NC2I-R

Coordination and Support Action

Co-funded by the European Commission under the

Euratom Research and Training Programme on Nuclear Energy

within the Seventh Framework Programme

Theme: FISSION-2013-2.4.1

Support to the emergence of a possible European Research Initiative on co-generation

Grant Agreement Number: 605167

Start date: 01/10/2013 Duration: 24 Months

D3.32 – Needs of R&D in support of the safety demonstration

Authors: L. Koszuk, C. Pohl, O. Baudrand

NC2I-R – Contract Number: 605167

Document title	Needs of R&D in support of the safety demonstration
Author(s)	Lukasz Koszuk (NCBJ), Christoph Pohl (TUV), O. Baudrand IRSN
Number of pages	53
Document type	Deliverable
Work Package	WP3
Document number	version 0
Issued by	NCBJ
Date of completion	
Dissemination level	Restricted

Summary

This report presents the synthesis of the R&D needs in the fields which could be necessary to support the safety demonstration of a demonstrator of HTR performing cogeneration. It is grounded on the analysis of lessons learned from past and present high temperature gas-cooled reactors. In particular this report integrates 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. The report takes also into account the work performed in the European projects on HTRs, and specifically the results of the ARCHER project (2011 - 2015).

As the demonstrator promoted by NC2I platform aims at building as far as possible on already available technologies and materials, this report presents also the qualification efforts envisaged for the licensing and potential tracks for future VHTR reactor.

Approval

Version	First author	WP leader	Project Coordinator
0	L. Koszuk, NCBJ	O. Baudrand, IRSN	T.Jackowski, NCBJ

Distribution list

Name	Organisation	Comments
P. Manolatos	EC	
All beneficiaries	NC2I-R	Through internet workspace

Table of contents

1 Introduction

This report is a synthesis of the main R&D subjects which are deemed to be needed to substantiate the safety demonstration of a future HTR demonstrator. It is written on the basis of the experience gained from the operation of gas cooled reactors. This experience has been collected and reviewed in particular by the Independent Technology Review Group (ITRG) of experts mandated by the NGNP project and in European projects (RAPHAEL). Although dating back to the beginning of the years 2000, the ITRG recommendations remain valid, at least to figure out the main domains liable for technology development and qualification. Since the work of the ITRG, significant progress has been made on the items identified, and some selected priority R&D subjects have been addressed in the ARCHER project (2010-2015). So, in the following chapters the results obtained by the ARCHER project are recalled and a list of remaining areas for R&D is proposed. To complete this approach, the efforts to be paid for the qualification of components are also mentioned, given that qualification plays also an important role in the plant licensing.

2 Recall of the R&D items derived from the operational feedback

In the table below the status of main technology challenges related to safety established from the analysis of the operational feedback and achievements of the recent projects (HTTR, NGNP, HTR-10, etc.) are summarized. The rational for this list is detailed in the annex 1.

Technology	Status	Comments
Moisture ingress	Mastered	Modeling tools available: some qualification needed Primary helium purification system should be designed to cope with moisture ingress Protection system integrates moisture monitoring
Insulation of primary circuit	Mastered by design and qualification	Insulation materials and fastening technology to be qualified
Sealing	Under improvement (current projects)	Important to reduce activity in the reactor hall
Helium flow bypass	R&D needed	Issue more important for block type reactor Needed to better estimate temperature distribution in the core and reflectors and to minimize the risks of hot spots
Helium flow induced vibrations	Mastered by design and qualification	Difficulty to design reduced scale loops representative of the reactor
Pipe corrosion	Mastered	Material qualification : up to date standards to be taken into account
Inspection of core internals	Under investigation	Needs for devices able to perform periodic inspection of

Technology	Status	Comments
	R&D needed	the internal structures
Graphite and graphite wear	Under investigation R&D needed	Results of INL demonstrate fabrication process for high quality graphite
Pebble Bed Graphite Dust	Under investigation R&D needed	Essential for source term evaluation
Prismatic Graphite Dust	Under investigation R&D needed	Essential for source term evaluation
Graphite under irradiation	Mastered	See ARCHER findings and database Subject for qualification tests
Cracks in fuel blocks caused by tensile stress	Under investigation R&D needed	Test and code development for new graphite grades Promising results from INL program
Fuel performance (fuel integrity, fission product release rate)	Under investigation	Several recent irradiation tests (UO_2 and UCO) confirm past results
Helium purification / fission product trapping	Mastered	
Moisture Effects on Reactivity Contr. Syst.	Mastered by design	
Control rods design and operation	Mastered	Need for qualification
Reactivity Cavity Cooling System	Mastered by design	Need for qualification (heat exchange coefficient, global performance, code qualification)
Materials for heat exchanger or steam generator (second barrier)	Mastered	Existing materials adapted for a demonstrator. Development of design rules needed especially for lifetime investigation
Steam generator design	Mastered	Know how has been partially lost: need for new design and fabrication rules / qualification
Hot duct materials	Mastered	Solutions developed and tested in the past Design rules to be developed
Design of cross vessel	Mastered	
High temperature valves	Under development	Existing designs Need for qualification

Technology	Status	Comments
Steam Generator Leak	Mastered by design	A safety approach has to be built to cope with steam leakage ranging from small leaks to tube rupture
Tritium production and migration	Under investigation	Capacity of retention in the fuel and graphite to be characterized Materials to be qualified
Fuel handling system	Mastered	
Instrumentation for core temperature evaluation	Under development R&D needed	Monitoring of safety margins Support for accidental conditions monitoring
Helium purification system	Mastered	
Reliability of auxiliary systems	Mastered by design and qualification	
Resistance to external hazards	Mastered by design	

3 Findings of the Independent Technology Review Group (ITRG)

A review of existing VHTR Technology was performed from November 2003 through April 2004 by a group of 26 experts and professionals in the HTR nuclear industry, known as the Independent Technology Review Group. The purpose of the ITRG was to conduct a review of technology alternatives for meeting the functional objectives for the NGNP. Its report, “Design Features and Technology Uncertainties for the Next Generation Nuclear Plant”, enumerated recommendations for the following areas: fuel development, reactor outlet temperature, power conversion concept, hydrogen production capability, and design uncertainties.

The recommendations for further R&D activities were as follows [2]:

- Fabrication, testing, and qualification of coated-particle designs and manufacturing processes that have the most extensive worldwide experience base (UO_2 kernel).
- Increasing fuel burnup to 10% fission per initial metal atom.
- Both uranium oxide (UO_2) and uranium oxy-carbides (UCO) should be to determine which performed best in thermal hydraulics and neutronics.
- A comprehensive licensing strategy need to be developed.

In the report issued by the ITGR, 59 items were analyzed which are in majority related to the Nuclear Heat Supply System and can be summarized in the following three areas:

1. Ingress or leakage events such as moisture ingress
2. Primary coolant flow issues such as bypass flow and flow induced vibrations
3. Fuel performance, fission product release, and graphite dust generation.

Important lessons were also learned in other areas not directly applicable to reactor components such as human error, licensing issues, and safety features.

4 The ARCHER European project

In line with the Sustainable Nuclear Energy Technology Platform (SNETP), Strategic Research Agenda (SRA) and Deployment Strategy (DS), the ARCHER project, has been established to extend the state-of-the-art European (V)HTR technology basis with generic technical effort in support of nuclear cogeneration demonstration. The partner consortium consisted of representatives of conventional and nuclear industry, utilities, Technical Support Organizations, R&D institutes and universities. The ARCHER project has been completed in January 2015.

In the field of R&D, the ARCHER project aimed at tackling items fundamental for safety demonstration and licensing which have been identified in previous European projects (RAPHAEL, EUROPAIRS). It brought elements to resolve some of the issues listed in the paragraph 2. So, the following issues have been investigated in ARCHER:

- graphite behaviour under irradiation (PIE on samples of recent graphite grades),
- interaction between metallic fission products and silicon carbide coatings (I and Ag),
- graphite dust characterization and wearing tests on structural and matrix graphite,
- study and modelling of dust transfer in the primary circuit,
- analysis of air ingress experience in NACOK test facility,
- flammability studies of gas mixtures potentially produced in case of air/water ingress in primary circuit,
- hot spot analysis in pebble beds,
- testing of materials used for heat exchanger, steam generator and other high temperature components,
- characterisation of the inventory of the fission products in irradiated particles (PIE) and code validation,
- reheating tests of irradiated pebbles.

Finally, ARCHER brought elements to resolve some of the issues listed in the paragraph 2. In this frame, the following chapters present a synthesis of the qualification and R&D needs for a demonstrator which take into account the above mentioned work performed in ARCHER. Some tracks are also envisaged for a future VHTR.

5 R&D needs for a prototype HTR dedicated to cogeneration of heat and electricity

The demonstrator reactor envisaged by NC2I platform would be dedicated to the production of process steam and electricity. For that, the helium maximum temperature targeted in such project would be limited to around 750°C. The core would burn uranium oxide or oxy-carbide at a nominal temperature and burn up already covered by recent irradiation testing. This given, a demonstrator could be built with a limited amount R&D in its narrower sense. As for any first-of-a-kind nuclear installation, significant large scale qualification testing will be required. Therefore in the following a distinction will be made between the R&D required to support the licensing of a steam generating demonstrator, related qualification efforts and R&D needs for further HTR development. These different items will be presented for each main component.

5.1 Fuel particles and fuel elements

R&D efforts for the demonstrator

The current modular HTR concept applicable to both pebble bed and block type fuel is based on the characteristics of the TRISO coated fuel particle developed in the 1980s. This type of fuel features very low releases of fission products due to manufacturing defects and operational irradiation up to about 1200 °C and a burn-up of about 80,000 MWd/t. During core heat-up accidents, this fuel shows excellent retention of fission products up to about 1600 °C. With these characteristics a demonstrator for production of process steam up to 550°C could be designed and licensed with a minimum R&D effort.

Nevertheless, it is to be noted that such statement shall be justified based on the precise characterization of the operating conditions for the demonstrator, taking into account the constraints which might be induced by the cogeneration.

As the computation of the fuel temperatures (and thus the safety margins) is affected by relatively large uncertainties, a short term solution could be the development of a core monitoring system able to give data to evaluate the fuel temperatures, at least for block type reactors. The development of such technology would probably require R&D.

