HIGH TEMPERATURE GAS-COOLED REACTOR
KNOWLEDGE MANAGEMENT
WHITE PAPER (2 OF 3)

Lessons Learned from Supporting NRC Licensing and Regulatory Activities for Fort St. Vrain (FSV) and from Supporting the NRC Pre-Application Licensing Review of the Modular HTGR

By

David L. Moses

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CONTACT: Jonathan Barr, NRC Project Manager
Phone: (301) 415-5330, Email: jxb7@nrc.gov

NRC INTRODUCTION

This white paper is one of three written for the NRC's High Temperature Gas-cooled Reactor (HTGR) knowledge management program and community of practice, as per contract N6217 at Oak Ridge National Laboratory (ORNL). Each of the three white papers was written by an individual with significant expertise in a segment of the HTGR knowledge base and discusses historical perspectives and lessons learned from their experiences with HTGR technology and licensing. These white papers serve as a mechanism for transferring tacit HTGR knowledge to the NRC and its staff. Each white paper was peer reviewed by either NRC or ORNL staff.

This white paper, authored by David Moses, discusses lessons learned from the licensing and regulation of the Fort St. Vrain reactor and pre-application licensing review of the MHTGR. These discussions range from technical issues (e.g. water ingress and metallurgy) to procedural issues (e.g. communication with NRC staff). Dr. Moses' perspectives draw from his experiences in the 1980s in which he was principal investigator and task manager for the NRC's Fort St. Vrain Technical Specifications Upgrade Program and for the NRC's Evaluation of the MHTGR Preliminary Safety Information document. Dr. Moses has a Bachelors Degree in Mathematics and Physics and Masters and Ph.D. degrees in Nuclear Science and Engineering.

This white paper was peer reviewed by NRC staff.

All opinions in this white paper are those of the author, not of the NRC.

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1. Introduction

As indicated by its title, this paper addresses two aspects of the history of gas-cooled reactor regulation by NRC: Fort St. Vrain (FSV) and the Modular MHTGR and the lessons learned that may be important in reviewing future license applications for advanced designs such as the Gas-Turbine Modular Helium-cooled Reactor (GT-MHR) and the Pebble Bed Modular Reactor (PBMR). Several of the lessons learned at FSV as discussed below are a revisit to issues previously and recently described in Appendix A to NUREG/CR-6830, *Fort Saint Vrain Gas Cooled Reactor Operational Experience*, September 2003.

2. Fort St. Vrain

When reviewing the licensing and regulatory experience at FSV as it may apply to the future licensing and regulation of new gas-cooled reactor plants, the present day reader should bear in mind that FSV was regulated not only in the past in a time period of an evolving regulatory process but also under a somewhat different oversight structure than was used at its contemporary light water reactors (LWRs). This paper attempts to explain some of the differences that existed and how these differences can impact subsequent licensing and regulatory analyses that would suggest looking back to FSV for precedents.

Both the construction permit and the operating license granted to FSV were issued under Section 104(b) of the *Atomic Energy Act of 1954*, as amended, (AEA). Therefore, the Atomic Energy Commission (AEC) and later the Nuclear Regulatory Commission (NRC) imposed “the minimum amount of such regulations and terms of license as will permit the Commission to fulfill its obligations under this chapter” of the AEA. Specifically, FSV was licensed under the provisions of *The Code of Federal Regulations*, Title 10—Energy, Part 50, Section 21, “Class 104 Licenses; for Medical Therapy and Research and Development Facilities (10 CFR 50.21). The original AEC licensing officials commented verbally that FSV was considered by them to be a “research and development reactor that could be shutdown immediately if there were any real safety problems.” Prior to the 1970 AEA amendment deleting the “practical value determination” previously required under Section 102 of the AEA, 10 CFR 50.21 had required that Class 104 licenses for the Power Reactor Demonstration Projects be converted to Class 103 licenses once the practical value determination had been made and so required the regulatory process applied to these nuclear power plants (NPPs) to anticipate the conversion of the license. Following the amendment of AEA in 1970 while FSV was still under construction, the regulatory requirement to convert to a Class 103 license was dropped. Thus, the NRC allowed a generous amount of latitude in regulatory interpretation of applicability to FSV consistent with the legal bases for its Class 104(b) operating license. This latitude may make it difficult to establish currently acceptable licensing and regulatory precedents from FSV that can be applied directly to new gas-cooled reactors that would be licensed or certified under the current provisions of 10 CFR Parts 50 and 52. However, this same experience may point to aspects that may need particular attention to ensure history does not repeat itself.

