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Fort Saint Vrain Gas Cooled Reactor Operational Experience

Oak Ridge National Laboratory

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Fort Saint Vrain Gas Cooled Reactor Operational Experience

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ABSTRACT

Oak Ridge National Laboratory (ORNL) performed an operational events assessment of Fort St. Vrain (FSV) nuclear power plant during 1981–1989. As part of that assessment, the ORNL principal investigator filed 96 monthly reports detailing the results of each month's analysis. This project reviewed those 96 monthly reports and identified pertinent operating experiences that occurred at FSV which may be applicable to current or future gas cooled reactor (GCR) designs.

Two hundred seventy–nine events were catalogued into seven general categories: (1) water incursion events or failures of moisture detection systems; (2) air or other unwanted gas incursion events and failures of gas detection systems; (3) fuel failures or anomalies; (4) failures or cracks in graphite, pipes, and other reactor structural components; (5) failures of nuclear instrumentation systems; (6) human factors and operator performance issues; and (7) other events or conditions that may be relevant to current GCR designs. Several events were reviewed over a period of years and appeared in more than one monthly report.

EXECUTIVE SUMMARY

INTRODUCTION

Much of the resurgent interest in developing new nuclear generating units has focused on newer and enhanced gas-cooled reactor (GCR) designs. Oak Ridge National Laboratory (ORNL) performed continuous operational experience studies of Fort Saint Vrain Nuclear Generating Station (FSV) for the Office for Analysis and Evaluation of Operational Data (AEOD) of the Nuclear Regulatory Commission (NRC) from 1981 through 1989 when FSV was permanently shut down. This study provides an analysis of the operational experience history of FSV as depicted in the monthly FSV operating experience reports produced by ORNL. It identifies operating experiences that may be pertinent to current GCR designs, and assesses the safety significance of events that occurred at FSV.

PROJECT

ORNL filed 96 monthly reports with the AEOD between 1981 and 1989. These monthly reports were reviewed, and the 279 events reported in the set of monthly reports were then catalogued into one of seven categories: (1) 29 water incursion events and failures of moisture detection systems; (2) 2 air or other unwanted gas incursion events and failures of gas detection systems; (3) 3 fuel failures or anomalies; (4) 2 failures or cracks in graphite, pipes, and other reactor structural components; (5) no failures of nuclear instrumentation systems; (6) 47 human factors and operator performance issues; and (7) 196 other events or conditions that may be relevant to current GCR designs. These categories represent only a characterization of the events and do not constitute a root cause determination or final effects and consequences. The systems and components that failed were identified.

Three of the categories, water or gas incursions and other, were further divided into multiple sub-categories. The 29 water incursion events were placed into one of 4 sub-categories as follows: (1) 18 were classified as thermal-hydraulic moisture outgassing events; (2) 4 were determined to be tube leaks; (3) 5 involved moisture detection instrumentation failures; and (4) 2 were a plugging of or an obstruction of process lines.

The category containing human factors and operator performance issues events was divided into six sub-categories distributed as follows: (1) 6 licensed operator error events, (2) 22 testing activity personnel error events, (3) 5 events involving maintenance or repair activity personnel errors, (4) 2 events resulting from installation activity personnel errors, (5) 2 events involving radiation protection activity personnel errors, and (6) 10 events that were "some other activity" personnel errors.

The 196 other events were separated into 7 sub-categories as follows: (1) 13 events related to secondary side systems or issues; (2) 24 events related to electrical distribution systems; (3) 50 events involved instrumentation and control (I&C) systems other than previously listed I&C systems; (4) 39 events related to auxiliary systems; (5) 40 events related to primary reactor systems; (6) 12 events related to waste management items; and (7) 18 events that did not fit into any of the previous sub-categories.

SAFETY SIGNIFICANCE

Moisture Intrusion

The safety consequences from moisture intrusion events at FSV are arguably the single most important issue identified from this review; they directly affect the plant's safety and accident analyses. While the final safety analysis report accident analysis accounts for large moisture incursions, the long-term effects from a small incursion (i.e., low volumetric or inleakage rates) were not clearly understood or appreciated, and ultimately these had a much greater effect on plant operations. Small amounts of moisture degraded both the control rod drive (CRD) and reserve shutdown systems. Moreover, six control rod pairs failed to scram during an event on June 23, 1984. This failure to completely guarantee a plant shutdown when required represented a significant safety hazard for plant operations.

Gas Intrusion

The two instances of helium leaks did not present a safety hazard at FSV.

Fuel Failures

A dropped Lucy Lock onto the top of the core during a refueling outage did not damage the fuel. Another event caused only slightly skewed radial and axial power profiles; however, there was no adverse power peaking. A third event was a failed surveillance on one hopper. However, the system would still have been able to perform its design safety function. Based on these assessments, the three incidents did not present a safety hazard at FSV.

Structural or Graphite Failures

Routine inspections discovered superficially cracked fuel element webs that were not considered a safety issue at FSV. Corroded prestressed concrete reactor vessel tendon wires also did not present any undue safety hazards to FSV.

Human Factors

There were no extraordinary human factors issues uncovered as part of this analysis, and they did not present a safety hazard at FSV.

Other

The potential safety consequences from the 196 other events were collectively representative of routine operational events at FSV. A more detailed analysis of the 196 events may reveal a hidden component or cause that was not apparent from this study.

PERTINENT OPERATING EXPERIENCE

Moisture Intrusion

The greatest hazards came from events that were characterized by the incursion of moisture over a long period of time, usually in small volumetric amounts. The chronic nature of the incursions is something that can be addressed and accounted for in future GCR designs.

Gas Intrusion

With so many connecting interspaces, FSV represented a challenge to preventing helium leaks. The issue of helium leaks (generic or otherwise) would probably be applicable to any new GCR designs. However, precise technical specification limits and strict test and inspection guidelines would be expected to minimize the effect of minor helium leaks.

Fuel Failures

The fuel handling mishap that occurred is neither unique nor applicable to only GCR designs such as FSV. Standard fuel handling operator training, specific and precise fuel handling technical specifications, and extensive refueling operations planning would be expected to prevent this sort of event.

Accidental injection of the shutdown material can be either prevented or mitigated in future designs that may use a design similar to or very much like the reserve shutdown system at FSV. The training and experience gained from achieving a high degree of performance and compliance for emergency safeguards systems in the commercial nuclear industry could also be applied to any new reserve shutdown system designs for future GCRs.

Future GCR designs could fully instrument emergency systems to preclude indication errors or oversights. It is also expected that future designs would improve the design of such systems to the point where accidental injection would be even less of a problem.

Structural or Graphite Failures

Both the PCRV tendon corrosion and the cracked fuel element web issues found at FSV would most likely be applicable for any new GCR design utilizing graphite or a PCRV. However, both issues, as demonstrated at FSV, can be addressed through a rigorous and thorough test and inspection program.

Human Factors

Human error can never be eradicated completely; however, a dedicated and rigorous program to reduce human error can be expected to reduce the human error rate to an acceptably low level. For example, the commercial nuclear industry has a good track record over the past 10–15 years regarding testing and calibration of equipment. The techniques, training, and attention to detail exhibited or practiced in the commercial nuclear industry could be implemented or applied for all future GCR designs.

Other

The nuclear power industry currently has a good record regarding secondary side, auxiliary, electrical distribution, I&C, primary reactor, and waste management systems over the past 10–12 years. The lessons-learned, experience, and training developed throughout the years by the commercial nuclear industry can be applied to any future endeavors or designs for GCRs.

FUTURE WORK

As noted in the introduction, this study provides an analysis of the operational experience history of FSV as recorded in the ORNL monthly reports. Other sources of FSV operational experience including a detailed analysis of the plant's licensee event reports and any operating experience reviews performed for or by the licensee, could yield additional information relevant to future GCR designs.

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LIST OF ACRONYMS

ac	alternating current
ACT	Atomic Energy Act of 1954
AEC	Atomic Energy Commission
AEOD	Office for Analysis and Evaluation of Operational Data
AGR	advanced gas-cooled reactors
BWR	boiling-water reactors
CFR	Code of Federal Regulations
CRD	control rod drive
dc	direct current
DOE	Department of Energy
EPRI	Electric Power Research Institute
FSAR	final safety analysis report
FSV	Fort Saint Vrain Nuclear Generating Station
GA	General Atomics
GCR	gas-cooled reactor
GFR	gas-cooled fast reactor
GT-MHR	gas turbine-modular helium reactor
HTGR	high temperature gas-cooled reactor
HTTR	high temperature test reactor
I&C	instrumentation and control
LANL	Los Alamos National Laboratory
LCO	limiting condition for operation
LER	licensee event report
LWR	light-water reactor
MHTGR	modular high temperature gas-cooled reactor
NPP	nuclear power plant
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
NSIC	Nuclear Safety Information Center
ORNL	Oak Ridge National Laboratory
PB1	Peach Bottom Atomic Power Station 1
PBMR	pebble-bed modular reactor
PCRV	prestressed concrete reactor vessel
PSAR	preliminary safety analysis report
PSC	Public Service Company of Colorado
PWR	pressurized-water reactors
RPV	reactor pressure vessel
SER	safety evaluation report
SG	steam generator
TS	technical specifications
TSUP	technical specifications upgrade program
UFSAR	updated final safety analysis report
Vac	volts alternating current
VHTR	very high temperature reactor

1. INTRODUCTION

1.1 PURPOSE

A resurgence of interest in developing new nuclear generating units, other than light-water reactor (LWR-boiling-water reactors and pressurized-water reactors [BWR & PWR]) concepts, has increased the need to maintain analytical expertise and incorporate operational experience "lessons learned" from previous reactor designs. Much of this recent interest has focused on newer and enhanced gas-cooled reactor (GCR) designs.[1–9] For example, South Africa has developed a revolutionary new concept called the pebble-bed modular reactor (PBMR), based on prior work and reactor testing in Germany, and the People's Republic of China is operating a small test reactor also based on the German design. Based on its previous designs using prismatic fuel, General Atomics (GA)[*] has designed a gas turbine–modular helium reactor (GT-MHR); and the U.S. Department of Energy (DOE) has proposed other GCRs as part of its Generation IV reactor concepts, including the very-high-temperature reactor (VHTR), which is a variation on Japan's high-temperature test reactor (HTTR) and France's gas-cooled fast reactor (GFR). Reviewing lessons learned, previously gained insights, experience, and history from previous work is necessary to proceed with the future design, licensing, and construction of the next generation GCRs. As part of the historical insight and experience, two licensed commercial high-temperature GCRs (HTGRs) have operated in the United States—Peach Bottom Atomic Power Station 1 (PB1) and Fort Saint Vrain Nuclear Generating Station (FSV). One aspect of preserving and continuing the expertise for HTGRs is to recover some of the knowledge and understanding acquired during the operation of these plants. Oak Ridge National Laboratory (ORNL) performed continuous operational experience studies of FSV from 1981 through 1989 when FSV was permanently shut down. Therefore, since information regarding FSV is readily available and the expertise regarding FSV operations still exists at ORNL, it was decided to focus on FSV. The purpose of this study, then, is to collect and maintain the knowledge and expertise acquired during FSV's operation for use in addressing potential safety concerns that may apply to current GCR designs. Please note: This report is limited only to ORNL's operational experience studies of FSV and is not intended to be a comprehensive study of events at FSV.