Qualification efforts for the demonstrator

As described above, a fuel with characteristics matching the requirements for a steam generating demonstrator is already feasible. However, this fuel has been produced in the 1980s and to a much small scale in later times. Even if based on ancient specifications, any fuel produced for a new HTR shall undergo a qualification procedure. This qualification would cover the fuel operating conditions and include power ramps and heat-up experiments to simulate accidental conditions including accidental withdrawal of the most effective control rod. Qualification shall be supported by the use of numerical fuel performance codes, these codes and their input data will have to be thoroughly validated.

Moreover, specific R&D would probably go along with the qualification of the fuel and coatings quality control technologies (including non-destructive examination of the fuel). The objective is to link the physical parameters measures on the particle to the failure probability in core. This would give a more robust safety demonstration by reducing the uncertainties on fuel integrity due to the influence of empirical fabrication parameters. Recent results obtained by the INL program show promising developments in this field [ix].

R&D efforts for follow-up HTR

At least, fission product transfer coefficients through particles coatings should be more accurately determined, including for tritium. Studies usually show overestimation of the release, but these coefficients, whose values remain affected up to now by large uncertainties, are important to evaluate the primary circuit contamination and the potential source term in case of accident. As an example, the applicant shall provide an evaluation of the radioactive doses incurred by the operators during maintenance and control operations of the equipment of the primary circuit.

Ongoing R&D efforts appear reasonable to increase temperature up to which coated particles retain their inventory of fission products. Promising candidates for such efforts are an improvement of the microstructure of the SiC layer and a replacement of SiC by ZrC, both with the aim to achieve an effective fission product retention up to 1800 °C.

Improving the resistance of fuel elements against corrosive attack by air in design extension conditions would also be desirable. R&D efforts of the past regarding coating of entire fuel spheres or impregnating the surfaces of fuel elements, applicable to both spheres and block type fuel elements should be resumed.

5.2 Core design and safety performance

A large experience of the neutronic and thermo-hydraulic design of HTR cores has been acquired in the past which encompass the computation of main core parameters (reactivity coefficients, flux, burn-up) and the modeling of the normal and accidental transients. The existing computer codes and proposed developments have been reviewed in ARCHER [vii]: one of the conclusions of this review is a consistent set of validated tools is already available to design the core of an HTR.

R&D efforts for the demonstrator

For VHTR core, a specific requirement would be a more precise characterization of the safety margin related to the maximum temperature of the fuel during normal and accidental operation. For that, coupled thermos-hydraulic/neutronic codes are being developed.

Qualification effort for the demonstrator

For the licensing of any kind of reactor, efficiency of power control in normal operation and accidental situations has to be demonstrated. In this frame, the evaluation of core neutronic feedback coefficients is of primary importance and would require a dedicated experimental program.

The load following capability of the reactor would also have to be investigated, taking into account the requirements induced by the cogeneration option. The applicant shall demonstrate that load following is consistent with the reactor safety margins. For example, the German HTR-Modul was designed for an unrestricted load-following between 100% and 50% in order to limit the reactivity bound in the control rods. for this purpose and thus limit the reactivity which could theoretically be inserted in severe reactivity accidents (e. g. inadvertent withdrawal of the most effective control rod). For load following from 100 % to below 50%, the temporal build-up and decay of Xe would also impose restrictions in time. In this domain, the safety demonstration shall be completed at least buy the analysis of the results of the start-up tests.

R&D efforts for follow-up HTR

For a VHTR, an increased material data base may be needed, covering specific high temperature effects.

Further R&D might be carried out to optimize the core in the fields of feedback coefficients, burn-up homogenization, softening of power peak, use of neutron poisons.

5.3 Control and Shutdown Elements

R&D efforts for the demonstrator

The control rods for pebble bed cores are positioned in vertical channels in the side reflector close the surface of the core. A proven design which has been adopted for the HTR-Modul was the design of the THTR reflector rods. They consisted of sintered rings of B_4C stacked in annular claddings made of austenitic steel for high temperature application (creep rupture values for material 1.4981 available up to 750 °C). One control rod was formed by several annular sections loosely jointed together. Since these control rods can be moved unimpededly in their reflector channels, they are not subject to significant strains apart from their own weight.

During normal operation the control rods are actively cooled by a bypass to the core in order to keep the metallic claddings below 600 °C. In the passive decay heat removal mode, they are not cooled, and their temperature is defined by the transfer of the decay heat through the side reflector to the RPV. For a reactor of about 200 MWth peak temperatures of about 850 °C have to be expected with values above 800 °C for about 100 h. These temperatures still allow the use of high temperature resistant austenitic steel as described above. For pebble bed cores with higher thermal output, or for control rods located in active block type cores, higher temperatures during passive decay heat removal, may demand to switch to other cladding materials, for example Ni-based alloys. Only if this should not suffice, R&D work for ceramic composite cladding has to be considered.

A specific temperature issue may arise for control rods and their guide tubes in the upper plenum of block type reactor cores: during passive decay heat removal with the primary circuit under pressure, the developing core-internal natural circulation will fill the upper plenum with hot gas. This might overheat the control rod guide tubes which penetrate the plenum vertically.

The second shut-down system of modular HTR uses small absorber spheres (SAS) (10 mm in diameter) containing about 10 % vol of B_4C in a graphitic matrix. In the absence of steel, these spheres should be resistant to very high temperatures (melting point of B_4C: about 2600 °C, whereas eutectics are formed with steel already at 1480 °C), so no R&D should be needed for this type of neutron absorber even if it is inserted into the hottest parts of the core during passive decay heat removal. Such locations may be the inner parts of block type cores or the central columns of both block type and pebble bed cores with rated powers above 350 MWth.

Qualification efforts for the demonstrator

Due to the high safety importance that control and shutdown systems have for reactors in general, the systems for the demonstrator will have to undergo thorough qualification testing once their design is fixed in detail. For the PMBR, 1:1 scale test rigs have been erected for this purpose. This testing will also have to cover the reliability of control rod drives and SAS mechanics at operational and accidents temperatures. For SAS to be inserted into the hottest parts of the core under passive decay heat removal conditions, also the thermal stability will have to be demonstrated.

Moreover, a minimum requirement for the control rods is their adaptation to current domestic regulations for seismic design of a nuclear plant. This usually means that systems shall be tested and qualified on a vibrating table to be able to demonstrate their operability after an earthquake.

R&D efforts for follow-up HTR

Future development of control rods will have to allow higher operational and accident temperatures. Apart from high temperature resistant metallic alloys, claddings of ceramic composites may be applicable after successful R&D efforts. These must consider that the control rods see high neutron fluxes. In the past, carbon fiber – carbon composites showed rapidly decreasing stability under such conditions.

5.4 Ceramic Core Structures

R&D efforts for the demonstrator

The HTRs built in the 1970s used raw material (natural Gilsonite) for their graphite structures which are no longer available. Already in the 1970s and 80s the search for alternative materials (pitch coke, petrol coke) was started. Meanwhile a number of graphite grades on the basis of these raw materials have been developed with different characteristics for different purposes within the core assembly. Specimens of these grades have been irradiation tested and examined within the ARCHER project and its predecessors [i], [ii]. The achieved matrix of irradiation conditions (temperature, neutron dose) appears sufficient for the purposes of a demonstrator, so no basic R&D will be needed for this application.

Qualification efforts for the demonstrator

In a licensing process it will have to be demonstrated that the characteristics of the selected graphite grades can also be achieved in an industrial graphite production process.

Besides the material properties, the mechanical and thermal behaviobehaviorr of the ceramic core assembly will have to be assessed during component qualification. In the case of pebble bed cores, the minimization of unintended bypass flows may become an issue for qualification of appropriate sealing techniques, considering the fact that the graphite close to the core surface will shrink during operation resulting in larger gaps between the graphite blocks. Similar issues may arise for top reflector.

Depending on the set of safety cases retained in the safety demonstration, graphite structures shall be qualified in the field of corrosion resistance (air, water). Parameters piloting the graphite corrosion kinetics shall be documented on specimens sampled from the fabrication. It is to be noted that representative graphite specimens should be irradiated in the demonstrator for periodic controls and PIE.

As mentioned in the paragraph 1.4.1 of the appendix 1, cracks have appeared in the past in HTRs graphite structures (FSV, AVR). Together with a set of specific design rules for graphite, qualification of the core internal elements shall be completed by an in service inspection program, especially for the permanent elements (reflectors, supporting blocks, insulation blocks). In service inspection should also give indications on corrosion and wear of the graphite elements.

R&D efforts for follow-up HTR

For future VHTR applications the database concerning irradiation properties may need to be enhanced to higher irradiation temperatures. Graphite specialists should investigate whether a development of graphite grades with less property changes under irradiation and higher allowable neutron dose appears feasible.

With the graphite grades currently available, the side reflectors of 200 MWth class pebble bed cores will reach their design lifetime after about 30 full power years. Increasing the unit power will reduce the reflector lifetime. While this is no issue for the demonstrator, concepts to replace the side reflector or its inner part (or the central reflector of annular cores) may need to be developed for higher unit powers and/or longer intended reactor lifetimes.

As for the claddings of control rods, ceramic composite material has been proposed for different purposes in the core assembly, e.g. for tie rods for suspension of top reflector elements or for circumferential restraint of the core assembly. Since the behavior under irradiation for these elements should be subordinate due to their locations rather outside the reflectors, the use of ceramic composites for such components will require less R&D than for control rods and might reduce to a qualification issue.

Finally, trend in licensing urges the development of a graphite reprocessing/decontamination technology to reduce the amount of waste generated by the irradiated core structures if future series of HTR were envisaged.

5.5 Metallic Core Structures

The metallic core structures comprise the core support, on which the ceramic core assembly stands, the lateral core barrel and its upper lid (pebble bed cores) or hemispherical shroud (block type cores). Usually the flow of the primary coolant in a modular HTR for steam production provides direct cooling of the metallic core support structures and direct or indirect cooling of the lateral core barrel by the coolant riser channels in the outer side reflector. The upper enclosure of the core barrel is isolated from the primary circuit either by ceramic insulation on top of the top reflector (pebble bed cores) or by an insulation directly on the inner surface of the upper shroud (block type cores).

During passive decay heat removal the core support structure will no longer be actively cooled and is therefore protected by ceramic insulation on its upper side. The upper enclosure must be protected from overheating by the core-internal natural circulation if the reactor remains under pressure. From this case follow the thermal design criteria for the upper insulation mentioned above.

Thus by design efforts the metallic core structures can be effectively protected from the high temperature produced inside the core.

R&D efforts for the demonstrator

Considering the explanations provided above, no significant R&D efforts are foreseen for the metallic core structures of the demonstrator. The operational and accident temperatures can be kept at levels for which proven steel grades are available depending on the unit power. For reactors in the range of 200 MWth, low alloy ferritic steel may be used, whereas for higher unit powers, due to higher temperatures, austenitic grades are feasible.