The early years of FSV start-up and operations were reviewed in a series of seven reports published by the Electric Power Research Institute (EPRI) between 1978 and 1982. These reports are tabulated below for reference:

Title	Author (s)	Date	Document Number
Comprehensive study of the operating and testing experience during the startup and initial operation at the Fort St. Vrain HTGR. Phase 1. Preoperational testing	Van Howe, K.R.; Raudenbush, M.H.	1978 Feb 01	EPRI-NP-697
Comprehensive study of the operating and testing experience during the startup and initial operation at the Fort St. Vrain HTGR. Phase 2. Core loading, physics, and low power testing	Van Howe, K.R.; Raudenbush, M.H.	1978 Feb 01	EPRI-NP-698
Comprehensive study of the operating and testing experience during the startup and initial operation at the Fort St. Vrain HTGR. Phase 3. Initial startup and operation at power phase - Key phase report	Van Howe, K.R.; Raudenbush, M.H ; Colgan,G.	1978Jun 01	EPRI-NP-760
Comprehensive study of the operating and testing experience during the startup and initial operation at the Fort St. Vrain HTGR. Summary report	Van Howe, K.R.; Raudenbush, M.H.	1978 Aug 01	EPRI-NP-890-SY
Comprehensive study of the operating and testing experience during the startup and initial operation at the Fort St. Vrain HTGR. Phase 4. Follow-on studies. Final report	Van Howe, K.R.; Raudenbush, M.H ; Colgan,G.	1979 Nov 01	EPRI-NP-1214
Fort St. Vrain experience: first refueling/maintenance outage. Final report	Van Howe, K.R.	1979 Dec 01	EPRI-NP-1292
Assessment of effects of Fort St. Vrain HTGR primary coolant on Alloy 800. Final report	Trester, P.W.; Johnson, W.R.; Simnad, M.T.; Burnette, R.D.; Roberts, D.I.	1982 Aug 01	EPRI-NP-2548

Table 1: EPRI reports which reviewed FSV start-up and operation

The period of FSV operations between 1974 and 1989 was one in which the NRC relied on the evolving regulatory process to deal with emerging safety issues such as the Browns Ferry fire, the Three Mile Island Action Plan, the need for environmental qualification of equipment important to safety, the need to update and maintain current the design bases presented in the Final Safety Analysis Report (FSAR), standardization of technical specifications, etc... The Class 104(b) operating license issued for FSV and the NRC cognizant-staff interpretation of the statutory basis for that license meant that FSV regulatory requirements were tailored to allow more flexibility than perhaps was afforded other contemporary NPPs that were licensed under Section 103 of the AEA. Some examples from FSV include cooling systems, reactor physics, structural-mechanical behavior, and radiological behavior, and these issues and lessons learned are addressed in the remainder of Chapter 2.

2.1 Applicant PSAR and FSAR

The initial assumption made by the reactor vendor (General Atomics) and license applicant (Public Service Company of Colorado) was that there were so many ways to cool the proposed FSV reactor core that no single one needed to be relied upon. The AEC Division of Reactor Licensing (DRL) required from the review of the FSV Preliminary Safety Analysis Report (PSAR) in 1967-68 that the applicant define the precise cooling methods to be relied upon during accident conditions assuming both a pressurized and depressurized helium coolant system. Per the footnote on page of 5 of ACN 8808050277 (not released in the NRC Public Document Room, PDR), the initially envisioned Class I safety-related cooling systems in the initially proposed Technical Specifications included both small (12.5% capacity) condensate pumps, both auxiliary boiler feed pumps, both condensate storage tanks, the decay heat removal exchanger, and the emergency condensate header. By inference drawn off the docket record, only the emergency condensate header out of this set of equipment was determined by AEC review to meet design requirements for the design basis earthquake and the maximum tornado. However, the FSV Updated FSAR still credited the use of this equipment as part of the primary success paths in response to non-seismically-induced anticipated transients such as loss-of-offsite power. The only fully seismically-qualified and wind-qualified cooling system had to rely on the electrical-driven and diesel-driven firewater pumps and the unpolished fire water reservoir both to drive the Pelton wheel water-turbine helium circulator drive and to provide cooling water to the steam generators and the cooling tubes on the liner of the prestressed concrete reactor vessel (PCRV). Having to rely on the fire water system to respond to a design basis event would mean so much contamination and corrosion in the secondary cooling system as to have likely precluded restart following a seismic or wind event. For both loss of reactor steam to drive the steam-driven feed water pumps and for certain depressurization events, the Updated FSAR indicated that the Pelton wheel turbine drives for the helium circulators would be supplied by the fixed speed electrical feed water pump; unfortunately, feed water flow from the fixed speed electrical pump to the Pelton wheel turbines had to be controlled by a throttle valve since the feedwater pump speed was not controlled. This led to problems in terms of damage to the throttle valve or the Pelton wheels so that the licensee was reluctant to use the fixed speed electrical-drive feed water pumps and in most cases involving loss of secondary steam pressure resorted to other methods and fortunately never had to deal with the Rapid Depressurization Accident (Design Basis Accident No. 2 or DBA-2) which would have had to rely on this system to provide sufficient cooling flow to cool the core during a depressurization from full power. A detailed review of the FSV safety related cooling systems was documented as part of the Technical Specification Upgrade Program (TSUP) in ACN 8808050277 (not released to the NRC PDR). Cooling system issues associated with the design basis fire are discussed next below.