1.2 SCOPE

This study provides an analysis of the operational experience history of FSV as depicted in the monthly FSV operating experience reports produced by ORNL. This report identifies FSV operating experience information that may be pertinent to current GCR designs and assesses the safety significance of events that occurred at FSV as these events apply to current designs. The events are separated into seven categories: (1) water incursion events, (2) air or other gas incursion events, (3) fuel failures, (4) structural failures, (5) nuclear instrumentation failures, (6) human factors issues, and (7) other events. The systems and components that failed are described, and the significance of the potential safety consequences concerning current GCR designs are developed.

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[*]The General Atomic Division of General Dynamics Corporation developed an HTGR concept in 1958.[10] This division was a forerunner of the Gulf General Atomic Company. Gulf General Atomic Company was a division of Gulf Energy and Environmental Systems, Incorporated, which was a subsidiary of Gulf Oil Corporation. Gulf General Atomic Company was also known as Gulf General Atomic Incorporated or General Atomic Company.[11] Any time any of these companies is referenced throughout this report, unless otherwise specified, the acronym GA will be used.

2. HISTORY

2.1 LICENSING HISTORY

FSV was a commercial nuclear power plant (NPP) owned and operated by Public Service Company of Colorado (PSC). FSV was granted an operating license by the Atomic Energy Commission (AEC) on December 21, 1973;[12] had its initial criticality on January 31, 1974;[12] and went into commercial operation on July 1, 1979.[12] FSV remained in commercial operation for a little more than 10 years. Then, on August 18, 1989, the plant was shut down to repair a stuck control rod pair; however, during the shutdown, numerous cracks were discovered in several steam generator (SG) main steam ringheaders. The required repairs were determined by the PSC board of directors to be too extensive to justify continued operation, so they decided to permanently terminate nuclear operations on August 29, 1989.[12]

2.2 REACTOR DESCRIPTION

The reactor at FSV was helium cooled, graphite moderated, and utilized a ^{235}U- thorium fuel cycle. The reactor's design employed many of the same fundamental principles that formed the basis for the prototype HTGR at Peach Bottom, Pennsylvania.[11] The prototype began supplying power to the Philadelphia Electric Company system in March 1967, commenced commercial operation in June 1967, and ceased operation in October 1974.[11] However, PB1 and FSV were very different. The most significant differences were that (1) PB1 had a 40-MWe power rating versus a 330-MWe power rating for FSV; (2) PB1 used a steel reactor vessel, and FSV had a prestressed concrete reactor vessel (PCRV); (3) PB1 used long, annular fuel elements with a solid graphite spine, and FSV used prismatic-block graphite fuel elements; and (4) PB1 used electric-motor–driven helium circulators with oil-lubricated bearings, and FSV used steam and water turbine-driven–helium circulators with water-lubricated bearings.

As noted in the FSV *Updated Final Safety Analysis Report* (UFSAR),[11] heat for the nuclear steam supply system was produced by fission from the uranium-thorium fuel cycle. Graphite was used for the moderator, fuel cladding, core and core support structure, and reflector. High-temperature helium was used as the primary coolant to produce superheated and reheated steam at approximately 1000°F. The active core was composed of 1482 hexagonal graphite fuel elements stacked in 247 vertical columns. The core was arranged in 37 individual flow-controlled core regions, and each region had a pair of control rods. The individual graphite fuel elements were approximately 14 in. across the flats and 31 in. high. Vertical coolant holes within each element were aligned with the coolant holes in the fuel elements above and below it by using dowels in the fuel blocks. The fuel was in the form of particles made from a mixture of the carbides of thorium and uranium. These particles were then coated with highly retentive coatings of pyrolytic carbon and silicon carbide. The fuel particles and a carbonaceous matrix formed bonded fuel rods. The reactor had two primary coolant loops, each consisting of a six-module SG, an SG outlet plenum, and two helium circulators. The circulators for each loop discharged into a common plenum below the core support floor. All of the helium flow passed upward around the core support floor and the core barrel to the core inlet plenum located above the reactor core. The helium coolant, at a pressure of 700 psia, then flowed downward through the reactor core where it was heated to a mean temperature of about 1430°F. The helium was then directed to the SGs beneath the reactor core. After passing through the SGs, the coolant was returned to the reactor at about 760°F by four steam-turbine-driven circulators, which operated on exhaust steam from the main turbine.

2.3 LICENSE OPERATIONAL PROVISIONS

PSC applied for a construction permit and Class 104 License for FSV with the AEC in October 1966.[10] Further, PSC was granted an operating license for FSV in accordance with Section 104 (b) of the Atomic Energy Act of 1954 (hereinafter referred to as the Act).[10, 11, & 13] The provisions of the Act allowed PSC ample leeway in its operation of FSV, and in fact, FSV was operated differently from other commercial NPPs that were in operation at the same time. The Act allowed the AEC, "In issuing licenses under this subsection, the Commission shall impose the minimum amount of such regulations and terms of license as will permit the Commission to fulfill its obligations under this Act...".[13] A more extensive treatment of this issue may be found in Appendix A of this report.

2.4 FSV OPERATIONAL EXPERIENCE REPORTS (1981–1989)

More than a year after FSV started commercial operation, the Office for Analysis and Evaluation of Operational Data (AEOD) of the Nuclear Regulatory Commission (NRC) "established a program for the analysis and evaluation of operational experience for FSV...".[14] Reference 14 also stipulated that licensee event reports (LERs) from FSV received after September 21, 1981, at the ORNL Nuclear Safety Information Center (NSIC) would be screened in accordance with AEOD procedures. According to AEOD Procedure No. 3, LERs were screened and assigned to one of four categories depending on their safety significance.[14, 15] Category I events were those requiring immediate action to ensure plant safety, Category II events were judged to be safety significant but not warranting immediate action, Category III events required additional review, and Category IV events were events with little or no safety significance. Generally, ORNL would analyze events that were classified as Category I, II, or III or events that were not filed as LERs but that NRC thought warranted further analysis and review. In addition, FSV monthly operating reports, FSV NRC inspection reports, Electric Power Research Institute (EPRI) reports or technical analyses regarding FSV, or any other technical reports regarding operational events at FSV were also reviewed. Category IV events would only be abstracted and indexed by NSIC. From October 1981 until September 1989, monthly reports from NSIC to AEOD were filed providing reviews of FSV operating experiences (e.g., LERs), inspection reports, significant engineering evaluations, industry reports, etc.

2.4.1 LER Screening and Classification

As noted above, LERs concerning FSV were classified by NSIC into one of four categories. LERs classified as Category I, II, or III were individually analyzed by NSIC. The detailed analysis for each of these events was reported each month to the AEOD. Further, an assessment was made by NSIC concerning the severity of the event and the potential impact the event had on FSV operations. For example, the radiological impact of an event may have been reported as minor and had no effect on plant operations, thus the event required no further analysis. However, some events may have required continuous tracking by the NSIC investigator, in which case the event analysis was updated in each successive monthly report after the initial report until the event analysis was resolved or completed.

For those events classified as Category IV events, NSIC wrote an abstract for the event and indexed the event in accordance with NSIC procedures and retained the indexing for future reference. After that, no further analysis or action was taken regarding Category IV LERs unless directed by the NRC technical monitor.

2.4.2 Other Event Screening and Classification

As noted earlier, many other sources of information that may have pertained to or affected FSV plant operations were reviewed. The licensee submitted monthly operating reports to the NRC which were subsequently reviewed by AEOD, as well as NRC inspection reports filed by the NRC FSV resident inspector. Both of these types of reports were regularly reviewed by the NSIC investigator. Frequently, an event would not meet the minimum reporting requirements for generating an LER; however, the event would have some meaning or importance relating to plant operations. In that case, these types of events would then also be reviewed by the NSIC investigator. His ensuing analysis classification would subsequently be reported to the AEOD in the next monthly report.

As part of the project, all LERs submitted to NSIC prior to September 21, 1981, going back to 1972, were given a cursory review during the initial phases of the program (MR1081). Also, the NSIC investigator reviewed existing documents regarding FSV. These documents included the DOE-sponsored FSV Inspection Program Plan (MR1081) and EPRI's FSV Experience Reports (MR1081). These latter reports covered problems encountered during the plant's post-construction, startup testing, power ascension, initial operation, and first refueling outage (MR1081). Other items initially reviewed by the NSIC investigator were foreign reactor events (MR1081) and FSV annual operating reports (MR1181).

The screening program was expanded in December 1981 to include a brief review of all FSV docketed documents routed to NSIC (MR1281). Further reviews included DOE-sponsored engineering analyses on the Research and Development Program and Surveillance and Testing Program at FSV; these reviews were subsequently classified by the NSIC investigator (MR0182). NRC technical reports were assessed and reported on as well (MR0182). Operational data and the LER screening process were discussed during an FSV site visit and an NRC Region IV visit (see MR0282). This established a "...direct access to NRC personnel with firsthand information on plant operations and performance." (MR0282) The visits enabled further clarification of the LER analysis, performance assessment, and subsequent evaluation. License change submittals were also reviewed (MR0382). In some instances, an event was initially reviewed at the request of AEOD prior to the issuance of an LER, which was further reviewed when the LER was submitted (MR0582 and MR0682). This process remained effective and in place until the plant permanently shutdown.

