and Development

No INL R&D activities are identified related to the helium purification system and excessive moisture. A panel of experts in areas related to the U.S. NGNP design assessed modular moisture ingress events for an HTGR using a Phenomena Identification and Ranking Table (PIRT) process. This assessment is documented in the report, Assessment of NGNP Moisture Ingress Events[117]. One of the report's conclusions was: "Considering resource limits and the lack of more detailed NGNP design information available for this assessment, many of the possible sequence options and design variations were not covered. As the design progresses, the assumptions should be revisited in any subsequent PIRT-like activities."

Summary

Lessons Learned	Status
Excessive moisture in the helium purification train caused the system to ice up.	This is a design issue. The reactor suppliers will address this lesson learned in the design.

5.3.2 Chemical Cleanup (Peach Bottom Unit 1)

Cross reference to Section 5.3.2 of HTGR Lessons Learned NGNP[118].

5.3.2.1 Lessons Learned

Peach Bottom demonstrated a potential method for cleaning tritium from the primary loop. Helium gas leaving the SG flows into the chemical cleanup system. The first part of the cleanup system is the oxidizer, in which tritiated hydrogen gas becomes tritiated water. The tritiated water is then removed by a molecular sieve-dehydrator. The rest of the helium continues from the sieve dehydrator to the fission product trapping system.

5.3.2.2 NGNP Implementation

Implementation Description

The cleaning of chemical impurities from helium is identified by reactor suppliers and researchers as a need because the impurities will impact performance of the primary coolant.

No DDNs are identified related to cleaning tritium from the primary loop.

Research and Development

There are no identified R&D activities related to the removal of tritium.

Summary

Lessons Learned	Status
Peach Bottom demonstrated a potential method for cleaning tritium from the primary loop.	This is a very specific lesson learned related to tritium cleanup. The reactor suppliers need to address tritium permeation in the design phase. The R&D activities are currently confined to understanding tritium transport and not tritium cleanup.

5.3.3 Helium Purification Piping (Dragon)

Cross reference to Section 5.3.3 of HTGR Lessons Learned NGNP[119].

5.3.3.1 Lessons Learned

Several leaks were found in the Dragon helium purification system. These leaks were found to have been caused by chloride corrosion. It was also found that the leaks had occurred at points in the piping marked with polyvinyl chloride (PVC) tape.

5.3.3.2 NGNP Implementation

Implementation Description

This is a very specific lesson learned related to an incident where PVC tape came into contact with piping that was susceptible to chloride corrosion from the tape. There is nothing in the NGNP design concepts related to this lesson learned (as would be expected since this is a design/operations/compatible materials issue).

No DDNs are identified related to the piping of the helium purification system.

Research and Development

No R&D activities are currently identified related to helium purification piping.

Summary

Lessons Learned	Status
Chloride corrosion from PVC tape caused leaks in the helium purification system piping.	This lesson learned will be addressed in the design and operations phases.

5.3.4 Auxiliary Systems Failures (FSV)

Cross reference to Section 5.3.4 of HTGR Lessons Learned NGNP[120].

5.3.4.1 Lessons Learned

Experience at FSV showed that various areas of the auxiliary systems have the potential to fail with significant consequences to plant operations. For example, a compressor malfunctioned in the helium purification system and charred and burned cables on several components of support systems to the helium circulators. Another event involved the failure of a joint in the circulating water system, resulting in flooding in a pump room.

5.3.4.2 NGNP Implementation

Implementation Description

Auxiliary systems provide the balance-of-plant operations to support the nuclear heat supply system, heat transport system, power conversion system, and possible process heat systems. The reactor designers should review the auxiliary systems such that they are either decoupled from directly (or immediately) affecting plant operations or incorporate redundant auxiliary systems to minimize operational downtime.

No DDNs are identified related to auxiliary systems.

Research and Development

No R&D activities are currently identified related to auxiliary systems.

Summary

Lessons Learned	Status
Experience at FSV showed that various areas of the auxiliary systems have a potential to fail with significant consequences to plant operations.	A review is required by reactor designers to reduce the operational dependence/coupling on auxiliary systems or incorporate redundant systems. This is a design phase activity.

5.3.5 Electrical Arcing (FSV)

Cross reference to Section 5.3.5 of HTGR Lessons Learned NGNP[121].

5.3.5.1 Lessons Learned

Moisture in the duct cooling system caused electrical arcing. There were no moisture detectors within the building cooling system.

5.3.5.2 NGNP Implementation

Implementation Description

This may have been prevented if moisture detectors had been installed in the system, which would have allowed a rapid system repair. NGNP would benefit by considering proper placement of sensors to ensure timely response.

The reactor suppliers need to address this lesson learned in the design phase and consider the placement of moisture sensors to prevent arcing.

No DDNs are identified related to moisture electrical arcing.