LESSON LEARNED:

An applicant needs to be very clear in the PSAR and FSAR about (1) the selection of the principle design criteria for cooling systems and how these bridge to and accommodate meeting the safety functions underlying the NRC's General Design Criteria (GDC), (2) the required seismic and environmental qualifications for the cooling systems and equipment to be relied upon as safety-related, and (3) the instrumentation and surveillance mechanisms that will be used to apply Technical Specifications to the equipment so as to satisfy the appropriate criteria for selecting Limiting Conditions for Operation and their associated Surveillance Requirements in 10 CFR 50.36(c)(2) and (3). One of the "truisms" often heard from ORNL researchers who have been involved in developing instrumentation and controls systems for some f the earlier advanced reactor concepts is that designers and promoters of concepts often do not think about such instrumentation and control systems until after the other systems are made a key part of the design; then one finds out they cannot measure or diagnose directly the safety or operating parameters needed. This author has observed that NRC staff looking at a new proposed reactor often fail to ask "What parameters and how are they going to be measured to demonstrate that this new plant is operating safely?" While 10 CFR 50.34 and the Standard Review Plan (NUREG-0800) imply that during the preparation and review of safety analyses some attention has to be paid to the requirements of 10 CFR 50.36 for technical specifications in looking at a new design, this author's finding from participating in the MHTGR pre-application review and from observing that for PRISM (Power Reactor Inherently Safe Module) and SAFR (Sodium Advanced Fast Reactor) is that this is rarely the case. The FSV licensing experience speaks volumes about that being the wrong approach. In its support for the pre-application review of the Modular MHTGR, the Department of Energy (DOE) promoted an Integrated Approach to design that involved defining its own set of top-down Principal Design Criteria and then bridging these to demonstrate how they met, satisfied or accommodated NRC's various regulatory requirements; DOE never produced the bridging document. Also missing was how to develop and practically implement technical specifications for the Modular MHTGR so DOE's Integrated Approach was not all that "integrated."

2.2 Fire Protection Program

In complying with 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," the method of "safe shutdown" cooling following the design basis fire was defaulted by NRC to the FSV Design Base Accident No. 1 (DBA-1) with fire water cooling the PCRV to maintain reactor vessel/primary containment integrity. In this scenario, fuel damage would occur during the resulting transient but would be contained to prevent and mitigate off-site doses resulting from the fuel damage, and "safe cold shutdown" would not be achieved within the 72 hours required in the regulations. In fact, a major question that concerned one NRC-Office of Nuclear Reactor Regulation (NRR) Project Manager was whether allowing for fuel damage actually met the intent of the regulation with regard to achieving "safe shutdown." However, since the FSV Appendix R accident mitigation approach prevented off-site consequences and since achieving "safe cold shutdown" as specifically required by the regulation would have been prohibitively expensive to implement (likely leading to an even earlier shutdown of the plant), the regulatory latitude permitted for a Class 104(b) licensee allowed for a compromise that did not adversely impact public safety. Today, the NRC would likely not permit such latitude in a new plant.

LESSON LEARNED:

As noted in Item 1 above, the applicant needs to demonstrate that fire protection and the mechanisms for responding to a fire to achieve hot and cold safe shutdown are consistent with regulatory requirements, and this consistency with regulatory requirements needs to be reflected in selecting Principal Design Criteria, equipment qualification requirements, and the Technical Specifications.

2.3 Probabilistic Risk Analysis Data

A long-standing claim by General Atomics is that the large HTGRs (including FSV) marketed by them in 1970s and early 1980s could be expected to have low probability of reactor trip and turbine trip during loss-of-offsite power events or operator-elected disconnection from an unstable grid. In their HTGR Accident Initiation and Progression Analysis (GA-A-13617), General Atomics claimed that turbine trip causing reactor trip in such events was a frequency as low as 0.1 per demand and was based on data from British Magnox reactor experience. As it turns out the General Atomics designs including FSV were based on using one large turbine-generator as opposed to multiple 90-100 MWe turbine-generators coupled to two Magnox reactors and often with a dedicated 30 MWe hotel-load turbine-generator used in Britain. At FSV, which was never able to test from full power, the initial design configuration could not handle even power rollbacks very well since the reheat steam coming from the exhaust of the high-pressure turbine would exceed the allowable temperature for driving the steam-turbines on the helium turbines. Therefore a system was developed to apply feedwater as a cooling spray into the reheat steam line to cool the reheat steam before it reached the helium circulator steam-turbine drives. Controlling the proper cooling water flow to get the needed reheat steam attemperation proved very difficult leading more often to circulator trip rather than to a successful power runback without reactor trip.

LESSON LEARNED:

Be very cautious of the logic presented as data in probabilistic risk analyses especially about the performance of non-safety-related systems and allow credit only if start-up testing demonstrates the reliability of the claimed advantage of a non-safety-related mechanism for assuring plant integrity during upsets.