3. PROJECT DESCRIPTION

3.1 FSV OPERATIONAL EXPERIENCE REPORTS ANALYSIS

This report identifies past operating experience at FSV that may be applicable to current GCR designs. Specifically, this document is a report on the analysis and review of the 96 monthly reports filed with the AEOD by NSIC between 1981 and 1989 as part of an NRC Project, Financial Identification Number B1661. After each monthly report was reviewed, all the events reported on in each monthly report were then catalogued into one of seven categories. Each individual event was analyzed as was each category of similar events. The sorting of events is described in the following sections. The events, as well as each category of events, were also analyzed for their applicability to current GCR designs.

3.1.1 Monthly Report Screening

Each of the 96 monthly reports was reviewed and divided into separate or individual events. An event may have been composed of or encompassed several diverse sources, such as LERs, meeting notes, inspection reports, etc. Typically, an event occurred on a specific day; however, the sources of information regarding the event could span several days or even months. In this latter case, the event was "tagged" with its inception date and tracked until the event was resolved or studies about it were completed.

The monthly reports also reported on general or routine occurrences concerning the project. Items such as project costs, expenditures, additional documentation, site visit reports, reports on general telephone conversations, etc. were considered routine reporting. No further analysis or review of these items or routine reports, along with the general "boilerplate" of the monthly reports, was performed unless they specifically pertained to an event.

3.1.2 Event Selection

In some cases an event was reported and classified as a Category IV event (see Section 2.4), but "...the nature of the event required some discussion and clarification..." (MR0482). Sometimes the AEOD requested a preliminary analysis and study before formal documentation was issued or received (MR0382). In both these types of cases, and whenever no clear event identifier was indicated but an event occurred nonetheless, the events were noted and categorized according to the present screening criteria and then analyzed as part of the report.

The analysis of an event found in Section 4 of this report, in every case, denotes all the original sources of information that produced the event and further indicates in which monthly report these sources were cited.

3.1.3 Event Classification

After review and analysis, each event was classified into one of seven categories: (1) water incursion events and failures of moisture detection systems; (2) air or other unwanted gas incursion events and failures of gas detection systems; (3) fuel failures or anomalies; (4) failures or cracks in graphite, pipes, and other reactor structural components; (5) failures of nuclear instrumentation systems; (6) human factors and operator performance issues; and (7) other events or conditions that may be relevant to current

GCR designs. These categories represent only a characterization of the events and not any root cause determination or final effects and consequences.

A number of events were easily and directly classified with very little confusion or conflicts. For example, the SG penetration helium leak that occurred on October 27, 1981 (MR0182, MR0282, and MR0382) was classified in the second category as an air or other unwanted gas incursion event. In this case the incursion gas was helium. However, when a core restraint device called a "Lucy Lock" was dropped onto the top plenum element on November 24, 1981, because of a logic error in the computer controlling the fuel handling machine (MR1281 and MR0282), the classification became more uncertain. Although the device did not directly impact the fuel blocks, initially there was concern for fuel element damage, which conceivably could make this a category 3 event (fuel failures or anomalies); however, the event could also be a category 7 event (other events or conditions relevant to current GCR designs) or a category 4 event (failures or cracks in graphite, pipes, and other reactor structural components). In this case, based on the information provided in two monthly reports, it was evident that this event should be placed into category 3 (fuel failures or anomalies). Another event reported on in November 1982 (MR1082) involved cracked fuel element webs. At first, this event could seemingly be classified as a fuel failure (category 3) or as a structural graphite failure (category 4). Early evidence indicated that an improper dowel fit was suspected of having created high enough stresses to have caused the failures; therefore, it was decided the most appropriate classification would be category 4.

Whenever it was possible to determine the cause of an event based only on the information presented in the monthly report, it was decided to classify the event according to that cause; otherwise, engineering judgment was used to assign a category.

3.2 FSV OPERATIONAL EXPERIENCE REPORTS TABULATION

Each of the 96 monthly operating experience reports from 1981–1989 for the AEOD project was scanned into a *.pdf formatted document suitable for viewing by anyone with a personal computer equipped with the Adobe Acrobat Reader ® software. Also, each monthly report was also scanned into a text-searchable format file.

4. RESULTS

Two hundred seventy–nine events were catalogued into seven general categories: (1) water incursion events or failures of moisture detection systems; (2) air or other unwanted gas incursion events and failures of gas detection systems; (3) fuel failures or anomalies; (4) failures or cracks in graphite, pipes, and other reactor structural components; (5) failures of nuclear instrumentation systems; (6) human factors and operator performance issues; and (7) other events or conditions that may be relevant to current GCR designs.

The distribution of the 279 events among the 7 categories was as follows: (1) 29 events were either a water incursion event or a failure of a moisture detection system; (2) there were 2 air or other unwanted gas incursion events; (3) there were 3 fuel failures or anomalies; (4) 2 events involved failures or cracks in graphite, pipes, or other reactor structural components; (5) there were no failures of nuclear instrumentation systems; (6) 47 events were classified as a human factors or operator performance issue; and (7) 196 other events or conditions may be relevant to current GCR designs but did not fit into any of the previous 6 categories.

Three of the categories were further divided into multiple sub-categories. The 29 water incursion or failure of moisture detection systems events were placed into one of 4 sub-categories as follows: (1) 18 were classified as thermal-hydraulic moisture outgassing events; (2) 4 were determined to be tube leaks; (3) 5 involved moisture detection instrumentation failures; and (4) 2 were a plugging of or an obstruction of process lines.

The category containing human factors and operator performance issues events was divided into six sub-categories. The 6 sub-categories and the distribution of events among them is as follows: (1) 6 licensed operator error events, (2) 22 personnel testing activity error events, (3) 5 events involving personnel maintenance or repair activity errors, (4) 2 events resulting from installation activity personnel errors, (5) 2 events involving radiation protection activity personnel errors, and (6) 10 events that were "some other activity" personnel errors.

The 196 other events not placed into the first 6 categories were separated into 7 sub-categories as follows: (1) 13 events were related to secondary side systems or issues; (2) 24 events were related to electrical distribution systems; (3) 50 events involved instrumentation and control (I&C) systems other than previously listed I&C systems; (4) 39 events related to auxiliary systems or issues; (5) 40 events related to primary reactor systems; (6) 12 events related to waste management items; and (7) 18 events that did not fit into any of the previous sub-categories were grouped by themselves.

4.1 WATER INCURSION EVENTS AND FAILURES OF MOISTURE DETECTION SYSTEMS

4.1.1 Summary of Monthly Operational Experience Reports

Early GCR designs indicated a concern with moisture intrusion into or a contamination of the helium gas that was going to be used as the primary coolant—"One problem of concern in the design of this reactor is contamination with water vapor because water will react rapidly with the carbon in the fuel-moderator assemblies."[16] It was thought that the "...carbon might be preferentially attacked and lead to increases in the permeability of the graphite."[16] In later designs, this concern was extended to include two additional problems associated with moisture in the helium primary coolant, especially since they pertained to FSV.

First, hydrolysis of the fuel particle coating of pyrolytic carbon (or pyrocarbon) and silicon carbide can lead to fuel failure and subsequent release of fission products. The FSV UFSAR[11] indicates that

> "The fissile particles contain both thorium and U-235 (93.15% enriched) and the fertile particles contain only thorium. These fuel particles have a four layered TRISO coating. The inner layer is a porous pyrolytic carbon, referred to as a buffer layer. The next layer is a high density isotropic pyrocarbon (or pyrolytic carbon). A thin layer of silicon carbide, which is highly impervious to metallic fission products, is deposited outside the inner isotropic pyrocarbon layer. The outermost layer is a strong high density isotropic pyrocarbon." The porous "buffer" layer stops the recoil of fission fragments from impinging on the inner layer of high-density pyrocarbon. The high-density pyrocarbon layers resist the leakage of gaseous fission products.

Second, the moisture can cause corrosion of the PGX graphite core support post. The FSV UFSAR[11] describes the design of the core support structure.

> "The core support structure consists of a graphite and composite metal and concrete structure under the core... . Its function is to support and laterally restrain the fuel and reflector elements and to direct the helium coolant flow to and from the core. ...Each of the 37 refueling regions of the core is supported and located on a graphite core support block of hexagonal cross section... . The graphite used in the core support blocks is PGX graphite. The core support posts are ATJ graphite and the permanent side reflectors are HLM graphite. The oxidation rates for PGX graphite used for the core support blocks were found to be higher than the rates for the H-327 and H-451 graphite used for the fuel blocks."

There are other sources of moisture besides contamination of the helium. One source—especially prevalent at FSV—is the moisture outgassing that occurs when the graphite is heated up and a so-called "drying out" of the graphite takes place.* Another source of moisture at FSV, according to MR0282, is "...moisture ingress from circulator bearing water. This source of moisture ingress is a generic problem at Fort Saint Vrain." The PCRV insulation (Kaowool) will also out-gas moisture that was entrained in the insulation during shutdown, following the heatup of the helium coolant. The PCRV liner is another potential source of moisture during startup.

There were 29 events related to either a moisture incursion or a failure of a moisture detection system. The distribution of events occurring in any one year was nearly bell-shaped with the peak in 1983, when nine occurrences were reported in the monthly reports. It should be noted that 76% of all the moisture incursions or failures of the moisture detection systems occurred between 1982 and 1985.

The events related to moisture intrusion were classed into four general categories: (1) thermal-hydraulic moisture outgassing, (2) tube leaks, (3) moisture detection instrumentation failures, and (4) plugging or obstruction of process lines. Thermal-hydraulic moisture outgassing events far outnumbered all the other events combined (i.e., more than half the events were in this category). Eighteen of the 29 events were

*Graphite has an initial water content (tons in a new core) that has to be "baked out" during startup.

included in this category. Five events were counted in the category moisture detection instrumentation failures; four events were tube failures; and two events were attributed to process line plugging or obstructions.

4.1.1.1 Systems and Components That Failed

A. Two of the four tube leak events occurred in 1982 one in May and the other November. MR0582 reviewed a report the licensee filed in accordance with *The Code of Federal Regulations,* Title 10—Energy, Part 50, Section 73, "Licensee Event Report System" (10 CFR 50.73) that became LER 50-267/82-014. LER 50-267/82-014 reported a crack in a weld on a PCRV liner cooling tube. Thermal expansion causes small leaks in welds to seal up when they are exposed to high power, as reported in MR0682. This phenomenon was expected to occur for the crack that was reported in LER 50-267/82-014. Almost 2 years later, MR0584 reported that the crack had been sealed.