In the passive decay heat removal mode, especially with depressurized reactor, the decay heat is transferred from the ceramic core assembly via the core barrel to the RPV mainly by thermal radiation. In order to limit the temperatures of the reflectors and the core barrel, a high thermal emissivity is therefore desirable (note that due to the transient storage and removal of heat in the core and the side reflector and the different points in time at which the maximum temperatures are reached in these components, the maximum fuel temperature is almost not influenced by the thermal emissivity of the core barrel). Currently there is only little knowledge of how to achieve a high emissivity on the surface and, as important, to retain it over the lifetime in a helium atmosphere with low impurities. This aspect may have to be investigated with some R&D efforts.

Qualification efforts for the demonstrator

While, as stated above, applicable steel grades are principally available, some qualification efforts enhancing their database may be required for their use under nuclear codes and standards.

Once a satisfying method has been developed to achieve and retain a high thermal emissivity of the surfaces, this method will have to be qualified for large scale application.

The non-destructive examination methods to be applied to the core metallic structures shall be qualified on representative mock-ups during the licensing phase of the demonstrator.

R&D efforts for follow-up HTR

For future VHTR applications it may be desirable or even demanding to construct further parts of the presently metallic core structure from ceramic composite material. Such applications may require additional R&D.

5.6 Hot Gas Duct

Modular HTR are usually connected to their power conversion units (steam generator, IHX) by coaxial ducts comprising a part of the primary pressure boundary (primary coolant pipe or, dependent on the design philosophy, cross duct vessel) and concentrically in it the hot gas duct, a tube with internal insulation. The hot gas produced in the core is transported to the steam generator or the IHX through inner tube, and the cold gas returning from the steam generator or the IHX is transported to the reactor again through the annulus formed by the hot gas duct and the primary pressure boundary. The cold gas is impelled by the primary blower; therefore its pressure is always higher than that of the hot gas inside the inner tube and any defect of this tube cannot lead to an impingement of the pressure boundary with hot gas.

R&D efforts for the demonstrator

A hot gas duct as sketched above has been developed in the 1980s in the German PNP project for hot gas temperatures of 950 °C [iii]. It has been tested with satisfactory results in a 1:1 scale test installation. Therefore no significant R&D is envisaged for the hot gas duct of a demonstrator with its lower hot gas temperatures.

Qualification efforts for the demonstrator

As with other important components of the demonstrator, a hot gas duct will have to be qualified by large scale testing. For the PBMR, such a test was planned, but not executed due to the stop of the project. Mock-ups were also planned for the Gas cooled Fast Reactor project.

R&D efforts for follow-up HTR

As stated above, a hot gas duct design has been developed and tested in the past for hot gas temperatures up to 950 °C. Therefore no basic R&D appears necessary for VHTR applications up this temperature.

5.7 Power Conversion Unit (Steam Generator, IHX)

The following statements will concentrate on steam generators as far as the demonstrator is concerned. He-He IHX will be addressed with respect to follow-up projects for VHTR applications.

R&D efforts for the demonstrator

As stated above, for the demonstrator the production of process steam up to a live steam temperature of 550 °C has been proposed. Since an intermediate steam cycle shall be used in order to minimize tritium permeation into the process steam, the first-to-secondary steam generator will have to produce steam at about 600 °C. This temperature is still close to the live steam temperatures of 530 °C to 567 °C that had been achieved in THTR and Fort St. Vrain and that the modular HTR were designed for. The material of the steam generator tubes, Alloy 800 for Fort St. Vrain and the HTR-Module, is suitable also for slightly increased temperatures, therefore no R&D will be required regarding steam generator tube material (THTR used austenitic X8 CrNiNb 16 13, 1.4961 for the final superheater).

In the gap analysis performed in ARCHER [iv], two other issues have been identified with regard operation and inspection of the steam generator which may require additional R&D or qualification efforts:

- Graphitic dust transported with the primary coolant is likely to accumulate on the surfaces of finned tubes. The effect of this deposition on the long term heat transfer needs further investigation.
- The helically wound steam generator tubes are much longer than tubes in conventional LWR. Existing LWR tube In-service Inspection (ISI) equipment does not provide sufficient length for inspection. In past German HTR R&D programs ISI was realized up to 90 m (2 x 45 m). A further increase of the range may be necessary to guarantee 100 % tube inspection.

Qualification efforts for the demonstrator

Some formal efforts may be necessary for nuclear qualification of Alloy 800 to live steam temperatures up to 600 °C.

Qualification of the detailed design solution in component tests will have to be performed.

In an application for co-generation, the thermos-mechanical loading of the steam generator tubes may be different from that for generation of electricity. The complete load collective has to be determined by analysis and the design has to be based on it.

Transfer coefficient for tritium shall be evaluated with the selected material for IHX and steam generators taking into account the oxidation layers which could form.[Study in ARCHER shows that a reduction of the transfer rate associated with clean tubes must be obtained for licensing]

R&D efforts for follow-up HTR

In the German PNP program in the 1970s and 1980s two types of tube He-He-IHX, one with helically wound tubes and one with U-shaped tubes in a compact arrangement, have been designed and 10 MW test components have been tested at 950 °C. For these types of IHX a basis is thus given for further development and implementation.

However, tube-type IHX would feature large dimensions, therefore the development of compact plate-type IHX was launched in the past. In the ARCHER project a test mock-up made of Alloy 800H with a thermal power of 91 kW was tested in air at 750-800°C [v]. According to the developers the compact stamped plate's mockup has been a technological challenge both to manufacture and to test to show its ability to withstand such working conditions. Nevertheless, this has been successfully achieved and it is considered that further analysis and investigation will be needed before the integrity and lifetime of the mockup and full size IHX can be fully evaluated.

For future steam generators for co-generation coatings of the tubes may be developed to reliably reduce tritium permeation.

5.8 Circulators

Helium circulators with a horizontal axis and a capacity sufficient to remove about 120 MWth from an HTR core have been developed and operated in the THTR and the Fort St.Vrain reactor. Current modular HTR designs feature circulators with a vertical axis on top of the steam generator pressure vessel and with capacities to remove 200 to 300 MWth. The Chinese HTR-PM currently under construction has a single loop with a single circulator to remove 250 MWth. From this it is concluded that helium circulators of this size can be considered as state of the art.

Helium circulators have not been touched in detail in the ARCHER project. In the gap analysis based on the technological standard of the THTR as the last HTR constructed and operated in Europe, magnetic bearings, especially their control system with regard to their availability and reliability of operation, have been identified as an item for further development. For the HTR-Modul, magnetic bearings have been indicated as an alternative to oil lubricated bearings already in the 1980s.

Nevertheless, reliability of the circulator appears to be a secondary concern for the licensing except for the three following points:

- contamination and subsequent doses delivered during maintenance and control,
- potential source of water/oil ingress (at least water would be necessary to cool motors),
- potential source of missile in case of turbine deblading.

These issues would be mastered by decontamination procedures and design of the circulator and surrounding structures.

5.9 Primary Pressure Boundary

All current modular HTR designs feature a steel primary pressure boundary comprising the RPV, one or more pressure vessels for the heat exchanging component(s) and the connecting cross duct vessel(s) (or primary coolant piping for complex arrangements such as the South African PMBR). In order to take maximum benefit from existing nuclear experience, the same low alloy steel grade shall usually be used as for commercial LWR. In this respect the irradiation damage by fast neutrons to the cylindrical part of the RPV is of particular interest. From LWR design and operation a vast database exists for irradiation at about 300 °C. Irradiation damage at a given neutron fluence is more pronounced at lower temperatures. The South African PBMR was therefore designed with a dedicated operational conditioning loop which held the cylindrical RPV wall at about 300 °C. For the German HTR-Modul, a different approach was followed: the outer layer of the side reflector was built of carbon bricks which containing boron in order to keep the fast neutron fluence at the RPV below a value at which the irradiation damage would become significant. With the carbon layer that also acts as a thermal insulation, the operational RPV temperature at the axial height of the core was calculated to about 180 °C.

In order to eliminate the potential of large air ingress, the primary pressure boundary of modular HTR should be designed and manufactured according to principles of Basis Safety (German) or incredibility of failure (British, French regulations implicitly follow a similar approach). In the ARCHER project, one work package was dedicated to identify potential gaps to be filled for application of such principles to HTR primary pressure boundaries [vi].

R&D efforts for the demonstrator

As long as the unit power of the demonstrator is small enough to keep the RPV temperatures in range of LWR applications, proven LWR RPV steel can be used and little additional R&D is required: in order to satisfy Basis Safety or Incredibility of Failure requirement, the database regarding corrosive effects of helium with low impurities should be broadened. Basis Safety or Incredibility of Failure concepts should be extended to the thermo-sleeve joint of the live steam line to the steam generator pressure vessel, because in this joint materials are used which are not used in LWR.

In order to limit the temperatures of the core barrel and the lateral restraints of the ceramic core assembly during passive decay heat removal, also the inner surface of the RPV should feature a high and stable thermal emissivity. It seems that some R&D is still necessary to confirm the stability of a high emissivity in the given helium atmosphere and neutron dose. For the ferritic steel used for LWR it is no problem to attain high emissivity from oxidation in air on the outer surface.

Qualification efforts for the demonstrator

A demonstrator in the power range of 200 to 250 MW will have RPV operating temperatures at or below the cold gas temperature of typically 250 °C. Under passive decay heat removal conditions, the maximum RPV temperature can be kept below 370 °C. Thus the temperature range for which LWR steel is qualified according to European standards will not be exceeded. Nevertheless, steel grade envisaged for an HTR demonstrator shall be qualified in helium environment (with potential impurities) and the applicant shall ensure that the irradiation dose would fall in the already validated domain.

Demonstrators with higher unit power may show the same operational temperatures, but higher maximum temperatures will occur during passive decay heat removal which may require additional qualification efforts. Section III of the ASME Code allows for SA-508 Class 3 material cumulative 3000 hrs at 427 °C in service Level B events and 1000 hrs (maximum of three events) at 538 °C in service Level C and D events. When assigning Service Level D to passive decay heat removal events it should be considered, however, that Level D Service Limits „permit gross general deformations with some consequent loss of dimensional stability and damage requiring repair, that may require removal of the component ... from service“ [ASME Section III, NCA-2142.4 (b) (4)]. For modular HTR with unit powers above 300 MWth, specific attention should be paid to the passive heat removal mode with the reactor under full pressure. The PMBR 400 showed high RPV temperatures in that case which would make this mode a Level D event for the RPV. Therefore modular HTR with higher unit powers may require other vessel material with higher allowable temperature and in the consequence more R&D and qualification efforts. On this point, it is to be noted that a low chromium steel (2 ¼ Cr 1 Mo) has already been qualified in Japan for the vessel of the HTTR, with a design temperature of 440°C.