2.4 NRC Staff Involvement

In initially implementing the provisions of *The Code of Federal Regulations*, Title10—Energy, Part 50, Section 71, “Maintenance of Records, Making of Reports” (10 CFR 50.71), issued in July 1980, with regard to updating and maintaining current the FSAR, the FSV initial update did not include a comprehensive revision of changes made in the NPP’s licensing basis between 1974 and 1982. The discrepancies became obvious during the FSV TSUP led by NRC-NRR in the late 1980s. One of the NRR TSUP criteria was to review the FSAR for safety-related commitments that were not reflected in the technical specifications appended to the operating license. It became evident during the FSAR review that in many cases, there were technical specifications that had been implemented since FSV start-up where no bases were documented in the FSAR. One of the most interesting examples was the base reactivity curve that had been implemented to address the large reactivity change observed in the expected critical position following a major water ingress event in 1974. The base reactivity curve was not explained in any documentation in the FSAR nor in any other topical report. The base reactivity curve was generated by the designer, General Atomics, reviewed by the FSV Nuclear Facility Safety Committee in which a staff person from General Atomics was required by the technical specifications to participate by direction of NRC, and was submitted to NRC Region IV but was never sent on to NRR for review. The exact purpose, meaning and utility of the curve to the safety of plant operations and how the curve related to any measurable parameter of the reactor were not obvious due to lack of documentation other than the cursory bases accompanying the technical specification, but it was a technical specification limiting condition for operation nonetheless.

LESSON LEARNED:

Keep the appropriate experts at NRC involved in following updates to the FSAR and the emerging changes to facility design and their associated changes in Technical Specifications.

2.5 Experimental Data and Level of Documentation

Besides the issue of the base reactivity curve, there were other aspects of the safety-related reactor physics and nuclear design that were different from most other contemporary licensed NPPs. The information documented in Section 4.3 of the FSV FSAR had little to do with the nuclear analysis techniques actually used by the designer and the licensee for the analysis of FSV, including generation of the base reactivity curve. The core-reload nuclear design reports were proprietary to General Atomics and were not submitted to NRC for review. The nuclear design-related start-up test data were reported as required by NRC Regulatory Guide 1.68, but were reported only as lists of calculated and measured data with no documentation nor analysis as to how the values reported were calculated, measured, or reconciled. This approach was distinctly different from that of other Power Reactor Demonstration Projects such as Yankee-Rowe where extremely detailed start-up testing reports were generated. During the AEC-DRL review of the FSV PSAR in 1967-68, a number of requests for additional information were made by the regulator, and, in Amendment 3 of the PSAR, the applicant made commitments to address the regulator's requests. However, at the discretion of the NRC licensing official FSV received its Class 104(b) operating license without the regulator ever revisiting the commitments to address the requests for additional information on the nuclear design methods and their basis of qualification. As discussed in the Appendix A to the Technical Evaluation Report of the Nuclear Design of the MHTGR (ACN 8903220327, Project No. 672), FSV contributed little to the closure of nuclear design issues for the MHTGR because so little of substance and detail had been documented to support the analytical methods used at FSV. It is understood that much of the latitude granted under FSV's Class 104(b) license was to allow the designer and the licensee the opportunity to develop additional data to support the design of future HTGRs. Unfortunately, the designer failed to take adequate advantage of this allowance with respect to securing nuclear design data as evidenced in the findings documented in Section 4.3.5.B of NUREG-1338, "Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor," March 1989, which states: "The staff found that (1) there is a paucity of relevant experimental data and (2) there is a lack of documented analysis of the existing data using the analytical methods employed for the MHTGR nuclear design. As a result of this review and DOE's reevaluation, DOE changed its original position on research needs and stated that it planned to develop a chapter on reactor physics in the RTDP (Reactor Technology Development Plan), as described in Section 4.3.4 of the RTDP. The end product of this program should be adequate integral data for the construction and validation of an acceptable methodology for the MHTGR nuclear design."

LESSON LEARNED:

The Class 104(b) licensing latitude allowed by the FSV regulators, who had declared in their safety evaluation that FSV was "thermally and neutronically sluggish," had the unfortunate impact of contributing to the "paucity of relevant experimental data" needed to support the reactor physics safety case for future gas-cooled reactors. If a new demonstration reactor is to be constructed and operated for testing, a rigorous documentation program based on Regulatory Guide 1.68 is required where such documentation is very explicit in detailing how calculations are done, how measurements are made (with all uncertainties accounted for), and how analytical and experimental results are reconciled.