   MR1182 reported a moisture intrusion during a surveillance test. A 2% power limit was imposed and the SG subheader was plugged. LER 50-267/82-049 reported the SG tube leak in Module B-2-3; this LER was reviewed as part of MR1282. MR0283 reported, "The leak poses no serious safety implications..." Nearly one year later, in MR1083, a GA Technologies report on the examination of the tubes from Module B-2-3 was reviewed. MR0283 reviewed LER 50-267/82-048 regarding a primary system leak into the purification cooling water in the heat exchanger for the "B" purification train cooler. Since the heat exchanger was scheduled to be replaced during the next refueling outage in September 1983, the SG tube leak reported in LER 50-267/82-049 would no longer be expected to be a problem.

   The final tube leak report was filed by the licensee in accordance with *The Code of Federal Regulations,* Title 10—Energy, Part 50, Section 72, "Immediate Notification Requirements for Operating Nuclear Power Reactors" (10 CFR 50.72) in November 1988. The event was reviewed as part of MR1188, which reported a leaking cooling tube on the enclosure liner of the core support floor. By the end of November 1988, it was reported that 1000 gal of water resulting from the leak had been removed from the reactor primary system.

B. Both process line plugging events occurred in 1985. The monthly report for May 1985 (MR0585) identified a licensee report filed in accordance with 10 CFR 50.72 on May 3, 1985, regarding a plugged helium pressurizing line. The helium pressurizing line was connected to the refueling interspace. The second event was first noted in MR0785. LER 50-267/85-007 was reviewed as part of MR0785 and reported several plugged helium pressurizing lines that were connected to the control rod drive mechanisms located in refueling penetrations. Corrosion caused by moisture in contact with the carbon steel piping collected at the interface between the 3/4 in. supply line and 1/8 in. inlet line. The plugged lines were cleaned and returned to service.

C. Four of the five instances reporting moisture detection instrument failures were reported in 1983. The remaining event was reported in 1989 in LER 50-267/89-001. MR0189 identified the 10 CFR 50.72 report filed by the licensee regarding an event that occurred as the result of a spike (or instrument noise) in a moisture protection circuit. MR0389 reported that the LER was actually two instances that occurred when the plant was already shutdown and "...in a condition of high levels of primary coolant moisture..."

Three of the four events that occurred in 1983 were the result of exceeding either minor moisture detection instrument limits or dew point limits, which resulted in technical specifications limiting condition for operation (TS LCO) violations. For example, MR0583 reviewed six dew point violations that were reported in LER 50-267/83-012. Similarly, MR0383 and MR0983 reported on moisture limit violations reported in LER 50-267/83-003 and LER 50-267/83-025, respectively.

The last moisture detection instrument problem was reported in LER 50-267/83-019 and was reviewed as part of MR0783. The event concerned errors in primary system moisture level indication.

D. As noted, moisture can be entrained in the primary coolant helium gas from a variety of sources, from the normal "drying out" process for graphite during ascension to power following a shutdown, to the moisture incursion following a circulator bearing water imbalance.

The first moisture incursion event reported, as part of this program was in MR1081 as part of a review of LER 50-267/81-049. MR1081 indicated that "Fort St. Vrain is subject to moisture outgassing from the core graphite during rise to power after refueling, and there are spikes observed in the oxidant levels (i.e., H_2O, CO_2, CO) during such outgassing." Two events were reported in 1982, one in MR0182 and the other in MR0382.

In 1983, one event was covered by two LERs (50-267/83-006 and 50-267/83-007) and reported in MR0483. A moisture ingress event was reported after a manual scram. A main steam depressurization valve failed open, causing a moisture intrusion, as reported in MR1183. It was noted in the following monthly report, MR1283, that a moisture detection instrument had failed, allowing the moisture intrusion. In 1984, a helium circulator tripped, causing a loop shutdown and a subsequent moisture incursion that was reported in MR0384. Excess moisture that accumulated during a refueling outage was reported in MR0484. An extended dryout during startup following the refueling outage was necessary to get rid of the water. Two other events were reported in MR0784. LER 50-267/84-006 was a high moisture loop shutdown followed by a scram, and LER 50-267/84-007 was a loop shutdown on the loss of helium circulator bearing water indication.

LER 50-267/85-012, reviewed as part of MR0985, reported a backup water seal failure causing a moisture intrusion that led to a loop shutdown and subsequent scram. MR1085 reported a loop SG penetration interspace low-pressure moisture ingress that was described in LER 50-267/85-013. The senior resident inspector's report reviewed as part of MR0486 reported a moisture ingress while the plant was shutdown.

Three events that occurred in 1987 and 1988 all reported moisture intrusions after a helium circulator tripped (see MR0787 and MR0588).

E. One moisture intrusion event requires a detailed examination because of the important ramifications and potential consequences arising from the event. The monthly report for June 1984, MR0684, reported that a reactor scram had occurred on June 23, 1984; however, six control rod pairs failed to fully insert in response to the scram signal. The licensee filed a report, LER 50-267/84-008, in accordance with 10 CFR 50.73. The LER underwent three revisions, and the licensee filed many reports regarding tests, examinations, and analyses conducted concerning this event. Portions of more than 17 monthly reports spanning more than 4 years were devoted to this event.

Listing the sequence of events leading up to the failure to scram is necessary to understand the importance of this event. The following sequence of events is compiled from several monthly report as well as the three revisions of the LER itself. Please note that all times are approximate.

The plant was operating at 50% power on June 22, 1984, at 1404 h when an auxiliary transformer sudden-pressure (or rapid-rise pressure) relay tripped. This caused a temporary loss of a 480 Vac essential bus, which in turn tripped the normal bearing water supply for the A and B helium circulators (Loop 1). When the backup bearing water supply came on line, there was a surge or upset in the buffer helium system. The preliminary moisture indication for the primary system was greater than 100 ppm. As a result of the buffer helium system upset, the A helium circulator tripped off. The operators experienced difficulty in setting the shutdown seal on the circulator. At this time, the indicated primary system moisture content was 40–70 ppm. The operators then decreased reactor power to 30%. The LER indicated, "Although moisture ingress was evident, the source was known to be characteristically finite and not considered to present the possibility for increasing PCRV pressure above the setpoint for the PCRV safety valves." The operators placed the interlock sequence switch into the "low power" position. At this time, several minutes into the transient, the moisture content of the primary system was noted to be increasing; and power had increased 40% as a result of the positive reactivity addition from the cooldown of the primary system. The A helium circulator was returned to service about 2 hours into the event. At about 2000 h, the plant was continuing to operate with full circulator operation in effect; however, the operators began a plant shutdown because of the elevated levels of moisture in the primary system. The excessive moisture had caused the operating helium purification train to "ice-up (i.e., sufficient moisture had bypassed the chiller that is used to precipitate water from the helium purification system so that ice formed on the liquid-nitrogen-cooled krypton trap downstream of the chiller.) " The LER indicates that

> "the icing of the helium purification system was not abnormal under the observed high moisture condition. Prior to the trip, the high pressure setpoint was decreasing as programmed with circulator inlet temperature (indicating a power reduction). When the depressurization flow path was blocked, the trip point was exceeded as temperature continued to decrease without a corresponding decrease in pressure."

The plant was at 30% power when the main turbine was tripped at approximately 2144 h. By 0029 h on June 23, 1984, power had decreased to about 23% when the plant experienced a scram on high vessel pressure. The operators first verified the reactor was subcritical; however, they also noted that six control rod pairs had failed to insert. MR0684 reported that the operators immediately attempted to input a manual scram signal, which also failed to insert the six control rod pairs. The operators then pulled the fuses for the scram breakers for the six control rod pairs, but that attempt also failed to insert the control rods. The operators reinserted the fuses and re-energized the control rod drive (CRD) motors. The control rods were then fully inserted using the CRD drive motors about 20 min after the initial automatic scram signal (approximately 0029 h). The licensee initially, in LER 50-267/84-008, Rev. 1, "...believed the six control rod insertion failures resulted from the migration of moisture from the PCRV into the CRD motor area" and subsequently declared, "It has been determined that the migration of moisture from the PCRV into the CRD motor area would not have been prevented even under full design purge flow conditions." It was noted in MR0684 that immediately following the event, the licensee inspected the CRDs and found that most of the stainless steel surfaces were clean and free of dirt; however, many carbon steel surfaces showed evidence of rust and corrosion.

MR0884 noted that a scram time of 2 minutes was acceptable based on the analyses done as part of the Chapter 14 accident analysis of the FSV FSAR; however, the principal investigator for the project also noted, "A key concern is that no common mechanism should cause failures to both the control rod and reserve shutdown systems." In addition to this concern, the principal investigator also noted in MR0884 that the licensee had reported a similar failure-to-scram event in 1982 (see MR0482) that was subsequently reported in LER 50-267/82-007. That event was ultimately attributed to some moisture collecting on the control rods. On February 22, 1982, the reactor was subcritical and the plant was undergoing a startup evolution. The operators inserted a manual scram in accordance with a loop impurity TS LCO; however, two control rods failed to scram, and the operators had to use the control rod drive motors to insert the binding or sticking CRDs. The licensee determined that the sticking or binding of the CRD mechanisms could have been caused by primary coolant system contamination and moisture. The similarity between the two events (LER 50-267/82-007 and LER 50-267/84-008) prompted the principal investigator to indicate,

> "Another concern arises in the case of a partial scram is that other conditions may also be affecting core reactivity. Since moisture appears to play some role in the cause of the two failure-to-scram events which have now occurred, the effect of moisture on core reactivity should at least be noted. The introduction of water or steam into the HTGR has three effects. Water can cool the core causing a positive reactivity effect through the negative temperature coefficient. Water or steam increases the slowing down of neutron past the resonance absorbers and thereby reduces neutron leakage both of which have a positive reactivity effect. But water also increases thermal neutron captures through parasitic absorption in hydrogen. If too much water enters the core, reactivity effect becomes negative due to the parasitic captures."

The principal investigator then summed up his concerns in MR0884,: "Therefore, in the worst case scenario for a partial scram during a major water ingress, water and steam may contribute to keeping the core critical but would probably also contribute to slowing or turning core heatup until the reserve shutdown system could be actuated by operator action."