R&D efforts for follow-up HTR

For components of the primary pressure boundary operating at higher temperatures where time dependent effects must be taken into account, the LWR Integrity Concept (IC: basis safety, incredibility of failure) must be extended to assure that the causes of possible operational damage mechanisms (creep and/or creep-fatigue processes) do not impact the component integrity. Due to the complexity of ongoing and competing damage mechanisms advanced simulation methods and appropriate material data must be used to assure the safe operation. The application of the IC in this range must be based on a reliable and sufficient materials data base.

5.10 Reactor Cavity Cooling System:

R&D efforts for the demonstrator

System designs with forced convection and with natural convection including evaporation cooling have been developed for different modular HTR concepts. All systems operate at low temperatures and pressures and pose no technological challenge. However, issues to be clarified remain. One issue comes from the fact that the maximum heat input into the cavity cooler is at a rather low position of the tall system. Steam may be generated there and be condensed again on its way upward, especially in systems working with natural circulation. It should be demonstrated that local steam production low in the system does not lead to unstable flow conditions. This issue may as well be a qualification issue.

The second issue, which may also be a matter of qualification rather than R&D, is that especially systems with forced convection must be tolerant against temporary loss of flow. A quenching of the hot, dry cavity cooler must be possible after limited periods of time.

Qualification efforts for the demonstrator

The performance of the cavity cooler will have to be demonstrated in sectional large scale test installations. It must be considered that also the bearings of the vessel support structures need to be cooled, which due to complicated geometries in these regions might be difficult with natural convection systems.

R&D efforts for follow-up HTR

Also follow-up HTR may be equipped with one of the existing design proposals for cavity cooling once the issues described above are solved. Therefore there are currently no R&D efforts envisaged for follow-up HTR.

5.11 Dust issue, filtered initial release:

R&D efforts for the demonstrator

Dust is produced especially in pebble bed HTR by the fuel spheres moving in the core and in the fuel handling system, corrosion of graphite structures contributes to the amount of dust, too. This dust is contaminated with fission products and activation products and, if accidentally released to the atmosphere, can increase or even dominated the dose to the public. In block type reactor cores less dust is produced because the fuel elements are stationary during operation, but it is not zero because the columns of fuel blocks rub against each other due to flow induced vibrations, corrosive effects and during core reshuffling operations.

Depending on national regulations and break sizes to be assumed, the initial release of dust loaded primary coolant in consequence of leaks/breaks of the primary pressure boundary will have to be filtered. Filters with sufficient retention capability during a blowout have to be developed. Allowable pressure differences may be limited by the allowable overpressure of the reactor building.

It is assumed that radioactivity released in case of breach occurring on the primary circuit is mainly associated with dust. Consequently, resuspension models are needed to evaluate the quantity of dust which could be released from the primary circuit. In ARCHER [viii], as an example, it was stated that the parameters characterizing the adhesion and thus the resuspension of dust could impact the source term by more than an order of magnitude. But validated parameters are not available today. Secondly, it is essential to determine how much radioactivity is bound to the dust. ARCHER concluded [viii] that large uncertainties must be allocated to the available correlations. These two phenomena would probably need R&D work to be able to present a robust safety demonstration. At least, for the demonstrator, experiments on dust trapped in the primary circuit filters could support a posteriori the assumptions made for the safety demonstration. Other phenomena, like dust deposition in containment, might be of less importance and thus potentially overcome by conservative assumptions.

Qualification efforts for the demonstrator

Mitigating measures in case of breach in the primary circuit will have to be qualified. As an example, the retention efficiency of the filters for accumulated dust will have to be demonstrated in mock-ups under realistic accident conditions.

Codes used to determine the dust activity and potential release would have to be qualified taking into account the design of the circuits, containment.

R&D efforts for follow-up HTR

For follow-up HTR, a strategy to cope with the dust issue may point into two directions:

- Design and manufacture the primary pressure boundary such that leak or break sizes can be excluded through which significant amounts of contaminated/activated dust could escape into the reactor building. Improve of computational capabilities to demonstrate the retention of dust in the reactor building. A first step in the latter direction has been taken in the ARCHER project by application of the COCOSYS containment code to depressurization accidents of a modular HTR.
- Develop concepts to reduce the production or deposition of dust in the primary circuit. To achieve this, the following proposals have been identified in the gap analysis [iv] performed in the ARCHER project: impregnation methods could be developed to reduce the abrasion of fuel spheres, dust filters in the primary circuit could be developed, flushing concepts could be developed to remobilize and discharge loosely deposited dust.

5.12 External hazards

Given the coupling features envisaged for a demonstrator with cogeneration (steam production), the safety demonstration of the demonstrator should address the issue of external hazards induced by the industrial environment of the plant. The nature of hazards potentially induced by industrial facilities has already been documented. Scenarios and load cases are available in the severe accident studies of the industrial sites (explosions, toxic releases, flooding). Moreover, this issue is also being examined for other nuclear plant concepts and the HTR will benefit from the results of these studies. So, it can be stated that no particular R&D would be needed for the demonstrator, given that he would be located at a reasonable safety distance of the end user’s industrial site.

6 Conclusion

Taking stock of the feedback of HTR operation, incidents and design evolutions allows to establishing a quite comprehensive list of items to be dealt with in relation with the support of the safety demonstration of a future HTR demonstrator. Recent work performed within ARCHER and past European projects reveals that only a few domains shall require significant R&D effort, assuming that no substantial evolution would be foreseen in the demonstrator design. In practice, R&D effort will be split between mastering of the issues that would arise along qualification of the reactor components and safety systems and the remaining questions like:

- the evaluation of the fission product transfer coefficient in the fuel coatings and graphite matrix,
- the means to evaluate the fuel temperature in standard and accidental operation,
- the achievement of a suitable radiative emissivity of the core barrel,
- In service inspection of the primary structures including graphite structures, fuel elements (blocks) and steam generator tubes,
- the evaluation of dust behavior, distribution in the primary pipes and components (potential for accumulation, plate-out, etc.), resuspension and dust bound fission products phenomena and, in general, the development of a complete chain of computer codes for the modelling of source term in case of depressurization scenarios.

Additional R&D shall obviously come along with the qualification of the materials and main components of the demonstrator like the circulator, steam generator, control rods, structural graphite or hot ducts. As an example, the qualification of the cavity cooling system shall validate the main design parameters governing heat exchange efficiency. This qualification would also be a tool to further improve the safety margins provided by the passive decay heat removal feature of the HTR. Licensing is backed mainly by the evaluation of normal and abnormal transients relying on tests and computer codes. A rather complete set of codes is already available, but shall certainly require limited R&D effort to be validated (except for source term evaluation). Finally, development of specific design rules shall also require R&D efforts (extension of ASME or RCC-M applicability).

Regarding the licensing of a demonstrator, it is to be noted that remaining uncertainties, like those affecting the assessment of the accidental radioactive source term, could be outweighed by a conservative approach and, if needed, by specific mitigation measures (confinement, filters, monitoring systems, etc.).

Finally, regarding the optimization and further developments of the HTR concept (VHTR), the R&D for support of the safety demonstration are assumed to be needed essentially in the field of development of high temperature resistant fuel and material together with an effort to enhance of the fuel quality control and reduce uncertainties on safety parameters (fuel maximum temperature and burn-up).

References

i. ARCHER D41.12: High dose experiments at 750°C and 950°C – Full PIE of INNGRAPH-1B and INNOGRAPH-2B – M.C.R. Heijna – 3/04/2012

ii. ARCHER D42.14: Microstructure analysis of irradiated graphite – M.C.R. Heijna – 27/11/2012

iii. IAEA Specialists' meeting on heat exchanging components of gas cooled reactors, Düsseldorf, FRG., 16-19 April 1984 – Design requirements on HTR main components for process heat application – K. Dumm

iv. ARCHER D12.11: Gap analysis – B. Wiechers, M. Esch, F. Sassen – 07/11/2014

v. ARCHER D43.41: Test and expertise report, including results of CFD post-test calculations – F. Pra – 28/01/2015

vi. ARCHER D25.21: Gap analysis – Integrity concept LWR-HTR – M. Hoffmann, K.H. Herter, A. Klenk, N. Kohtz – 16/10/2013

vii. ARCHER D13.11: Code inventory in Europe – J. Lapins, J. Bader, M. Buck – 24/08/2012

viii. ARCHER: Application of the new CCSYS model for recent HTR confinement strategies – R. Gehr, S. Jühe – 07/11/2014

ix. Status of Very High Temperature reactor R&D – D. Petti – NC2I-R final conference 14,15 September 2015 - http://www.snetp.eu/nc2i-conference-2015/

APPENDIX I: Synthesis of the R&D needs derived from the operational feedback

1 Nuclear heat supply system

1.1 Reactor Pressure Vessel

1.1.1 Insulation and Moisture Issues (Fort St. Vrain)

Water at the FSV reactor was inadvertently injected into the primary system through the circulator bearings. The moisture tended to be absorbed into the Prestressed Concrete Reactor Vessel (PCRV) insulation (Kaowool: a ceramic fiber blanket material) adjoining the circulator. Desorption of water from the insulation and into the primary coolant helium was usually the longest activity during moisture clean up [7]. The Kaowool also out-gassed moisture that was entrained in the insulation during shutdown. Water trapped in the PCRV liner is another potential source of moisture ingress into the helium during startup because of small cracks in the weld; they would seal up because of thermal expansion when exposed to high power [7].

NGNP Implementation

Current NGNP Project design concepts for circulators do not use water-cooled bearings. The NGNP design concept (GA Vessel System SDD3) has a drain installed at the base of the RPV to facilitate water removal, if necessary.

No R&D activities specifically related to circulator bearing integrity are actually conducted.

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

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 - 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.