2.6 Industry Codes and Standards

In 1981 when ASTA, Inc. reviewed the licensee-proposed FSV in-service inspection requirements for the NRC under contract through Los Alamos National Laboratory, ASTA concluded (ACN 8201130206) that, for the PCRV penetration double closures, the requirements of Section XI of the *American Society of Mechanical Engineers Boiler and Pressure Vessel Code* for Category I and II structures would not be met by the external visual inspections only (i.e., no surface or volumetric inspections) of the outer closure as proposed by the licensee. However, consistent with the regulatory latitude afforded under FSV's Class 104(b) license, NRC (ACN 8303150001) accepted the licensee proposal for visual inspections only without further addressing or reconciling the regulatory conclusion with the technical opinion from ASTA. Although not documented in the record, NRC recognized that the ASTA recommendation would have been extremely difficult to implement due to limited access for performing volumetric inspections, and thus NRC granted a less stringent requirement consistent with the plant's Class 104(b) license and the recognition that a closure failure was unlikely to occur and equally unlikely to cause significant off-site exposures. The importance of the discrepancy in NRC conclusions on in-service inspections requirements was noted during a review (ACN 8801080075) of the licensee's probabilistic re-analysis of the likelihood of occurrence of a rapid depressurization accident (i.e., event DBA-2 as discussed in ACN 8603050288). The DBA-2 re-analysis was initially part of the TSUP but was reviewed as part of NRC's response to the Chernobyl accident. The licensee's re-analysis of the DBA-2 likelihood was performed by the designer and was based on the argument that the failure probability of a FSV PCRV large-sized penetration double closure was analogous to the accepted frequency (10^{-7} per year) for the rupture of a LWR pressure vessel. The fallacy in the designer's logic was that the accepted frequency for LWR vessel failures is based on the assumption that the vessel is inspected to the requirements of Section XI of the *American Society of Mechanical Engineers Boiler and Pressure Vessel Code*, and this was not the case per the NRC-chartered ASTA review. Since the circulating radioactivity (source term) in the FSV coolant was more than two orders of magnitude below the value assumed in the DBA-2 bounding analysis in the FSAR, the uncertainty in the estimate in DBA-2 frequency had little safety significance, but the regulatory allowances to FSV under its Class 104(b) license meant that FSV was quite often the exception to standard regulatory practice as applied to other commercial NPPs. This fact makes it very difficult to draw any generalizations from FSV licensing and operations that can readily be applied to the licensing of future gas-cooled reactors without a careful consideration of the specific circumstances that were applicable to FSV. Also, since at the time, the NRC staff did not always clearly document in its safety evaluations whether a Class 104(b) exception was being granted, it is often difficult to understand the thought process behind a given regulatory decision for FSV.

LESSON LEARNED:

Consistent with the regulatory requirements of 10 CFR 50.55a and the guidance of various Regulatory Guides endorsing industry standards when appropriate or applicable, industry codes and standards should be applied in a reasonable but consistent manner to new and innovative designs; exceptions should be acknowledged, explained and documented with regard to limitations to avoid bad precedents that could lead to indefensible safety evaluations or worse to an unrecognized vulnerability.

2.7 Water Ingress

FSV suffered many operational issues due to water ingress. Few were truly safety significant; some were. A summary listing of unique but significant water ingress problems is given as follows:

- The major problem was associated with the use of water-lubricated shaft-bearings on the steam-driven helium circulators and the complex buffer-mid-buffer system that separated the water lubrication injection and bleed from the primary helium system using a purge of clean helium from the helium purification system. Upsets in the purified helium purge led to frequent and significant water ingress events. The next generation HTGRs will likely use either oil-lubricated bearings as used in Peach Bottom Unit 1 and in the British and German gas-cooled reactors, although oil ingress is a problem, or the gas-lubricated magnetic bearing similar to those used in the pumps for natural gas distribution. One of the largest water ingress events occurred during an early shut down in 1974 and was not noticed until the effect of the water on the expected critical position of the control rods was noticed on restart (see Item 4 above). In the last few years of operation, water ingress was still occurring; one of the more significant events at this time was due to a water leak into a steam generator penetration interspace purged by the helium purification system but ignored by the plant operators because the non-safety-related moisture alarms on the interspace "were always going off."

- Large water ingress events at FSV led to substantial amounts of water vapor entering the helium purification stream. A water ingress event during shutdown led to moisture hold-up in the core graphite structures and in the kaowool insulation on the PCRV liner so that water levels would rise as the reactor system heated up on restart. Substantial amounts of water vapor entering the FSV helium purification system from the primary coolant would pass through the high-temperature carbon trap for iodine, overwhelm the chiller used to precipitate out smaller amounts of water vapor, saturate the titanium sponge filter (molecular sieve) that was supposed to capture any remaining water, CO, CO_2, or hydrogen molecules, and break through to the liquid nitrogen cooled charcoal bed krypton trap leading to icing on the trap and the passage of water vapor or droplets and gaseous fission products into the helium purge gas. In this way, water ended up in the carbon steel (not more expensive stainless steel) piping and tubing of the purified helium supply to various purge locations including the control rod drive mechanisms. Due to water breakthrough, corrosion (rust) formed in the carbon steel purge gas tubing leading to flow restrictions and the movement of both water and corrosion particles into bearings on both the circulators and the control rod drive mechanisms. Evaluation of the second of two partial failure to scram events (the first involving two control rods and the second six control rods) found that there was insufficient purge gas flow to the control rod drive motors (essentially a wench with a cable hanging off with the control rod at the end of the cable); corrosion particles made it to the wench causing sticking and the inconel cable from which the control rod was suspended showed evidence of stress corrosion likely caused by the mixture of moisture and chlorine exhaled from the graphite and/or the kaowool insulation. Following the second partial failure to scram event, FSV had to implement a controversial Technical Specifications Surveillance Requirement involving rod drop testing with the wench motor acting like a generator to produce a back electromotive force (EMF) that would be measured to determine if the rod was falling freely or had evidence of cable sticking caused by corrosion or debris. A better method for establishing the operability of the control rod drive mechanism was needed.