Thus, a preliminary examination and analysis of the event soon after it had occurred clearly indicated that a long-term effect had developed and could possibly affect future operations requiring a scram. As part of the corrective actions taken by the licensee in response to NRC concerns, several studies were conducted on the CRDs and orificing assemblies that included an operational evaluation, a refurbishing program, a preventative maintenance program, a failure-to-scram report and replacement of all CRD stainless steel cables with Inconel 625 cables.

It was reported in MR0984 that during a follow-up test of the CRDs, a cable for a control rod pair had broken and was jammed in its guide tube. MR1084 indicated that more than 300 gal of water were removed from the primary coolant system when reactor pressure was raised above 100 psi.

MR1284 reported that, as part of the follow-up corrective actions associated with LER 50-267/84-008, the licensee conducted tests of the reserve shutdown system hoppers in November 1984 and discovered that several hoppers had failed to discharge their reserve shutdown material (see MR1284 and Section 4.3 of this report). MR1284 also reviewed the first revision of LER 50-267/84-008. The principal investigator noted, "...the plant was being operated with possibly both the primary and secondary shutdown systems simultaneously in a degraded condition." He further noted "The failures may be related in one way or another to water ingress events which have been of relatively low inleakage rates in the past."

MR0385 noted that the licensee had implemented procedures directing operator actions following the loss of the helium purge flow to the CRDs. The following month, MR0485 reviewed the second revision of LER 50-267/84-008 and examined the potential consequences should water in sample lines cause an erroneous indication of moisture content in the primary coolant system and how such a situation may have contributed to the events depicted in the LER.

Over the next several months, the monthly reports conducted various reviews of different reports and documents. For instance, MR0685 reviewed the licensee's CRD cable replacement report, and MR1185 reviewed the licensee's proposed interim TS. The latter review specifically focused on two issues pertaining to the failure-to-scram event. The first was the TS relating to the moisture monitoring instrumentation and associated sample lines, and the second was the TS concerning the helium purification trains and the potential for icing of the low-temperature adsorber. MR0286 reviewed additional licensee reports, and MR1186 addressed reactor physics issues related to LER 50-267/84-008 versus advanced reactor safety studies. At the end of 1986 and the beginning of 1987, the monthly reports reviewed safety evaluations that pertained to this event. MR1286 reviewed the safety evaluation of chlorides in the primary coolant system; MR0187 reviewed TS changes regarding the steam line rupture detection system; MR0387 reviewed the licensee's inservice inspection and test program description; and MR0887 analyzed the licensee's proposed plant changes that were filed in an annual report in accordance with *The Code of Federal Regulations* Title 10—Energy, Part 50, Section 59, "Changes, Tests, and Experiments" (10 CFR 50.59). MR0887 was the last monthly report that examined any aspects of this failure-to-scram event.

4.1.1.2 Potential Safety Consequences

The safety consequences from moisture intrusion events at FSV are arguably the single most important issue identified from this review; they directly affect the plant's safety and accident analyses. As indicated in MR0884, a large moisture incursion was accounted for in the FSAR accident analysis; however, the long-term effects from a small incursion (i.e., low volumetric or inleakage rates) were not clearly understood or appreciated, and ultimately these had a much greater effect on plant operations. The two most important systems required for immediate plant shutdown (the CRD and reserve shutdown system) were seriously degraded. The CRD system had been undergoing degradation due to slight amounts of moisture for more than 2 years. Moreover, the CRD system failed to fully perform its intended design function at a time when it was most needed. The reserve shutdown system was also degraded over a period of time that probably exceeded 2 years. The degradation of the reserve shutdown system was traced back and was also shown to result from small amounts of moisture coming in contact with the shutdown material. This failure to completely guarantee a plant shutdown when required represents a significant safety hazard for plant operations.

4.1.2 FSV Operating Experience Pertaining to New Gas-Cooled Reactor Designs

The nuclear power industry currently has an excellent record regarding emergency shutdown systems over the past 10–12 years, and their performance is unmatched by any other industry for reliability. The lessons learned, experience, and training developed throughout the years by the commercial nuclear industry could be applied to any future endeavors or designs for GCRs. The events previously discussed that represented the greatest hazards were almost all characterized by the incursion of moisture over a long period of time and usually in small volumetric amounts. That is, small leaks over long periods did more damage and posed a greater risk to safety than any large water incursions experienced in short periods of time. This crucial item that the leaks all shared, the chronic nature of the incursion, is something that can be addressed and accounted for in future GCR designs. Another design flaw that could be corrected in future GCR designs, as noted in LER 50-267/84-008, is the helium purge flow system's inability to prevent the primary coolant from migrating to the CRD areas where entrained moisture or contaminants could degrade the CRDs.

4.2 AIR OR OTHER UNWANTED GAS INCURSION EVENTS AND FAILURES OF GAS DETECTION SYSTEMS

4.2.1 Summary of Monthly Operational Experience Reports

There were two events involving either an air or other unwanted gas incursion or a failure of the gas detection systems. However, neither event occurred because of a failure of the gas detection systems, nor did they involve an air incursion. Rather, both events were due to a leakage of helium gas. The first event was initially reported and reviewed in MR1181. The event was noted in a report filed in accordance with 10 CFR 50.72. Another 10 CFR 50.72 report that was reviewed as part of MR0787 described the second occurrence. These events are presented in more detail below.

4.2.1.1 Systems and Components That Failed

A. As discussed in MR0182, the licensee detected a SG penetration helium gas leak on June 4, 1980, and subsequently reported the leak to the NRC in accordance with 10 CFR 50.73 via LER No. 50-267/80-030 as part of a letter from the PSC (DCN 8006170692) on June 5, 1980.

   The licensee sought exemptions to its plant TS regarding allowable leakage to the penetration closure interspace. MR0182 reported that the NRC granted these exemptions provided that the current licensee administrative controls remained in effect. Those licensee administrative controls included an increase in the frequency of surveillance and a commitment by the licensee to investigate the source of the leak and make the necessary repairs during the upcoming refueling outage. Following the refueling outage, the licensee found an isolation valve in the purified helium pressurization system had been inadvertently closed. The valve had been closed for more than 5 days and was subsequently reopened. This event was reported in LER No. 50-267/81-067. After this valve had been opened, a helium leak was noticed on October 24, 1981, in SG Module B-2-3 for the Loop 2 SG. This leak was initially reported in accordance with 10 CFR 50.72 and subsequently reviewed as part of MR1181. The penetration leakage to the cold reheat section of the SG was thought to be the same leak that was reported in LER No. 50-267/80-030 (see MR0182).

MR1181 indicated that "...for a small leak, the event does not appear to have significant safety impact." Based on the initial safety assessment, the NRC approved, as detailed in MR0182, a licensee proposal (P-81270, dated October 26, 1981) to reduce the leakage of helium to the reheat steam system. The licensee proposed to maintain the leaking module at a pressure (686 psig) slightly above the cold reheat steam pressure (600 psig) but just below the reactor pressure vessel (RPV) pressure (698 psig). This was initially described in MR1181 and subsequently clarified in MR0182. Following the approval, the licensee issued LER No. 50-267/81-068 detailing this event.

Subsequently, MR0282 reported that the leak was in a seal weld and was in a nearly inaccessible location. Several repair options were discussed. One option was to use remote welding techniques to repair the leak; however, if that was not feasible, the SG would have to be removed from the penetration, and a physically small welder could then possibly crawl into the region to make the required repairs. MR0282 also reported that the licensee would likely decide to keep in place the administrative procedures that had been in effect since October 26, 1981, in lieu of these repair options.

As reported in MR0382, the licensee also discovered additional leaks in SG Modules B-2-2 and B-2-6 of the Loop 2 SG. The total number of detected leaks was three. MR0382 also reported that the licensee (letter P-82007) applied for a TS change on January 8, 1982, that would permanently establish the aforementioned administrative controls. The TS changes would require that the helium leak to the secondary steam system at a differential pressure of 10 psi remain less than 400 lbm/day. Further, the main condenser air ejector radiation monitor, which could detect a leak as low as 1.4 lbm/day (based on a current circulating activity of noble gas nuclides of approximately 170 curies), would be continuously monitored to ensure that the daily leak rate (through the air ejector) would be less than 1.4 curies/day. This latter value would ensure that "The resultant annual exclusion area boundary whole body gamma dose and beta skin dose, based upon the release limit of 1.4 curies/day and a plant capacity factor of 0.8, is calculated to be 0.14 millirads and 0.06 millirads respectively." In addition, the licensee attached sample lines to the penetration interspaces and routed the lines to the condensate activity monitors, which were then used to measure the gross activity released to the interspaces.

However, in MR0382, the principal investigator for the project thought these measures did not address two other, related issues. First, the principal investigator speculated that the administrative controls did not account for possible corrosion caused by steam leakage that might accompany the helium leak. Furthermore, the principal investigator noted that the inadvertent closure of the helium pressurization block valve discussed in LER No. 50-267/81-067 had not been addressed. The licensee responded to these concerns by declaring that any moisture in the interspace would be detected by the existing moisture monitors; and if the block valve were inadvertently closed during loop operations, the moisture monitors would again be expected to detect any steam leakage. The licensee also indicated that even if the valve were open and helium leaked into the interspaces, it was expected that this would also be detected. Finally, the licensee stated that any potential corrosive effects of the steam leak would be minimized by maintaining dry, purified helium flow to the interspaces.

MR0584 reported that both LERs No. 50-267/80-030 and 50-267/81-068 had been closed out and the TS limits of 400 lbm/day would be maintained "Since the leak appears not to be readily fixable..." The monthly report again assessed the safety inference for this event concluding, "There are no significant safety implications resulting from the purified helium leak."

The monthly report for July 1986 (MR0786) reviewed the licensee's in-service inspection and testing program and reported the principal investigator's continuing concern regarding long-term moisture intrusion resulting from the limited, allowed steam-water leakage over the years into the PCRV interspaces, such as that from the helium leak of LER No. 50-267/81-068. It was speculated that the existing administrative controls might not be adequate to prevent additional corrosion. Eight months later, these issues were again discussed in the monthly report for March 1987 (MR0387) regarding another review of the licensee's inservice inspection and test program.

B. MR0787 indicated that the licensee had filed a report in accordance with 10 CFR 50.72 on July 29, 1987. The monthly report only indicated " 'D' circulator interspace closure leak of purified helium thru seal above Tech Spec limit" and noted that the licensee intended to file LER No. 50-267/87-018 in accordance with 10 CFR 50.73.