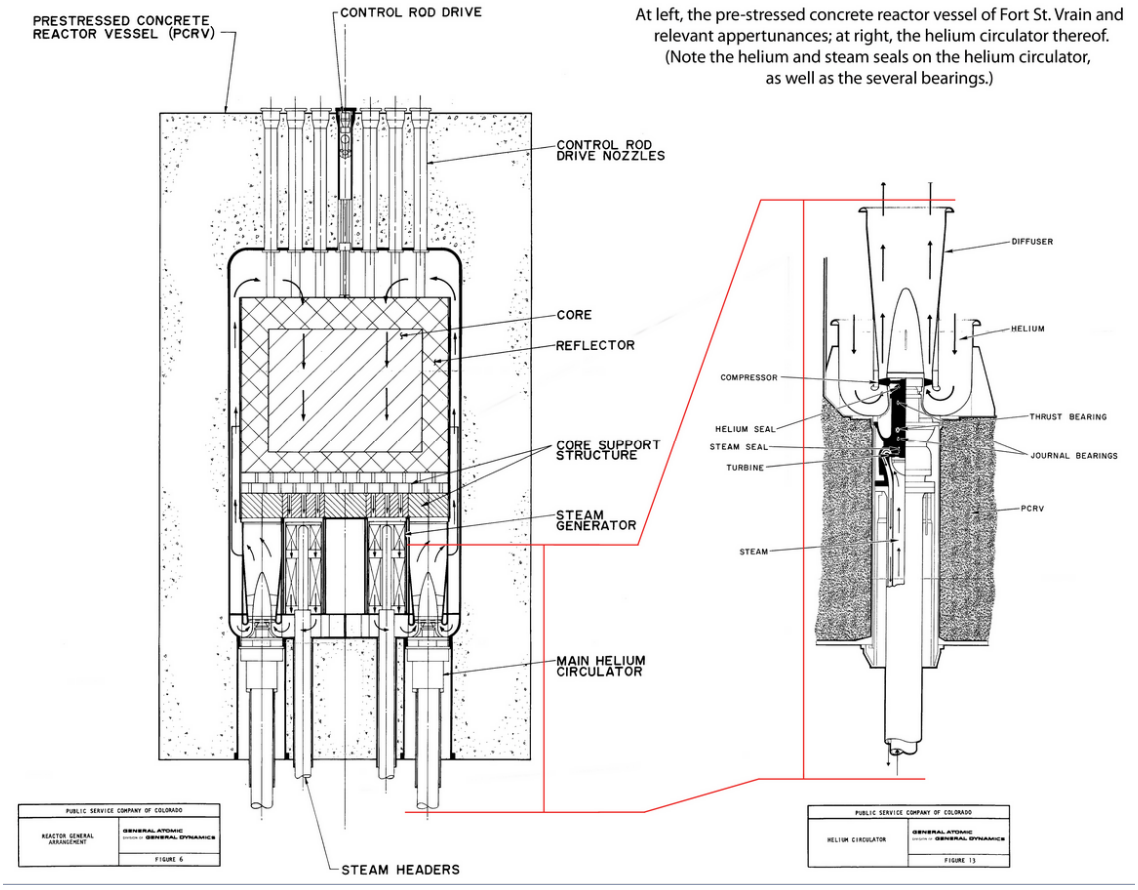

Figure nn. Fort Saint Vrain cross section

NGNP Implementation

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

No R&D activities, related to RPV sealing and flanges, are actually needed.

1.1.3 Keeping the RPV from Overheating (Dragon)

It was assumed in the Dragon reactor design 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.

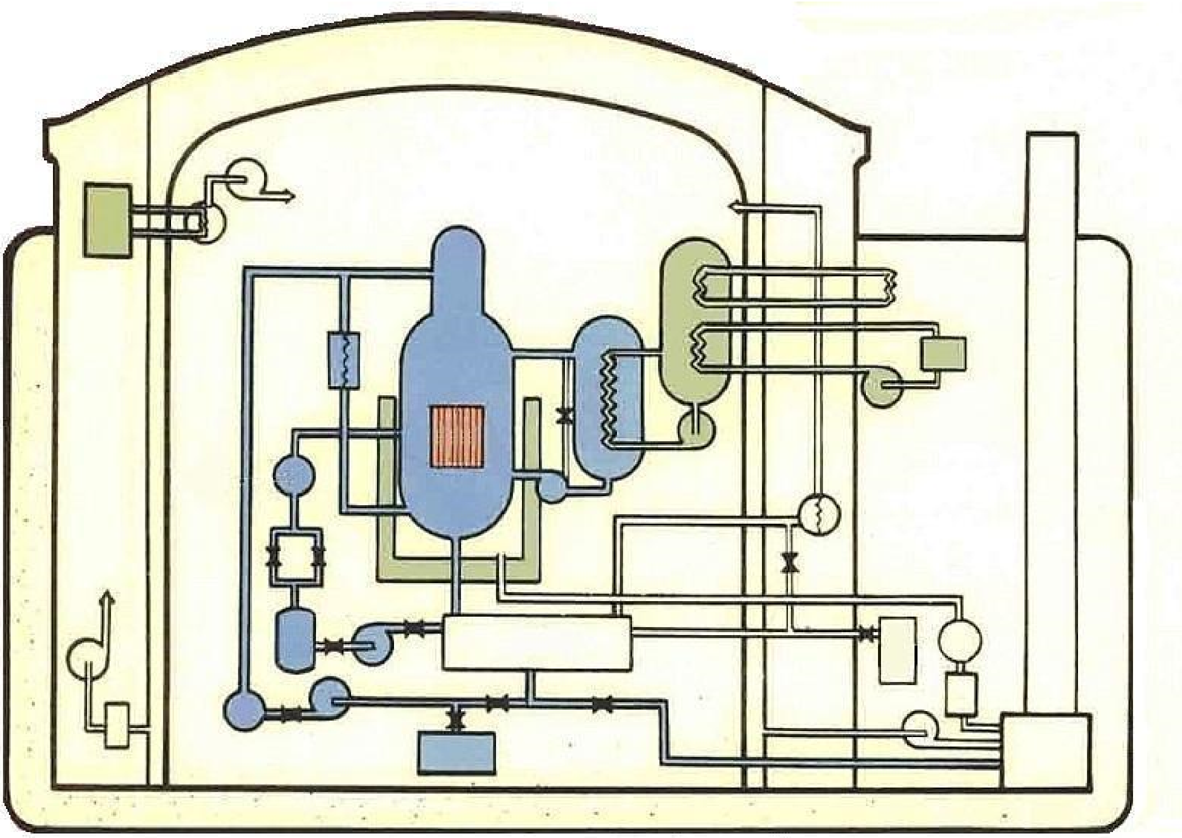

Figure 3. Dragon reactor diagram.

NGNP Implementation

This strategy has already been incorporated into the HTGR design concepts under consideration for the NGNP. Consequently, in the NGNP design concepts the RPV is generally exposed to primary helium at the reactor core inlet temperature (250 to 288°C).

No R&D activities, related to RPV cooling, are actually needed.

1.2 Reactor Vessel Internals

1.2.1 Temperature Rise in Core Support Plate (HTTR)

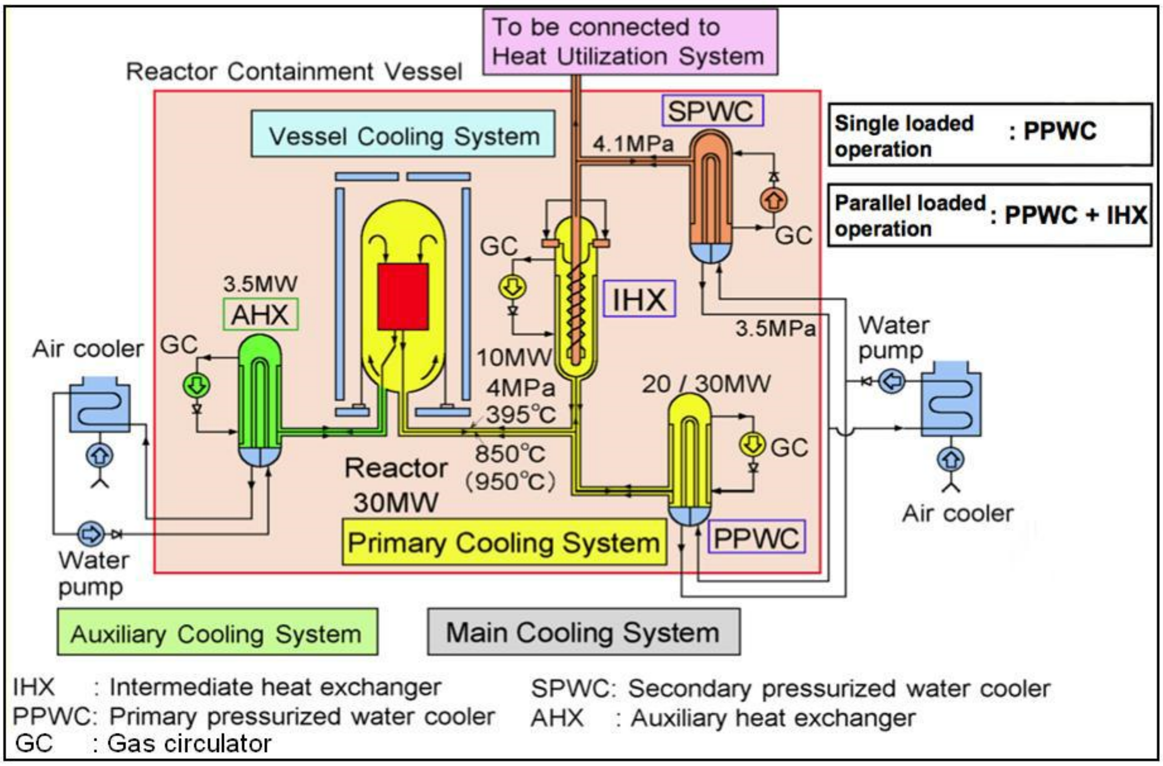

Figure 4. HTTR Reactor

During a power rise test, the core support plate of the Japanese, prismatic block, High Temperature Engineering Test Reactor (HTTR) showed an unexpected temperature rise. The rise in temperature of the core support plate was attributed to bypass flow. To reduce the helium leakage flow in the spaces between the 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 specific reactor core design.

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.

1.2.2 Flow-Induced Vibrations (AGR)

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.

NGNP Implementation

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.

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] postulates that methods should be developed to detect and evaluate vibration that could damage NGNP components. 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.

1.3 Reactor Core and Core Structures

1.3.1 Moisture Ingres Issues (FSV)

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.

NGNP Implementation

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.

Research and Development

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

1.3.2 Helium Pressurization Line (FSV) [13]

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.

NGNP Implementation

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.

Research and Development

1.3.3 Core Temperature Fluctuations (FSV)

In November 1977, FSV experienced some dynamics of the power levels and small fluctuations of temperature [18]. 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.