- Water ingress also entered through the purified helium purge gas system into the hoppers holding the reserve shutdown system (RSS) above the core. The RSS used small balls of boron carbide (B_4C) and graphite. During a serious Anticipated Transient Without Scram (ATWS), the RSS was to provide an alternate shutdown system that was actuated when the hopper doors were opened by the operator and the balls were allowed to fall into the control rod drive holes drilled through the graphite core elements that accommodated control rods. The licensee's procurement of the B_4C balls had failed to specify a sufficiently low level of boron oxide (B_2O_3) constituent so that water ingress from the helium purge caused the leaching of B_2O_3 out of the balls and the formation of boric acid crystals on the surfaces of the balls so that the RSS material stuck together and the balls would not fall into the core when the hopper was opened. This problem was discovered during periodic technical specification surveillance.

- Water leaks also developed in the steel piping that made up the PCRV liner cooling system; ingress into the reactor was limited and would only occur when the reactor was depressurized during shutdown. Attempts to correct this problem using a polymer epoxy flush had limited success. The cooling tubes were inset into the concrete of the PCRV underneath the steel liner wall and its covering insulation that prevented hot helium from impinging on the liner surface.

- FSV used two similar detection systems for detecting water ingress into the primary coolant system. Downstream from the steam generators, the dew-point moisture detectors sampled helium flow from each loop and used a light and photocell arrangement to detect fogging or frost formation on the surface of a mirror chilled by nitrogen gas from a liquid nitrogen supply. These detectors were designed to detect a large break in the steam generator tubes and to alarm and to actuate the plant protection system to scram the reactor and trip off secondary water flow to the affected steam generator. The sampling for the analytical moisture detectors was situated further away from the steam generators and consisted of a similar set up except that chilled water was used to cool the mirrors and the signal from the detectors provided diagnostic and alarm functions. The analytical moisture detectors were equipped with a pumped sample line so that during low flow or low pressure conditions of the reactor at low power or during shutdown moisture monitoring could still be performed. There were other type detectors in some of the PCRV penetration interspaces but these detectors had only local alarms, and as noted previously were routinely ignored due to frequent alarms.

LESSON LEARNED:

FSV taught us to avoid the possibility of getting water into the reactor primary system of an HTGR, and FSV also taught us that ingress during shutdown conditions is likely to be the highest risk with subsequent moisture hide-out occurring due to water being absorbed into the graphite and in-vessel insulation. The proposed Modular HTGR shared with FSV the use of a steam generator with secondary high-pressure steam-water that would cause in-leakage of steam or water to the primary coolant system in the event of a steam generator tube leak. The proposed GT-MHR and PBMR will operate with high-pressure helium on the primary side and lower pressure cooling water in the secondary side heat rejection system. These proposed designs will be most at risk for water ingress during shutdown, and one area needing particular attention for detection and monitoring will be the heat exchanger in the shutdown cooling system (SCS). If a large water ingress event were to occur during shutdown, the use of a helium purification system similar to that used at FSV for removing large quantities of water is not advised since a large water ingress event tends to overwhelm this system's capacity (typically about 0.3 percent of the total helium flow during operation) as seen at FSV. A dedicated dry-out system that would then purge the dried helium though the helium purification system might be advantageous but the best approach would be better moisture detection systems tied to the most likely places where water ingress could occur during shutdown so that quick action can be taken to terminate and limit the amount of the ingress. The most likely sources of ingress are the heat exchanger in the SCS, the pre-cooler and inter-cooler, and any water cooling jackets on the refueling machine but with the first likely to be the largest and most likely source of ingress. Finally, small helium lines providing purge gas from the helium purification system should most likely be fabricated from a ferritic stainless steel to resist both corrosion and stress corrosion cracking so as to preclude line blockages, carrying of corrosion debris to bearings or other moving parts, and tube cracking leading to small helium leaks or loss of purge flow to essential equipment such as the control rod drive mechanisms.