MR1087 reported on the review of LER 50-267/87-018. The LER lists the sequence of events leading up to the discovery of the leak. FSV had undergone several evolutions before the plant operators noticed the leak. While operating at 70% power on July 22, 1987, both the C and D helium circulators unexpectedly increased speed because of a control system problem. The C helium circulator tripped on a circulator-speed-to-feedwater mismatch. Almost 5 hours after the trip and after power had been reduced to 30% to recover the C helium circulator, the D helium circulator also tripped. It was eventually discovered sometime later that D helium circulator had tripped because of excessive shaft wobble. C helium circulator was self-turbining when D helium circulator tripped, which caused a loss of primary coolant system flow on Loop 2. The operators took Loop 2 out of service and reduced reactor power to 2%. About two days later, on July 24, 1987, D helium circulator was started in preparation for increasing reactor power; however, it was shut down due to "an unacceptable amount of wobble on the circulator shaft." Power was then increased with the three remaining helium circulators on their steam-turbine drives. At this point, the operators discovered the helium circulator turbine water drain tank was being pressurized with helium gas. The operators' initial investigation determined that the most likely source for the helium was from the D helium circulator water turbine piping. The licensee also determined that the circulator had sustained substantial damage. The engineering evaluation of the damage and the root causes for the damage were detailed in LER 50-267/87-019 (see MR0787). Since the licensee decided to replace the D helium circulator with the spare circulator and removed the D circulator from the penetration, the exact location of the interspace leak was never determined; however, the licensee was able to verify that there was no primary seal leak. Finally, the damaged circulator was sent to GA Technologies for disassembly and inspection.

MR0787 also noted the pressurization gas flow monitoring system was deficient. Purified helium flow is measured instantaneously, and the flow is constantly fluctuating. The flow fluctuations are due to PCRV pressure changes and previously identified leakage in the Loop 2 SG penetration interspace (see Section 4.2.1.1.a). These fluctuations also cause intermittent actuations of the high flow alarm. It was also possible, therefore, that the intermittent alarms could mask an actual larger or additional leakage problem. The principal investigator for the project pointed out in MR0787 that this masking could hide more serious issues depending on, among other things, how long the alarm condition had existed. For example, the governing TS LCO for instrumentation systems may have needed clarification or could have been subject to interpretation, which could have resulted in or contributed to a delayed response by the operators.

4.2.1.2 Potential Safety Consequences

There were no safety hazards resulting from either of these events. The first helium leak was tracked continuously from 1980 until the FSV permanently shutdown in 1989. The monthly reports for this project followed the event for nearly 7 years. In that time, no corrosion was noted. Furthermore, MR1181 indicated, "...for a small leak, the event does not appear to have significant safety impact." Also, MR0786 noted, "There are no significant safety implications resulting from the purified helium leak." The second incident represented a small addition to the total leakage that was already being tracked. Based on these assessments, these two instances of helium leaks did not present a safety hazard at FSV.

4.2.2 FSV Operating Experience Pertaining to New Gas Cooled Reactor Designs

With so many connecting interspaces, FSV represented a challenge to preventing helium leaks. Apparently, the plant learned to live with a certain amount of leakage and maintained a detailed tracking system for the leaks. The issue of helium leaks (generic or otherwise) would be applicable to any new GCR designs. However, precise technical specification limits and strict test and inspection guidelines would be expected to minimize the effect of minor helium leaks.

4.3 FUEL FAILURES OR ANOMALIES

4.3.1 Summary of Monthly Operational Experience Reports

There were three events reported on in the monthly reports that involved some sort of fuel failures or anomalies. Each of these events is discussed in more detail below.

4.3.1.1 Systems and Components that Failed

A. The first event involved potential fuel damage during fuel handling maneuvers. A core restraint device called a "Lucy Lock" was dropped during fuel handling activities on November 24, 1981. The licensee reported the event in accordance with 10 CFR 50.72, and the event was reviewed as part of MR1281. The licensee had submitted a safety analysis report regarding core region constraint devices (DCN 7904040182, dated March 23, 1979) that was reviewed as part of MR1281. A grappling device was used to remove the Lucy Lock from the top of the core. MR1281 concluded "...there appears to be little chance for fuel damage." MR0282 indicated that GA was "...redesigning the fuel handling system to improve efficiency and reliability in future refueling operations."

B. The second event was an observed tilt in the core power distribution caused by an inadvertent actuation of the reserve shutdown system. MR0882 reviewed a licensee report that was made in accordance with 10 CFR 50.72. The Region 27 reserve shutdown system boron balls (also denoted as boronated graphite balls) had been injected into the core. The licensee first observed a slight power tilt on the core outlet thermocouples. A follow-up investigation confirmed that the boron balls had been injected into the core. Following the initial report, MR0982 reported that even with the boron balls in the core, there was no adverse power peaking. Next, MR1082 reviewed an NRC inspection and enforcement report indicating that the licensee had imposed control rod position limits to compensate for "...the flux tilt due to suspected reserve shutdown material insertion." Finally, the boron balls were removed during an extended maintenance outage as reported in MR1082.

C. The third event was a condition where one reserve shutdown system hopper failed during a surveillance test. Some background information is necessary in order to better understand the event evolution. MR0884 reported that the licensee was going to test two of the reserve shutdown system hoppers and inspect the shutdown material contained in the hoppers. The monthly report also discussed an earlier moisture intrusion event that had occurred in 1975 (see LER 50-267/75-007). Event report LER 50-267/75-007 describes the moisture ingress into the reserve shutdown system hoppers. This intrusion was also reported in LER 50-267/75-018. High primary system pressure allowed moist helium to leak through the control penetrations into the hoppers, where, after several months' exposure to the moisture, it was thought that a contaminate, B_2O_3, had leached out of the boronated graphite balls (B_4C). Subsequently, boric acid was found on the outer surface of the reserve shutdown system material. The reactor vendor, GA, filed a report (GA-A13742, *Status Report on Reserve Shutdown System*, dated November 1975) wherein it was reported "...the boric acid crystals did not adversely affect the operability of the hoppers nor the flow of balls from the hoppers." MR1284 reported further licensee analyses regarding the B_4C balls and noted "...the more highly boronated material were stuck together apparently due to boric acid crystals." An additional source of water somewhere in the purified helium train was suspected of causing the leaching of the B_2O_3 that formed the acid crystals, so the licensee issued LER 50-267/84-012 in accordance with 10 CFR 50.73. The reserve shutdown materials with high B_2O_3 concentrations were located in 18 of the system's 37 hoppers. LER 50-267/84-012 details results of a functional test of the reserve shutdown system hopper for control rod drive and orifice No. 21. The hopper discharged about half of the full amount of the poison (i.e., boronated graphite balls) material during the test.

The following month, MR0185 reviewed additional licensee information regarding the boric acid buildup on the boronated graphite balls. The licensee indicated that "...there is no evidence of B_4C degradation into either B_2O_3 or boric acid due to moisture." However, the licensee also noted that it would replace the material (boronated graphite balls with high B_2O_3 content) with a new reserve shutdown material that had a purity level more than a magnitude greater than that of the old material. At the time of this report, the source of the moisture contamination had not yet been determined.

Several months later, MR0485 reported that the licensee had determined that the moisture in the hoppers came from one of "...two sources: (1) 'breathing' of primary coolant moisture through the hopper purge outlet line during shutdown and other periods when primary coolant pressures could have been higher than that in the hopper and/or (2) possible water leakage into the purified helium stream used to purge the hopper." At this time, it was thought that some sort of moisture intrusion was the culprit that caused the balls to stick together. More than 7 months later, MR1185 reported additional controls implemented by the licensee to ensure the reserve shutdown system would operate when required. The licensee issued an interim TS to verify this; however, the principal investigator for this project expressed, in MR1185, his concerns that the root cause determination for the reserve shutdown system material failure was insufficient and that the interim TS might not be rigorous enough or precise enough to preclude future problems. Finally, MR0286 reported that the licensee had issued a close-out notice for LER 50-267/84-012.

4.3.1.2 Potential Safety Consequences

There were no safety hazards resulting from any of these events. The first event happened during a refueling outage, and no fuel damage occurred as result of the dropped Lucy Lock or its subsequent retrieval from off the top of the core. The second event only slightly skewed the radial and axial power profiles; furthermore, there was no adverse power peaking. The administrative controls imposed by the licensee provided ample protection until the reserve shutdown system boronated graphite balls could be removed during an outage. The third event was a failed surveillance on one hopper, but it represented an apparent ongoing problem since 1975. However, adverse power peaking was neither noticed nor reported during operations from the time when it was first noticed that boric acid was building up on the balls in 1975 until one hopper failed to inject all its material in late 1984. Additionally, even though the reserve shutdown system was degraded, the emergency shutdown capability of the system was not affected. That is, the system would still have been able to perform its design safety function. Moreover, it took almost 6 months to conclude in April 1985 that moisture in the hopper was the apparent (but never definitely determined) "root" cause for the boric acid and the subsequent sticking together of the balls. Based on these facts, it was concluded that this event did not present an undue safety hazard. Therefore, based on these assessments, these three incidents did not present a safety hazard at FSV.

4.3.2 FSV Operating Experience Pertaining to New Gas-Cooled Reactor Designs

The first of these three events, where a fuel handling mishap occurred, is neither unique nor applicable to only GCR designs such as FSV. Rather, the event represented a common type of problem that may occur or be encountered at any LWR or GCR. Therefore, standard fuel handling operator training, specific and precise fuel handling TSs, and extensive refueling operations planning would be expected to prevent this sort of event. Since future GCR designs will likely include a back-up shutdown system such as the reserve shutdown system at FSV (where boronated graphite balls are injected into the core) in order to meet the requirements of General Design Criterion 26 (reactivity control system redundancy and capability), then the lessons learned from the other two events would provide valuable insight for future designs.

Accidental injection of the balls can be either prevented or mitigated in future designs that are similar to or very much like the design used at FSV. Prevention can be accomplished, for example, by applying the experience and training gained from the past 10 years in the commercial nuclear industry. The frequency of accidental initiation of an emergency safeguards systems could be minimized; moreover, the emergency systems can have an extremely high degree of reliability such that they will perform their intended design safety function when required to do so. This experience and learning from the commercial nuclear industry may also be adapted or applied, for example, regarding emergency systems that are required to perform a safety function in very rare instances. The training and experience gained from achieving a high degree of performance and compliance for emergency safeguards systems could also be applied to any new reserve shutdown system designs for future GCRs.