NGNP Implementation

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.

1.3.4 Inner Reflector (AVR)

Inner reflector inspection was difficult due to unusual design [20]. 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.

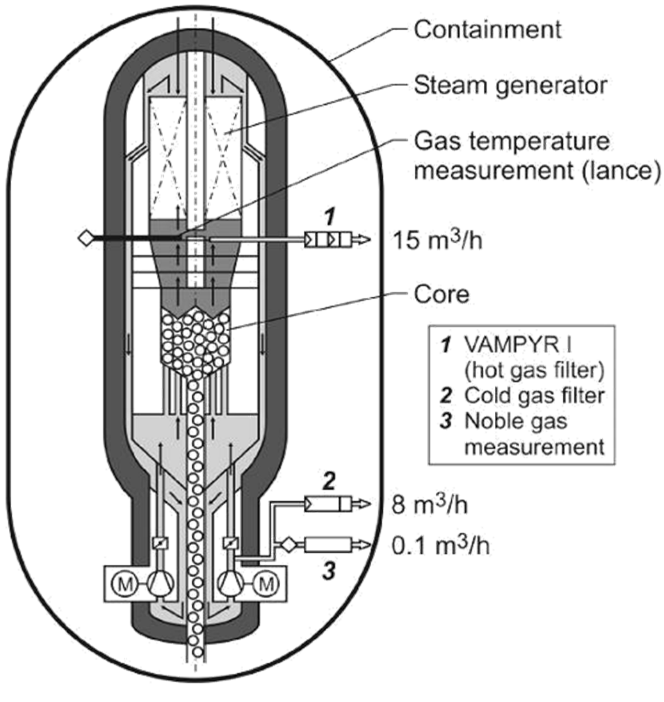

Figure 5. AVR basic diagram.

NGNP Implementation

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

Research and Development

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

1.3.5 Graphite and Graphite Wear (HTR-10) [21]

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.

NGNP Implementation

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).

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.

1.3.6 Outer Reflectors Bowing (Peach Bottom Unit 1) [24]

The Peach Bottom reflector element B16-01 was made by the National Carbon Company's (Union Carbide Corporation, UCC) from 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 dose at high temperatures.

NGNP Implementation

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.

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 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.

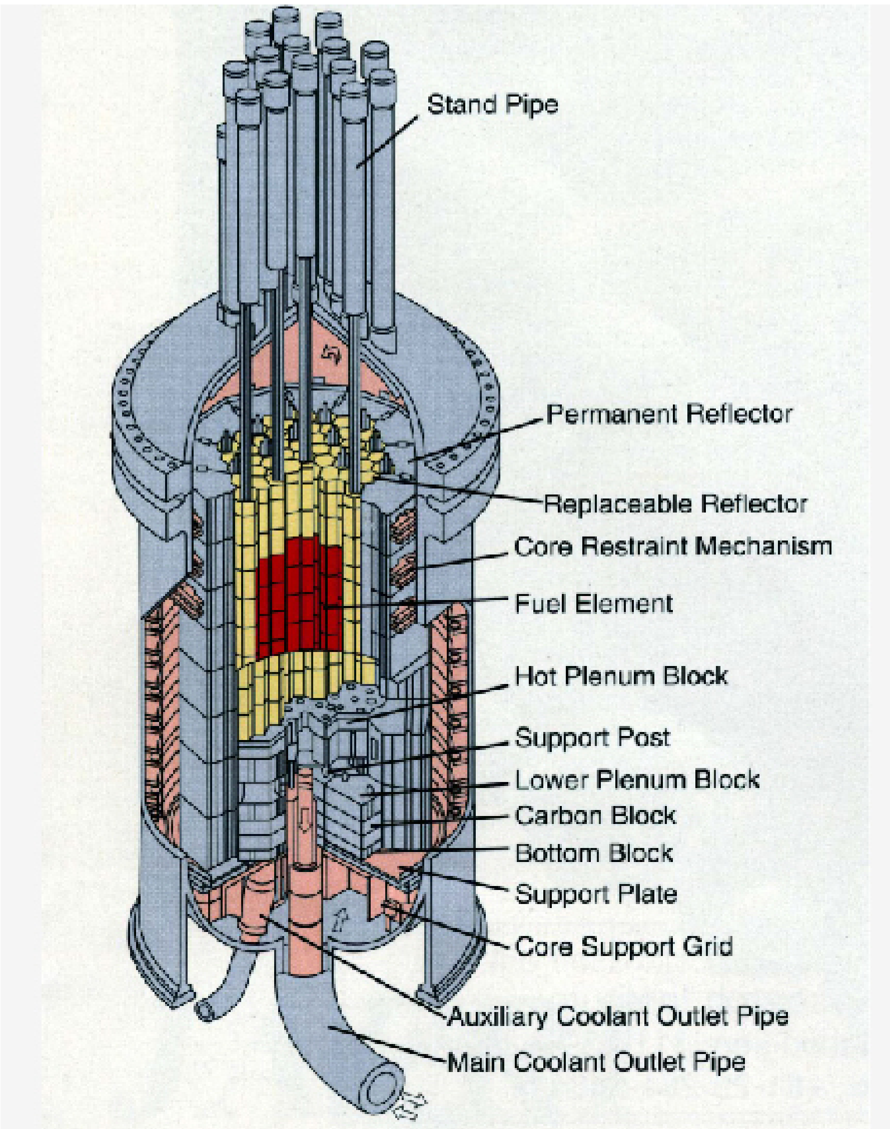

Figure 6. Peach Bottom Unit 1.

1.4 Fuel Elements

1.4.1 Fuel Cracks Caused by Tensile Stress (FSV)

Sometime before October 1981, the FSV licensee discovered that a crack had propagated through two stacked fuel elements [26]. 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.

NGNP Implementation

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.

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.

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.

1.4.2 Prismatic Fuel Performance (FSV)

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 [30].

NGNP Implementation

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.

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.

1.4.3 Pebble Bed Fuel Performance (AVR) [31, 32]]

In September 1988, the Jülich Institut 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.

NGNP Implementation

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 UO2 particle kernels).

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.

1.4.3.1 Immediate Post-AVR Perspective [35]

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.

NGNP Implementation

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.

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.

1.4.3.2 Direct Operational AVR Experience [39]

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.

NGNP Implementation

NGNP would benefit by considering AVR issues such as the pebble elements becoming too tightly packed for initial operations. 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.

1.4.4 Fuel Summary (HTR-10) [40]

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.

NGNP Implementation

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 topics are related to fuel failure experiments and modeling:

- Fuel—Heat-Up Experiment Modeling of Irradiated Fuel Particles in ATLAS.
- Core Element Failure Mode Data.

Research and Development

The INL AGR fuel qualification program has a number of irradiation experiments [41] identified in its program. One of the irradiation experiments is 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.

1.4.5 Pebble Bed Graphite Dust (AVR)

Several sources of graphite dust have been identified at AVR: notching, spalling, pitting, fuel-sphere fracture, and peeling of the fuel spheres [42]. The dust from these sources has contributed to activity concentration in the primary coolant.

NGNP Implementation

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.

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.

1.4.6 Prismatic Graphite Dust (HTTR) [46]

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. Two phenomena are identified in the report [47] from this meeting: 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.

NGNP Implementation

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.

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].

1.4.7 Fission Product Release Monitoring (HTTR)

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 [52]. 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.

NGNP Implementation

There are a number issues related to fission product release:

- 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
- Fission product speciation during mass transfer
- Fission gas release from core materials
- 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.

1.4.8 Fission Product Trapping (Peach Bottom Unit 1) [53]

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.

NGNP Implementation

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 R&D activities are currently identified related to trapping fission products.

1.4.9 Amoeba Effect (Peach Bottom Unit 1)

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 [56]. 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.

NGNP Implementation

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.

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]

1.5 Reactivity Control Systems

1.5.1 Moisture Effects on Reactivity Control System (FSV) [59]

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.

NGNP Implementation

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].

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."

1.5.2 Control Rod Cable Failure (FSV) [62]

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.

NGNP Implementation

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.

Research and Development

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."

1.5.3 Control Rod Temperature Anomalies (HTTR) [65]

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.

NGNP Implementation

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.

1.5.4 Control Rods in Side Reflectors and Lubricants in a Helium Environment (HTR-10) [67]

Lubrication for the control rods at HTR-10 was a concern because of the requirement to maintain 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.

NGNP Implementation

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

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

1.5.5 Control Rods Placements (Peach Bottom Unit 1) [68]

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.

NGNP Implementation

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).

1.5.6 Control Rod Lubrication (Peach Bottom Unit 1) [69]

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.

NGNP Implementation

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

1.5.7 Oil Leaks (Peach Bottom Unit 1)

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.

NGNP Implementation

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.

1.5.8 Fatigue in Control Rods (Peach Bottom Unit 1) [71]

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.

NGNP Implementation

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.

1.6 Reactivity Cavity Cooling System

1.6.1 Reactivity Cavity Cooling System Leakage [72]

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.

NGNP Implementation

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 issues identified, related to the characteristics of an RCCS in operation:

- 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.
- 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.
- RCCS Optional Large Scale Test—Data Range/Service Conditions: Under irradiation, not under irradiation, under normal operating conditions, under accident conditions.
- RCCS Characteristic Effects of Particulate and Plate-out on Radiation Heat Transfer.
- Wind Tunnel Test of RCCS I/O Structure.
- 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].

1.7 Reserve Shutdown System

1.7.1 Inadvertent Actuation of Reserve Shutdown System (FSV) [74]

At FSV the reserve shutdown system (RSS) was inadvertently actuated, which caused injection of 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.

NGNP Implementation

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.

1.7.2 RSS Failure to Deploy Boronated Graphite Balls (FSV) [75]

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.

NGNP Implementation

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.

2 HEAT TRANSPORT SYSTEM

2.1 Circulators

2.1.1 Circulator Seals and Stress Corrosion Cracking (FSV) [78]

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.

NGNP Implementation

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.

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].

2.1.2 Use of Active Magnetic Bearings (HTR-10) [81]

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.

NGNP Implementation

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.

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.

2.1.3 Oil Ingress in Compressor (Peach Bottom Unit 1) [83]

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.

NGNP Implementation

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.

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.

2.1.4 Friction Damage (Dragon) [85]

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.

NGNP Implementation

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

2.2 Intermediate Heat Transfer

2.2.1 Intermediate Heat Exchanger Materials (HTR) [86]

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.

NGNP Implementation

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.”