2.8 Metallurgy of Primary Coolant System Boundaries

The event that finally brought FSV operations to an end was the severe cracking of the incoloy-800 steam generator super-heater headers. Replacement of the headers was deemed too expensive to justify a plant restart given the long history of operational problems at the plant. The header cracks were caused by vibration and thermal cycling of the header material which turned out to have large course grain sizes that made the metal structure prone to cracking. Had the microstructure of the inconel been held to a fine grain size during fabrication and acceptance inspection, the problem may never have occurred leading to a much longer plant life. Although the header cracks were not really a safety issue, attention to the acceptability of metallurgy is a key consideration. During the DOE-funded work on the New Production Reactor Modular HTGR, which was being pursued in parallel to the civilian Modular HTGR, the reactor vendor argued that the once-though steam generator superheater section should be fabricated from Alloy 800H with a bimetallic weld to the 2.25Cr-1.0Mo ferritic stainless steel evaporator section. However, Alloy 800H has a higher carbon content and was thus very susceptible to sensitization and stress corrosion cracking when exposed to water. In meetings with DOE, the reactor vendor claimed that such concerns were not an issue since the superheater would be operated with dry steam; however, when it was asked how many hours the superheater would be operated flooded for shutdown cooling, the answer was about 5000 hours over the life of the plant which was plenty of time for sensitization and the onset of stress corrosion cracking in the hottest section of the steam generator. If superheater flooding occurred during a plant trip caused by detection of a caustic breakthrough in the demineralizer resins cleaning the condensate and feedwater systems, the superheater section would likely become very sensitized to stress corrosion cracking.

LESSON LEARNED:

The detailed metallurgy on all boundaries of the primary coolant system including the safe shutdown cooling system, the helium purification system, etc…, needs to be well understood with regard to its ability to maintain integrity during normal operations and upset conditions. Fatigue, corrosion, creep rupture and all other threats to integrity must be addressed; never assume anything will stay dry or not be exposed to corrosive chemicals that either may be used in the reactor system somewhere or may have been used in the fabrication of components and thus may be present in small contaminant-level quantities within the reactor.

2.9 Radioactive Contamination of Primary Coolant

As indicated previously, FSV had a much lower level (two orders of magnitude lower) of circulating radioactivity than that assumed in the DBA-2. The most troubling radiological release event came near end of life from tritiated water drained from the helium purification system during maintenance where both the chiller and the titanium molecular sieve pick up tritium in the form of tritiated water molecules. Tritium is formed during reactor operation primarily in neutron capture by the small amount of helium-3 in natural helium recovered from natural gas wells. The event that caused concern at FSV was the fact that water being drained from the helium purification system was improperly dumped into a drain to the reactor building sump instead of the liquid waste drain; this was a design flaw in the drainage system. Grab samples on reactor building sump discharge found that tritium was present and led to an attempt to find a way to continuously monitor oily-water sump discharges for a beta-emitter. This proved impossible to do. The event could have been avoided if the plant had carefully mapped the discharge points of all floor drain lines and recognized that liquids drained from the primary system must be considered radioactive until tested and should be properly routed to waste drains when in doubt.

LESSON LEARNED:

Always assume that items or material (solid, liquid or gas) coming from primary coolant exposures may be radioactive or have radioactive contamination. Operators should plan the disposition routes for such items or material and write the inspection and operating procedures accordingly

3. Modular MHTGR

Several issues associated with the Modular HTGR have already been addressed above in comparison to events at FSV. As the author of the Technical Evaluation Report of the Nuclear Design of the MHTGR (ACN 8903220327, Project No. 672) supporting the staff's conclusion in NUREG-1338, this author found that the main issues for nuclear design of the Modular MHTGR were the lack of appropriate detailed experimental results and evaluations of the reactor physics characteristics of the core. Also as pointed out in ACN 8903220327, it is necessary to perform a detailed and rigorous analysis of reactor physics parameters and their uncertainties against the plant's proposed Principal Design Criteria and/or the NRC GDC relevant to the neutronics aspect of the core design while keeping in mind how one would write Technical Specification and Surveillance Requirements to meet commitment in the FSAR. Additional concerns include:

- Near end of cycle, there is a need for rigorous quantification of the uncertainties in positive reactivity contributions of the 0.3 eV fission resonance of plutonium-239 and the precipitous drop in the capture cross sections of xenon-135 and samarium-149 between 0.1 and 0.3 eV to the sign of the moderator temperature coefficient of reactivity at the normal operating temperature of the core graphite. Under certain conditions near end of cycle when decay heat loads will be highest, it may be possible that the temperature reactivity is zero or positive during a positive reactivity insertion transient (such as rod withdrawal) until the core graphite heats up enough during the transient for the tail on the thermal capture resonance of plutonium-240 to cause the sign of the temperature reactivity coefficient to go negative. Given a loss of forced cooling event occurring immediately after the reactivity transient, then the initial fuel and core graphite temperatures could be much higher than that assumed for loss-of forced cooling from normal full power conditions. Higher values in core temperature and stored heat in the initial core conditions for the transient can lead to higher peak fuel temperatures during the core conduction cool down.

- There is a need for quantification of the reactivity effects of water/steam ingress during power operations and the uncertainties in these values. There are no good water ingress critical experiments at the temperatures in which the core operates.