The accidental injection of the shutdown material was somewhat mitigated in 1982 when only one region of the reserve shutdown system balls was injected and no adverse power peaking resulted. Also, it should be noted that the accidental injection of the boronated graphite balls went undetected for almost a month

(MR0882 reported that the event occurred "...in mid-July 1982 but was not confirmed until August 4, 1982."), since there was no indication that the hopper door had failed or was open, and it was only noticed after a power heat balance performed using the core outlet thermocouples indicated a core power distribution imbalance. Future GCR designs could fully instrument such systems to preclude these types of oversights. While applicable, it is also expected that future designs would improve the design of such systems to the point where accidental injection would be even less of a problem.

The last event apparently occurred because some moisture got into the purified helium train. Moisture or any contamination of a purified gas train, whether it is helium or some other type of gas, will always play an important part in any future GCR designs. Again, the nuclear industry has demonstrated an excellent record in performance and surveillance over the past 10 to 15 years in many systems where purity is of the upmost importance. This track record of performance could be put to use in dealing with this issue.

4.4 FAILURES OR CRACKS IN GRAPHITE, PIPES, AND OTHER REACTOR STRUCTURAL COMPONENTS

4.4.1 Summary of Monthly Operational Experience Reports

Two events occurred that specifically involved failures of reactor structural components or graphite. The first event was initially reported in MR1082 and was identified during a review of a licensee submittal regarding the status of cracked fuel element webs. The second event was first reported in MR0484, which indicated that several PCRV tendons had failed their 10-year surveillance inspection. Each of these events is covered in more detail in Section 4.4.1.1.

4.4.1.1 Systems and Components That Failed

A. Sometime before October 1981,* the licensee discovered that a crack had propagated through two stacked fuel elements. The monthly report for October 1982 (MR1082) reported on the review of a GA inspection report. The inspection report indicated that the results of a stacking test showed there was adequate clearance in "...the dowel and dowel socket fit" (MR1082). Thus, even though high stresses resulting from an improper dowel fit were suspected of causing the crack, this mechanism was ruled out as the cause of the cracking. GA indicated it would continue its investigation. The licensee also reported that inspections of other fuel elements found no evidence of cracks or failures.

   MR0783 reviewed three GA Technologies, Inc. reports on "...Nondestructive Fuel and Reflector Block Examinations." The review noted that the crack in the web of fuel element "...S/N-1-2415 did not penetrate to the internal fuel stick."

   The monthly report for May 1984 (MR0584) indicated that several documents pertaining to the licensee's current inspection program and the reactor vendor's analysis, testing, and inspections had been reviewed. The licensee and the reactor vendor concluded, based on calculational models, that high tensile stresses induced by "gap flows" and irradiation-induced stress resulted in high stresses on the "interregional faces" of the two cracked fuel elements. Further, they thought that the use of

*This discovery must have occurred before the program that is the subject of this NUREG began because no monthly reports since the program's inception refers to this particular problem.

H451 Graphite would improve the strength of the elements, although cracking in this material was still possible.* It was noted that if the mechanism for crack propagation is high tensile stress, then crack propagation may be reduced or stopped altogether in the presence of a crack which acts to reduce stress. Load tests indicated that even with cracks, the fuel elements' strength was essentially unaffected.

The NRC Region IV Office funded an independent review and assessment of these results which was to be conducted by Los Alamos National Laboratory (LANL). Following the initial inspection report, the licensee responded to NRC and LANL concerns on August 13, 1984 (licensee response P-84275); that report was reviewed as part of the monthly report for September 1984 (MR0984). The report states that analytical studies conducted at GA Technologies concluded that "...neither seismic loads nor other dynamic loads through the dowel socket system represent stresses in excess of those previously considered in the investigation of crack propagation behavior."

In February 1985 (MR0285) a licensee submittal concerning power peaking factors was reviewed (P-84532, December 17, 1984, GA Document 907079, "Postirradiation Examination and Evaluation of FSV Element 1-2415"). The GA report "...indicated that the time-averaged radial power tilt in element 1-2415 was measured to be 11% based on gamma scanning of Cs-137 fission product distributions." Further, the maximum time-averaged axial peaking factor was determined to be 1.06. This meant that the local peaking factor as determined by the principal investigator was 1.177. The principal investigator raised concerns about how this local peaking factor value compared with the assumed FSAR maximum local peaking factor value of 1.3 for fresh fuel. The principal investigator concluded that the "somewhat higher" empirical value seemed to warrant further analysis even though it only represented measured data from a single fuel block.

Finally, as reported in the monthly report for May 1986 (MR0586), an NRC Office of Nuclear Reactor Regulation (NRR) letter, dated April 10, 1986, regarding a request for additional information on the licensee's proposed fuel surveillance program was reviewed. This review compared the prior GA Technologies report on the post-irradiation examination of the cracked fuel element with a report that appeared in *Gas-Cooled Reactors Today*, Volume 3 ("The Role of Post Irradiation Examinations of GACR Fuel in Evaluating and Improving Predictions of Power Distributions" by J. T. Dawson and D. J. Edens). Essentially, the United Kingdom's Central Electricity Generating Board used gamma scanning to assess in-core power peaking margins in fuel elements in advanced GCRs (AGR). The GA Technologies report (see MR0285) claimed results from its testing and analysis confirmed the FSV calculational methods; however, these results were not correlated to results from models that were used to generate power peaking limits in the FSAR. Therefore, it appeared that the quantitative measure of thermal margins by the British was more comprehensive than the qualitative method of determination used by the NRC and FSV.

B. The monthly report for April 1984 (MR0484) reported that on March 28, 1984, the licensee reported in accordance with 10 CFR 50.72 that it had discovered that a "significant" number of the tendon

*The FSV UFSAR describes the various types of graphite used in the fuel and reflector elements. "The primary fuel and reflector structural material in the initial core and the first and second reloads was H-327 (needle-coke) graphite. Beginning with the third reload, H-451 (near isotropic) graphite was used in the fuel and reflector elements. Approximately one half of the fuel and reflector elements in reload 3 utilized H-451 graphite. H-451 graphite will be the primary structural material used in all fuel and reflector elements installed after reload 3. Analysis has confirmed (Ref. 1) that H-451 graphite improves the mechanical, thermal, and fluid flow characteristics of the reactor."

wires used in the FSV PCRV were either corroded or failed. The discovery occurred during the 10-year surveillance inspection of the PCRV.

The PCRV tendon system consists of 448 tendons and each tendon consists of 152 or 169 1/4 in. wires. The 448 tendons may be classified into four distinct groups: 24 each in the top cross head and bottom cross head classes, 90 in the vertical or longitudinal class, and 310 in the circumferential class. Twenty-seven load cells are used to detect any loss of pre-stress in the PCRV and are installed on selected tendons. The top cross head and bottom cross head classes have two load cells each, while the vertical class has six load cells. The remaining 17 load cells are associated with the circumferential class. The tendons are in sealed boxes that were opened for the inspection, and selected sample wires were removed for further inspection. None of the sample wires showed signs of corrosion; however, the licensee found broken or corroded tendon wires in the center of at least six tendons, and up to 30 of a possible 169 wires were found broken in a single tendon. Upon this notification, the NRC's Region IV Office restricted FSV to 2% power or less following its then-current refueling outage. The licensee issued LER No. 50-267/84-005 in accordance with 10 CFR 50.73 on April 26, 1984.

MR0484 further indicated that the apparent root cause of the corrosion was moisture in the sealed boxes, along with the lack of cover grease on the ends of the tendon wires. A preliminary inspection found "...iron oxides and hydroxides are present where grease is not..." Several possible explanations for the loss of the cover grease were presented in MR0484. The following monthly report (MR0584) reviewed two letters issued by the licensee (P-84110 dated April 12, 1984, and P-84119 dated April 25, 1984)—one that provided additional information regarding the tendon corrosion problem and the other requesting permission to return to full power following the refueling outage. Test data indicated that the PCRV strength and integrity were not affected by the corroded wires, and the licensee had begun a monthly monitoring of all 27 load cells with a semi-annual inspection of the tendon terminations. MR0584 also reported that the NRC had issued a safety evaluation report (SER) that supported the licensee's conclusions. The monthly report for August 1984 (MR0884) indicated that additional chemical tests had been performed, and the licensee was increasing the frequency of its surveillance tests of the tendon system. Additionally, it was reported that the corrosion extended no more than 36 in. from the end points. Further, it was reported that "...lift-off testing verifies tendon operability for all tendons with failed wires," and the licensee was waiting for the test results from H_2S chemical testing. A licensee response (P-84287, dated August 20, 1984) regarding inspections and possible modifications concerning the PCRV tendon system was reviewed the following month in MR0984. It had been determined that moisture was a common element in all the cases of corrosion that were discovered, and possible microbiological corrosion was being investigated as a source or mechanism for the corrosion. Methods of keeping the atmosphere within the sealed boxes dry and free from moisture were being evaluated, as well as sealing of all possible moisture admission paths.

The monthly report for November 1984 (MR1184) reported that the licensee had submitted a report in accordance with 10 CFR 50.72 on November 1, 1984, indicating that microbiological attack on the protective grease for the tendons was thought to be responsible for the corrosion of the tendon wires. The protective grease (No-Ox-Id) used was a sulfonate-based grease and was more prone to microbiological attack than an alkaline-based grease. It was also thought that the microbiological attack was manifested in the formation of formates and acetates that were found on the wires. The next time this issue was addressed was in the monthly report for February 1985 (MR0285) that reviewed two licensee submittals. The first submittal (P-84523, dated December 14, 1984) concerned PCRV tendon accessibility and a proposed tendon surveillance program. The second submittal (P-84543, dated December 31, 1984) was a PCRV tendon engineering report. The engineering report determined that most of the 448 tendons were accessible for both a visual and lift-off inspection for "...at least one end..." Three tendons were visually inaccessible from either end, and 34 tendons were inaccessible for lift-off testing at either end. The corrosion mechanism was verified to be microbiological action on the grease resulting in the formation of acetic and formic acids that led to the corrosion. Licensee testing confirmed that corrosion would be stopped in the absence of oxygen. Furthermore, the licensee proposed to use a nitrogen blanket in the tendon enclosures to halt the corrosion, and all that remained to do was to ensure the leaktightness of the enclosures. According to the licensee, while purging the enclosures of water vapor was considered important, it was not as important as eliminating the oxygen content.