Research and Development

No R&D activities are currently identified related to moisture and electric arcing. A panel of experts in areas related to the U.S. NGNP design assessed modular moisture ingress events for an HTGR using a Phenomena Identification and Ranking Table (PIRT) process. This assessment is documented in the report, Assessment of NGNP Moisture Ingress Events[122]. One of the report's conclusions was: "Considering resource limits and the lack of more detailed NGNP design information available for this assessment, many of the possible sequence options and design variations were not covered. As the design progresses, the assumptions should be revisited in any subsequent PIRT-like activities."

Summary

Lessons Learned	Status
Electrical arcing because of moisture could have been detected with the correct placement of sensors.	This lesson learned will be addressed in the design phase by the selected reactor supplier.

5.3.6 High Winds (FSV)

Cross reference to Section 5.3.6 of HTGR Lessons Learned NGNP[123].

5.3.6.1 Lessons Learned

On December 8, 1983, high winds at the FSV site caused a fire detector to come loose and malfunction, activating the RAT deluge system. This incident illustrates the susceptibility of the plant auxiliary transformers to externally induced events.

5.3.6.2 NGNP Implementation

Implementation Description

Meteorological environments and conditions are briefly discussed by the reactor suppliers. A review of the robustness and integrity of external sensors linked to plant auxiliary systems would be of benefit for NGNP operations.

No DDNs are identified related to high winds or meteorological conditions.

Research and Development

No R&D activities are currently identified related to high winds and weather.

Summary

Lessons Learned	Status
High winds caused an external sensor to malfunction and activate the RAT deluge system.	This lesson learned will be addressed in the design phase by the selected reactor supplier.

6. CONCLUSION

The lessons learned discussed in this document differ in their relevance to the NGNP project. Some are highly relevant to the NGNP design, and will influence the design as it progresses in detail; others have influenced early pre-conceptual and conceptual design; while others are no longer relevant or may become irrelevant in the future via design choices. A number of the lessons learned are issues that need to be addressed in the design phase of the project and require no R&D activities. There are lessons learned that require R&D activities to investigate the associated issues. Those lessons learned that have associated DDNs will be addressed by either R&D activities or during the detailed design phase (or both).

The NGNP Project to-date has addressed the identified lessons learned to differing degrees. Some of the lessons learned have DDNs associated with them and have related R&D activities being performed or identified. Some have been addressed in on-going design efforts. Others may have not have had any of the above in place to address the issues related to the lessons learned. In order to ensure that all of the lessons learned (that are relevant and applicable to NGNP) are captured and addressed appropriately, entries will be made in the NGNP risk register that will track the lessons learned item, risk handling strategy, and actual risk reduction throughout the lifecycle of the NGNP project.

There are a number of plant area topical issues where similar lessons learned were identified at several of the HTGRs that were evaluated and for which no activities have been identified that will completely mitigate or overcome the associated issues. These issues will need to be considered in future Project R&D and/or design activities. These issues include:

- Moisture Ingress – The sequence of events leading to moisture ingress into the primary system that are evaluated in 'Assessment of NGNP Moisture Ingress Events'[124] will need to be revisited once plant design has progressed.

- Helium Leakage – The seals for bolted or other mechanical closures will need to be addressed during detailed component design.

- Dust – 'HTGR Dust Safety Issues and Needs for Research and Development'[125] comprehensively summarizes the current knowledge regarding graphite dust in the primary system. This will need to be revisited once plant design has progressed.

- Internal Inspections per ASME BPV Code, Section XI – Detailed plant and component designs will need to account for in-service inspection requirements. Some components of concern include inner reflectors, steam generator tubing, and IHX heat transport surfaces.

- Plant instrumentation needs – There are a number of DDNs associated with instrumentation, but to-date there are no associated R&D activities that have been identified. Specific needs will need to be addressed once plant design has progressed.

- Various SSC Development – There are a number of lessons learned associated with plant SSCs for which mitigation has been identified in the Heat Transport Test Plan[126], but no activities have been initiated. These include lessons learned on active magnetic bearings (AMB), component oil seals, helium seals in the secondary system, and secondary system and Balance of Plant (BOP) instrumentation. These will need to be considered once plant design has progressed.

- The Heat Transport Engineering Test Plan identified above is not part of any R&D programs currently progressing NGNP technology. Unless the test plans and activities identified in this document are adopted by R&D, there will be a gap in the technological advancement of several key components.

There are a number of lessons learned in which alternate materials to those currently being addressed by NGNP R&D efforts are identified. It is generally considered that the R&D program plans are addressing the likely materials candidates, but this will need to be revisited as the plant and component designs have progressed.

A common concept for the mitigation of the issues identified in the lessons learned is that the mitigation will be addressed in the design phase. The design of the NGNP needs to progress from its current status in order to better inform and focus the needed R&D efforts.

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