- Axial and azimuthal xenon stability in tall annular cores needs to be validated. The fuel loadings of tall HTGRs are designed to have the highest fissile loadings in the top half of the core so that power is highest at the top of the core with the cooler inlet flow of the down-flow helium coolant and the power is lowest at the bottom where the exiting coolant gas is hottest. The burn-out of fissile uranium-235 and the burn-in of plutonium-239 in the fertile fuel coupled with xenon-135 distribution may tend to push the power toward the bottom of the core at end of cycle with the control rod nearly fully withdrawn and their initial insertion worths at a minimum; however, the thermal capacity of the core with a gas coolant should compensate for any power distribution anomalies. Careful attention to axial fissile and fertile fuel loading distributions and their impact on power distribution over core life is required to assure that the power stays top-peaked, and the normal operating temperature of coated fuel particle at the bottom of the core does not exceed design limits (<1250°C). Azimuthal xenon stability requires attention in the analysis given the possibility of continued operation with a misaligned outer reflector control rod where the misalignment can subsequently be corrected. The analysis needs to determine how much control rod misalignment can be allowed for continued operation assuming the problem can be fixed without shutdown. It may be possible to show that there is no possibility for azimuthal xenon oscillations due to power anomalies caused by a misaligned control that is subsequently restored to normal operation; if that is the case, the limit would be on the resulting peak fuel temperatures both during a cool down event with loss of cooling flow wherein the degree of misalignment would be tied to the maximum operating temperature of the core with a misaligned rod.

The major safety-related reactor physics issue stemming from the pre-application review of the Modular HTGR was that of the safety function of the reactor operator and whether the remote operator station needed to be classified as safety-related. DOE and the reactor vendor had taken the position that the operator had no safety function but the original design of the Modular HTGR called for the control and shutdown rods to be clad with Alloy 800H. Scramming the inner reflector control and shutdown rods, which are needed to achieve cold shutdown but not hot shutdown, was not to be initiated by the plant protection system because the metal-clad could be damaged by high temperatures if a core conduction cool down event ensued after scram. Such damage would likely bow the metal clad rods to the extent that the rods could not be withdrawn to restart the core. Thus the operator had the job of manually actuating the reserve shutdown system if the core condition cool-down event without active cooling started so that boronated graphite balls would be poured into the inner-reflector control rod holes to assure sufficient negative reactivity insertion for cold shutdown. Since the GDC requires the ability to achieve cold shutdown, this gave the operator a safety function and meant any remote operating station would most likely have to be qualified as safety-related. The initial DOE response at a meeting of the Advisory Committee on Reactor Safeguards in 1988 was to declare that they would have to seek a rulemaking change, contrary to their original position. This expected request by DOE for rulemaking would be to allow the Modular HTGR to be exempted from cold shutdown to assure the safety of the public. Such a request would mean that the reactor would go subcritical upon scram of the outer reflector control rods, and then, after ~36 hours for xenon-135 decay during the design basis core conduction cool-down, the core would be allowed to go re-critical at a power level of a fraction of a percent equal to the rate at which heat is conducted from the core. The core would equilibrate to a re-critical temperature consistent with the degree of negativity of the moderator temperature coefficient of reactivity. DOE's initial response would likely not allow for successful rulemaking so subsequently, as part of the NPR Program, DOE invested in initiating the development of carbon-bonded-carbon-fiber control rod clad material so that all rods could be scrammed without regard to the onset of a core-conduction cool down without active cooling. However, even if all control and shutdown rods are scrammed, the operator must still confirm either from the control room or the remote operator station that all rods are inserted and, if not, manually actuate the reserve shutdown system if assurance of cold shutdown has not been accomplished by the plant protection system.

Finally, during the meetings and discussions with NRC staff in which this author participated in the late 1980s, there was concern over the safety classification and the seismic and environmental qualification of the non-safety-related (but perhaps important to safety) cooling water supply and electrical power for the SCS. While the design basis loss-of-forced-cooling accident was to be accommodated by the passive cooling afforded by the core conduction cool-down through the reactor vessel wall to the passive cooling system on the wall of the reactor vessel cavity, there was discussion of the potential need for some intermediate level of quality and surveillance requirements to assure that the safe shutdown cooling system remained available since it was not desirable to have every plant upset lead to a design basis passive cool-down. This issue was never completely resolved.

A related unresolved issue was the reliance on probabilistic risk assessment where there was need for reliability estimates for passive safety systems and for determining event frequencies where some key equipment for preventing design basis events, such as the SCS, both lacked qualification requirements and technical specifications surveillance requirements since such equipment was not to be on the official “primary success path”. Similarly, there was discussion about surveillance requirements on in-vessel components to assure that parameters assumed in the safety analysis remained unchanged. The major item of concern here was the continued assurance that the minimum emissivity value of 0.8 assumed for the inner and outer surfaces of the Alloy 800 core barrel and the inner and out surfaces of the ferritic stainless steel reactor vessel remain unchanged.