2.2.2 Successful Operation at High Temperatures (HTTR) [88]

HTTR was able to successfully use and test a 10 MWth 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.

NGNP Implementation

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.

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).

2.2.3 Helium Leakage in Secondary Loop (HTTR) [90]

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.

NGNP implementation

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.

2.2.4 Steam generator integration design (HTR-10) [92]

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.

NGNP Implementation

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."

2.3 Hot duct and cross vessel

2.3.1 Hot duct materials (THTR) [93]

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.

NGNP Implementation

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.

2.3.2 Successful use of high temperature hot duct (HTTR) [95]

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 SiO2 and Al2O3.

NGNP Implementation

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.

2.3.3 Loadings on cross vessel (HTR-10) [96]

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.

NGNP Implementation

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.

2.4 High temperature valves

2.4.1 Mitigating air ingress through valves (HTR-10) [97]

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.

NGNP Implementation

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, "If the high temperature isolation valves (HTIVs) in the secondary system were 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].

3 Power conversion system – Steam generator

3.1 Cracks/Leakage in Steam Generator (FSV) [100]

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.

NGNP Implementation

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.

Research and Development

Weld and component requirements are discussed in the NGNP Steam Generator and IHX Materials R&D Plan101. 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 Events102. 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."

3.2 Bolt Shearing (FSV) [77]

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.

NGNP Implementation

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.

3.3 Steam Generator Leak (AVR) [103]

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.

NGNP Implementation

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.

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.

3.4 Hot Steam Headers for the Steam Generators (THTR) [106]

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.

NGNP Implementation

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.

3.5 Fatigue Analysis (HTR-10) [107]

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.

NGNP Implementation

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.

3.6 Materials Used and Migration of Tritium (Peach Bottom Unit 1) [108]

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.

NGNP Implementation

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.

3.6.1 Dynamic Stresses and Side Flow Maldistribution (AGR) [110]

Cross reference to Section 4.1.6 of HTGR Lessons Learned NGNP.

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.

NGNP Implementation

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.

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.

4 BALANCE OF PLANT (BOP)

4.1 Fuel Handling System - Design Issues (FSV) [112]

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.

NGNP Implementation

Fuel handling systems for the NGNP will be designed with regulatory agencies having oversight of those systems.

Research and Development

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

4.2 Instrumentation and Control

4.2.1 Instrumentation Failure (FSV) [113]

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.

NGNP Implementation

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.

Research and Development

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 Cool-down Temperature Monitoring Instrumentation.”
- GA DDN C34.01.01. “Verify Core Inlet and Outlet Helium Temperature Measurement Instrumentation.”
- GA DDN C.34.01.02. “Verify Plate-out 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.”

4.2.2 Core Temperature Instrumentation (AVR) [115]

A 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.

NGNP Implementation

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.

Research and Development

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."

5 Auxiliary Systems

5.1 Experience at FSV [120]

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.

NGNP Implementation

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.

5.2 Helium Purification System Issues (FSV) [116]

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.

NGNP Implementation

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 an issue to be addressed in the design phase.

5.2.1 Chemical Cleanup (Peach Bottom Unit 1) [118]

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. 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.

NGNP Implementation

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.

5.2.2 Helium Purification Piping (Dragon) [119]

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.

NGNP Implementation

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).

6 Others

6.1 Electrical Arcing (FSV) [121]

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

NGNP Implementation

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.

6.2 High Winds (FSV) [123]

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.

NGNP Implementation

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.

7 SUMMARY OF HTR R&D NEEDS

The issues discussed in the original papers (INL/EXT-11-21545, INL/EXT-10-19329) were organized according to their relevance to the NGNP project, see Table 1. Some were 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). Out of 59 issues listed, 24 were considered as not requiring further R&D activities at the moment.

Several issues will need to be considered in future Project R&D and/or design activities, namely:

- 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 Plan126, 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.

Two issues are of primary importance to safety of HTR:

- Collection of graphite dust, which may lead to explosions,
- tritium migration to the environment.

In this paper the graphite dust issue was covered in: 2.3.5, 2.4.5 and 2.4.6. Tritium migration was covered in 4.1.6 and 5.3.2.

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.

Table 1. Summary of HTR R&D Needs

		Issue	R&D Needs
1.	2.1.1	Insulation and Moisture Issues	No
2.	2.1.2	Sealing and Flanges of the RPV	No
3.	2.1.3	Keeping the RPV from Overheating	No
4.	2.2.1	Temperature Rise in Core Support Plate	Yes – Important issue
5.	2.2.2	Flow-Induced Vibrations	Yes – Important issue
6.	2.3.1	Moisture Ingres Issues	No
7.	2.3.2	Helium Pressurization Line	Yes – Research needed
8.	2.3.3	Core Temperature Fluctuations	Yes – High priority
9.	2.3.4	Inner Reflector	No
10.	2.3.5	Graphite and Graphite Wear	Yes - Necessary
11.	2.3.6	Outer Reflectors Bowing	Yes

12.	2.4.1	Fuel Cracks Caused by Tensile Stress	Yes – Experiments & Numeric models
13.	2.4.2	Prismatic Fuel Performance	Yes – Using past experience
14.	2.4.3	Pebble Bed Fuel Performance	Yes – Using past experience
15.	2.4.4	Fuel Summary	Yes – HTR 10
16.	2.4.5	Pebble Bed Graphite Dust	Yes
17.	2.4.6	Prismatic Graphite Dust	Yes
18.	2.4.7	Fission Product Release Monitoring	Yes - Key area
19.	2.4.8	Fission Product Trapping	No
20.	2.4.9	Amoeba Effect	Yes – Key area
21.	2.5.1	Moisture Effects on Reactivity Contr. Syst.	Yes
22.	2.5.2	Control Rod Cable Failure	Yes
23.	2.5.3	Control Rod Temperature Anomalies	Yes
24.	2.5.4	Control Rods in Side Reflectors/Lubricants	No
25.	2.5.5	Control Rods Placements	No
26.	2.5.6	Control Rod Lubrication	No – Design issue
27.	2.5.7	Oil Leaks	No
28.	2.5.8	Fatigue in Control Rods	No
29.	2.6.1	Reactivity Cavity Cooling System Leakage	Yes
30.	2.7.1	Inadvertent Actuation of Reserve Shut. Syst.	No
31.	2.7.2	RSS Failure to Deploy Boron. Graph. Balls	No
32.	3.1.1	Circulator Seals and Stress Corrosion Crack.	Yes - Important
33.	3.1.2	Use of Active Magnetic Bearings	Yes – Design issue
34.	3.1.3	Oil Ingress in Compressor	Yes - Important
35.	3.1.4	Friction Damage	No
36.	3.2.1	Intermediate Heat Exchanger Materials	Yes - Extensive
37.	3.2.2	Successful Operation at High Temperatures	Yes - material issue
38.	3.2.3	Helium Leakage in Secondary Loop	Yes
39.	3.2.4	Steam Generator Integration Design	Yes – Design issue
40.	3.3.1	Hot Duct Materials	Yes – Material issue
41.	3.3.2	Successful use of High Temp. Hot Duct	No
42.	3.3.3	Loadings on Cross Vessel	No – Design issue
43.	3.4.1	Mitigating Air ingress Through Valves	Yes - Important
44.	4.1.1	Cracks/Leakage in Steam Generator	Yes - Important
45.	4.1.2	Bolt Shearing	No

46.	4.1.3	Steam Generator Leak	Yes – Material issue
47.	4.1.4	Hot Steam Headers for Steam Generators	No
48.	4.1.5	Fatigue Analysis	No – Operational issue
49.	4.1.6	Materials Used and Migration of Tritium	Yes – Materials issue
50.	4.1.7	Dynamic Stresses and Side Flow Maldistrib.	Yes
51.	5.1.1	Design Issues	Yes
52.	5.2.1	Instrumentation Failure	Yes
53.	5.2.2	Core Temperature Instrumentation	Yes
54.	5.3.1	Helium Purification System Issues	No
55.	5.3.2	Chemical Cleanup	No
56.	5.3.3	Helium Purification Piping	No
57.	5.3.4	Auxiliary Systems Failures	No – Design issue
58.	5.3.5	Electrical Arcing	Yes – Design issue
59.	5.3.6	High Winds	No – Design issue

References of appendix 1

1. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011.
2. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 2.1.1, p16.
3. NGNP-S00222, “GA's Vessel System SDD”, Section 2.1.1.3.
4. INL/EXT-11-21397, “Assessment of NGNP Moisture Ingress Events”, Rev. 0, Idaho National Laboratory , April 2011.
5. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 2.1.2, p16.
6. NGNP-T00001, “General Atomics, Next Generation Nuclear Plant (NGNP) Prismatic HTGR Conceptual Design Project, Conceptual Design Project – Steam Cycle-Modular Helium Reactor (SC-MHR), Engineering Development Plan (EDP)”, Rev. 1, 12/23/2010, Table 2-28, p89.
7. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 2.1.3, p16.
8. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 2.2.1, p17.
9. PLN-2498, “Next Generation Nuclear Plant Methods Technical Program Plan,” Rev. 2, Idaho National Laboratory, 09/27/2010
10. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 2.2.2, p17.
11. PLN-3305, “Heat Transport System Components Engineering Test Plan” Rev. 1, Idaho National Laboratory, 06/30/2010, Section 4.1.7.4 Vibration, Section 4.1.4.2 Turbulence and Noise, p33.
12. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, April 2011, Section 2.3.1, p18.
13. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, April 2011, Section 2.3.2, p19.
14. PLN-3305, “Next Generation Nuclear Plant Heat Transport System Components Engineering Test Plan” Rev. 1, Idaho National Laboratory, 06/30/2010, Section A-2.1 HTHL, p72.
15. PLN-2803, “Next Generation Nuclear Plant Reactor Pressure Vessel Materials Research and Development Plan”, Rev. 1, Idaho National Laboratory, 07/14/10.
16. PLN-2804, “Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan,” Rev. 1, Idaho National Laboratory, 09/23/2010.
17. INL/EXT-11-21397, “Assessment of NGNP Moisture Ingress Events”, Rev. 0, Idaho National Laboratory , April 2011.
18. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 2.3.3, p19.
19. PLN-2498, “Next Generation Nuclear Plant Methods Technical Program Plan,” Rev. 2, Idaho National Laboratory, 09/27/2010, Section 4.1.2.1, p17.
20. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 2.3.4, p20.
21. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory , April 2011, Section 2.3.5, p20.
22. PLN-2497, “Graphite Technology Development Plan,” Rev. 1, Idaho National Laboratory, 10/04/2010.

23. INL/EXT-11-21097, " HTGR Dust Safety Issues and Needs for Research and Development", Rev. 0, Idaho National Laboratory, June 2011.

24. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory , April 2011, Section 2.3.6, p21.

25. PLN-2497, "Graphite Technology Development Plan," Rev. 1, Idaho National Laboratory, 10/04/2010, Sections 2.1.2.2 and 3.1.1, p5 and p10.

26. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.4.1, p21.

27. PLN-3636, "Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program," Idaho National Laboratory, 09/30/2010.

28. PLN-2497, "Graphite Technology Development Plan," Rev. 1, Idaho National Laboratory, 10/04/2010, Section 5.1.2, p25.

29. PLN-2497, "Graphite Technology Development Plan," Rev. 1, Idaho National Laboratory, 10/04/2010, Section 5.2.1, p38.

30. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.4.2, p22.

31. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.4.3, p22.

32. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.4.3.1, p13.

33. JOI-2234 ISSN 0366-0885, "Fission Product Release Profiles From Spherical Htr Fuel Elements At Accident Temperatures", September 1988.

34. PLN-3636, "Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program," Idaho National Laboratory, 09/30/2010, Section 3.5.10, p51.

35. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, April 2011, Section 2.4.3.2, p24.

36. Gottaut, H., and K. Krüger 1989, December. Results of Experiments At The AVR Reactor, Forschungszentrum Jülich GmbH, Germany, and Arbeitsgenmeinschaft Versuchreaktor GmbH, Jülich, Germany.

37. PLN-3636, "Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program," Idaho National Laboratory, 09/30/2010, Section 3.5.10, p51.

38. INL/EXT-11-21097, "HTGR Dust Safety Issues and Needs for Research and Development", Rev. 0, Idaho National Laboratory , June 2011.

39. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.4.3.3, p24.

40. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.4.4, p25.

41. PLN-3636, "Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program," Idaho National Laboratory, 09/30/2010.

42. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.4.5, p26.

43. INL/EXT-11-21097, ''HTGR Dust Safety Issues and Needs for Research and Development'', Rev. 0, Idaho National Laboratory , June 2011, Section 2.1, p6.

44. INL/EXT-11-21097, “HTGR Dust Safety Issues and Needs for Research and Development”, Rev. 0, Idaho National Laboratory, June 2011.

45. PLN-2497, “Next Generation Nuclear Plant Graphite Technology Development Plan”, Rev. 1, Idaho National Laboratory, 10/04/2010, Section 5.1.4.3.

46. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory , April 2011, Section 2.4.6, p26.

47. ORNL/NRC/LTR-09/03, “Workshop on Nuclear Graphite Research”, September 9, 2009, Oak Ridge National Laboratory and Nuclear Regulatory Commission.

48. INL/EXT-11-21097, “HTGR Dust Safety Issues and Needs for Research and Development’’, Rev. 0, Idaho National Laboratory , June 2011, Section 2.2, p8.

49. INL/EXT-11-21097, “HTGR Dust Safety Issues and Needs for Research and Development”, Rev. 0, Idaho National Laboratory , June 2011.

50. PLN-2497, “Next Generation Nuclear Plant Graphite Technology Development Plan”, Rev. 1, Idaho National Laboratory, 10/04/2010.

51. PLN-2497, “Next Generation Nuclear Plant Graphite Technology Development Plan”, Rev. 1, Idaho National Laboratory, 10/04/2010, Section 2, p3.

52. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory , April 2011, Section 2.4.7, p27.

53. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 2.4.8, p28.

54. 12-9149697-000, AREVA NP Inc., “Pebble Bed Reactor Technology Plant Design Description”, Technical Data Record, 01/31/2011, Section 3.5, p107–119.

55. INL/EXT-10-19533, “NGNP Reactor Coolant Chemistry Control Study”, Rev 1, Idaho National Laboratory , November 2010.

56. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 2.4.9, p29.

57. PLN-3636, “Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program,” Idaho National Laboratory, 09/30/2010, Section 1, “Introduction,” p2; Section 3.4, “Fuel Performance Modeling,” p33–41.

58. PLN-3636, “Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program”, Idaho National Laboratory, 09/30/2010, Section 1, Page 5.

59. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory , April 2011, Section 2.5.1, p30.

60. PLN-2804, “Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan”, Rev. 1, Idaho National Laboratory, 09/23/2010, Section 4.1.2, Page 37.

61. INL/EXT-11-21397, “Assessment of NGNP Moisture Ingress Events”, Rev. 0, Idaho National Laboratory , April 2011.

62. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 2.5.2, p31.

63. 12-9151714-000, AREVA NP Inc., “Pebble Bed Reactor Technology Readiness Study, Technical Data Record”, 01/31/2011, Table 6-1, p140; Section 6.8, p153.

64. INL/EXT-11-21397, "Assessment of NGNP Moisture Ingress Events", Rev. 0, Idaho National Laboratory, April 2011.

65. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.5.3, p31.

66. PLN-2498, "Next Generation Nuclear Plant Methods Technical Plan", Rev. 2, Idaho National Laboratory, 04/19/2010, Section 4.1.2.1, p17.

67. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.5.4, p32.

68. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.5.5, p32.

69. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.5.6, p33.

70. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.5.7, p33.

71. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.5.8, p33.

72. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.6, p34.

73. PLN-2498, "Next Generation Nuclear Plant Methods Technical Plan", Rev. 2, Idaho National Laboratory, 04/19/2010, Section 4.1.2.5, p24.

74. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.7.1, p34.

75. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 2.7.2, p35

76. INL/EXT-11-21397, "Assessment of NGNP Moisture Ingress Events", Rev. 0, Idaho National Laboratory , April 2011.

77. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 3.1.1, p36.

78. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 3.1.2, p36.

79. PLN-2804, "Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan," Rev 1, Idaho National Laboratory, 09/23/2010, Section 6.5.1, p96.

80. PLN-2804, "Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan," Rev 1, Idaho National Laboratory, 09/23/2010, Section 5.1.3.1.2, p57.

81. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 3.1.3, p36.

82. PLN-3305, "Next Generation Nuclear Plant Heat Transport System Components Engineering Test Plan" Rev. 1, Idaho National Laboratory, 06/30/2010, Section 2.4.2.2.2, p21.

83. INL/EXT-10-19329, "HTGR Lessons Learned NGNP", Rev. 1, Idaho National Laboratory, April 2011, Section 3.1.4, p37.

84. PLN-3305, "Next Generation Nuclear Plant Heat Transport System Components Engineering Test Plan" Rev. 1, Idaho National Laboratory, 06/30/2010, Section 4.2.4.1, p46.

85. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 3.1.4, p37.

86. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 3.2.1, p37.

87. PLN-2804, “Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan,” Rev. 1, Idaho National Laboratory, 09/23/2010

88. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 3.2.2, p38.

89. PLN-2804, “Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan”, Rev. 1, Idaho National Laboratory, 09/23/2010.

90. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 3.2.3, p38.

91. PLN-3305, “Next Generation Nuclear Plant Heat Transport System Components Engineering Test Plan”, Rev. 1, Idaho National Laboratory, 06/30/2010, Section 4.2.4.1, p46.

92. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 3.2.4, p38.

93. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 3.3.1, p39.

94. PLN-2804, “Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan”, Rev. 1, Idaho National Laboratory, 09/23/2010.

95. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 3.3.2, p39.

96. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 3.3.3, p39.

97. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 3.4.1, p39.

98. PLN-3305, “Next Generation Nuclear Plant Heat Transport System Components Engineering Test Plan”, Rev. 1, Idaho National Laboratory, 06/30/2010, Section 4.2.5, p47.

99. INL/EXT-09-16598, “Next Generation Nuclear Plant Technology Development Roadmaps: Forward for 750-800C Reactor Outlet Temperature”, Idaho National Laboratory, August 2009. Section 3.4, p69–74.

100. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 4.1.1, p41.

101. PLN-2804, “Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan”, Rev 1, Idaho National Laboratory, 09/23/2010

102. INL/EXT-11-21397, “Assessment of NGNP Moisture Ingress Events”, Rev. 0, Idaho National Laboratory, April 2011.

103. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 4.1.2, p42.

104. PLN-2804, “Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan”, Rev 1, Idaho National Laboratory, 09/23/2010, Appendix A, Table 15, p203.

105. PLN-3305, “Next Generation Nuclear Plant Heat Transport System Components Engineering Test Plan”, Rev 1, Idaho National Laboratory, 06/30/2010, Section 4.2.3, p46.

106. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 4.1.3, p43.

107. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 4.1.4, p44.

108. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 4.1.5, p44.

109. PLN-3479, “Next Generation Nuclear Plant R&D Test Control Plan - High Temperature Hydrogen Permeation through Nickel Alloys”, Rev. 1, Idaho National Laboratory, 04/19/2010.

110. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 4.1.6, p45.

111. PLN-3305, “Next Generation Nuclear Plant Heat Transport System Components Engineering Test Plan”, Rev. 1, Idaho National Laboratory, 06/30/2010, Section A-8.1 Steam Generator Model.

112. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 5.1.1, p46.

113. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 5.2.1, p46.

114. PLN-3305, “Next Generation Nuclear Plant Heat Transport System Components Engineering Test Plan”, Rev. 1, Idaho National Laboratory, 06/30/2010, Section 4.1.7, Instrumentation, p35.

115. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 5.2.2, p47.

116. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 5.3.1, p47.

117. INL/EXT-11-21397, “Assessment of NGNP Moisture Ingress Events”, Rev. 0, Idaho National Laboratory, April 2011.

118. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 5.3.2, p47.

119. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 5.3.3, p47.

120. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 5.3.4, p47.

121. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 5.3.5, p48.

122. INL/EXT-11-21397, “Assessment of NGNP Moisture Ingress Events”, Rev. 0, Idaho National Laboratory, April 2011.

123. INL/EXT-10-19329, “HTGR Lessons Learned NGNP”, Rev. 1, Idaho National Laboratory, April 2011, Section 5.3.6, p48.

124. INL/EXT-11-21397, “Assessment of NGNP Moisture Ingress Events”, Rev. 0, Idaho National Laboratory, April 2011.

125. INL/EXT-11-21097, “HTGR Dust Safety Issues and Needs for Research and Development”, Rev. 0, Idaho National Laboratory, June 2011.

126. PLN-3305, “Next Generation Nuclear Plant Heat Transport System Components Engineering Test Plan”, Rev. 1, Idaho National Laboratory, 06/30/2010.