The monthly report for August 1985 (MR0885) examined the licensee's proposed corrective actions and long-term resolution for the tendon corrosion problem. Also reviewed during this period were two SERs forwarded via a letter dated July 8, 1985, from the NRC's Region IV Office to the licensee. The two SERs were "PCRV Tendon Corrosion Investigation and Proposed Remedy" and the LANL evaluation of restart activities. Finally, nearly a year later, the monthly report for July 1986 (MR0786) reviewed a licensee discovery of a tendon enclosure more than half-filled with water. The licensee had been inspecting the enclosures with increasing frequency and still pursuing options, such as a nitrogen blanket, to prevent further corrosion. On March 26, 1986, the licensee discovered that one of the void enclosures (approximately 55 gal) was more than half-filled (28 gal) with water. It was determined by the licensee that the source of the water was neither the service water system nor the PCRV cooling water system. Two proposed sources for the water were being investigated: water from an external source through the tendon closure seal gap had leaked since initial installation into the enclosure, or water from the construction cleanup washdown 12 years ago had collected in the void. Twenty wires in the tendon were found to be failed; however, "...testing demonstrated that the tendon still exceeded the minimum acceptance criteria for design loading." An NRC-NRR request for additional information dated June 12, 1986, was also reviewed as part of MR0786.

4.4.1.2 Potential Safety Consequences

Based on the inspection report listed in MR1082, the licensee assessed the safety analysis applicability of potential coolant flow blockage from failed fuel elements. The licensee concluded that the cracks observed so far did "...not constitute a significant hazard." The cracked fuel element webs appeared to be superficial and were discovered through routine inspections. Based on this information, the cracked fuel element web issue was not considered a safety issue at FSV.

As reported in MR0584, "Test data indicated that the PCRV strength and integrity were not affected by the corroded wires,...", and, as part of that monthly report, "MR0584 also reported that the NRC had issued a safety evaluation report (SER) that supported the licensee's conclusions."

Next, in MR0885 it was reported that "...lift-off testing verifies tendon operability for all tendons with failed wires." Finally, MR0786 reported "...testing demonstrated that the tendon still exceeded the minimum acceptance criteria for design loading." Therefore, based on the foregoing, it is concluded that the PCRV tendon corrosion issue did not present any undue safety hazard to FSV.

4.4.2 FSV Operating Experience Pertaining to New Gas-Cooled Reactor Designs

Both the PCRV tendon corrosion and the cracked fuel element web issues found at FSV would most likely be applicable for any new GCR design utilizing graphite or a PCRV. However, both issues, as demonstrated at FSV, may be readily dealt with through a rigorous and thorough test and inspection program. Microbiological attack was a slow-evolving phenomenon and by the end of FSV's lifetime, it had been stopped. Additionally, post-irradiation examination of the cracked fuel element webs indicated that controlling key parameters, such as peaking factors, during plant operation would limit the cracking phenomenon (i.e., self-arresting). In both these instances, precise and specific TS limits and strict test and inspection guidelines would contribute to any recommendations for any future GCR designs.

4.5 HUMAN FACTORS AND OPERATOR PERFORMANCE ISSUES

4.5.1 Summary of Monthly Operational Experience Reports

There were 47 events reported in the monthly reports that resulted from human factors or some other sort of personnel error. A summation of the nature and character of the various kinds of human factors involved in these events follows, and a general examination of the events is discussed in more detail.

4.5.1.1 Systems and Components That Failed

It is recognized that the "root cause" for all events can ultimately be traced back to some type of human error—a manufacturing error, engineering design error, installation error, etc.—for this review it was decided to classify only those events resulting directly from a human error and to group those events resulting directly from human error into six human activity codes. The six codes will be explained first, then the numbers and types of events classified into these six areas will be discussed.

The first class is licensed operator error. This cause code captures errors of omission or commission committed by licensed reactor operators during plant activities while performing their licensed-operator duties. These errors may initiate events or may be committed during an event. Licensed operator errors typically occur as a result of carelessness, lack of experience or training, fatigue, stress, attitude, poor work habits, or a combination of all or some of these factors. Improper supervision is also included whenever the event was the result of improper instructions given by a licensed operator such as an operations supervisor or control room shift supervisor. Two examples of operator error would be "the operator withdrew the control rods out of order" or "the operator failed to bypass a secondary trip function following the primary trip actuation, resulting in a second trip."

The second class is other personnel testing activity error. This error class is similar to licensed operator errors except that these errors are committed by staff personnel other than licensed operators. This cause code captures errors of omission or commission committed by non-licensed personnel involved in plant activities, not performing licensed activities, and performing some sort of testing activity. Included in this category would be plant staff personnel (such as technicians, maintenance workers, equipment operators) and contract personnel. Not included in this category are administrative control types of problems, such as incorrect procedures or inadequate planning activities, that caused personnel to take inappropriate actions. Two examples of this testing activity type of personnel error would be cases where test personnel inadvertently shorted two wires while performing a test, or where the steps in a surveillance procedure were performed out of order.

The third class is other personnel maintenance or repair activity problem. Activities included in this category are maintenance and repair. This cause code is the same as the second code, except that in this instance the personnel were performing maintenance or engaged in a repair activity. This category would include, for example, a failure to repair or maintain a motor-generator set that caused that motor-generator set to trip on high vibration because of a worn out flywheel bearing. Other examples would be maintenance personnel omitting two fasteners while reassembling a valve operator, loose battery connections found during a surveillance, or a valve packing leak. These errors may have been committed by either plant or contractor personnel staff during the performance of equipment repair or replacement activities, and the error may be one of omission or commission. Additionally, this personnel error may be due to an intrinsic error by personnel performing the activity or task or to an error caused by incorrect procedures.

Similarly, the three remaining classes are also activity-related. Two of these classes are installation activity personnel error and radiation protection activity personnel error. These two classes are self-evident and, like the previous two classes, include events where a personnel error was committed during a particular activity. Anything not included in the previous five categories was put into a category called "other activity personnel error." Even though this last class is general and somewhat vague in description—and the classification of an event into this category is subject to interpretation based on experience, opinion, or other factors—it was intended to group the events reported in the monthly reports in a fashion that lent itself to potentially useful diagnosis.

The 47 events are distributed among the 6 human activity classes as follows:

Table 1. Personnel Activities Distribution

Personnel Activity Error	Number of Events	Percent (%)
Licensed Operator	6	13
Testing	22	47
Maintenance/ Repair	5	11
Installation	2	4
Radiation Protection	2	4
Other	10	21

Each of these groups of human factors is discussed in more detail in the following paragraphs.

A. The licensed operator errors may be characterized by two specific types of failures. The first is a misinterpretation of, or failure to follow, a procedure during the performance of licensed duties. Failing to follow a procedure may be defined as the licensed operator's having failed to perform the procedure, performed the procedure steps in the wrong or improper order, or left a step out when performing the procedure. Two examples of this type of failure may be found in MR0383 (see LER 50-267/83-001) and MR0388 (see LER 50-267/88-001). The other type of operator failure is an intrinsic failure by the licensed operator to perform an intended action. An example of an intrinsic failure is placing a switch on the control panel in the wrong position.

Four of the six licensed operator errors caused a missed TS surveillance resulting in a TS LCO violation. The remaining two licensed operator errors resulted in a loop shutdown. These latter two instances are described in MR0186 (see LER 50-267/86-010) and MR0987 (see LER 50-267/87-020).

B. The period with the highest occurrence of testing activity personnel errors was during 1985 and 1986. Twelve of the 22 testing activity personnel error events occurred during that period—6 in 1985 and 6 in 1986. No testing activity personnel errors were found for events that occurred in 1981 and 1982. Of the 22 testing activity personnel error events, 5 led directly to a reactor scram, 6 led directly to a helium circulator trip, and 4 were due to a TS error that led to a TS LCO violation. The last class of events occurring as the result of a testing activity personnel error (seven events) were scattered throughout various areas with no discernible grouping evident.

These last seven events were all miscellaneous in nature. MR0883 reported that a technician attempted to close a breaker onto a bus that was already powered from another source. In another event, personnel inadvertently placed a continuous sampler out of service (see MR1284 or LER 50-267/84-010). Others, such as a test engineer testing the wrong level controller (see MR0985 or LER 50-267/85-011) during a surveillance or a maintenance technician pulling the wrong fuses for a test (see MR0486), were classified as belonging to this miscellaneous category.

The five events resulting from a testing activity personnel error that directly resulted in a scram occurred randomly and were not clustered in any particular time frame. However, two of them did happen within a few days of each other. MR0985 reported that the licensee had filed reports in accordance with 10 CFR 50.72 that reported two events resulting in scrams. The first event happened when a technician committed an error during a surveillance on September 16, 1985, causing a scram. Ultimately, this event was reported in LER 50-267/85-009. The other event occurred on September 25, 1985, when a technician shorted two leads together during a test, causing blown fuses and resulting in a reactor scram. This event was also eventually reported in LER 50-267/85-015.

Another testing activity error event was reviewed almost 5 years after the event had occurred. MR0684 (see LER 50-267/79-028) reviewed an event that resulted when an instrument technician grounded an instrument panel during testing, causing a loop shutdown and subsequent scram. A technician cleaning contacts in an instrument caused a scram of all the even-numbered control rods. This event was reported in the plant monthly operating report and reviewed as part of MR0887. An operator was attempting to balance feedwater flow during testing and did not realize that the plant was already in the power mode of operation (feedwater flow between the steam-driven feedwater pumps must be balanced before the plant is placed in the power mode of operation). The subsequent imbalance in feedwater flow caused a loss of feedwater flow transient and resulted in a reactor scram. This event was first reported in MR0687, indicating that the licensee had filed a report in accordance with 10 CFR 50.72; subsequently, the licensee filed LER 50-267/87-016 in accordance with 10 CFR 50.73.

The six events that resulted in a direct trip of a helium circulator as a result of a testing activity personnel error all occurred between 1985 and 1988. Just like the two scrams discussed previously, two of these events occurred within a few days of each other. The first happened on October 15, 1985, and was initially reported in accordance with 10 CFR 50.72 (see MR1085) and subsequently reported in LER 50-267/85-022 (see MR1285). An equipment operator conducting a surveillance test inadvertently drained the water from an instrument sensing line causing the B helium circulator to trip. The second event was also reported in accordance with 10 CFR 50.72, but this event occurred on October 23, 1985 (see MR1085). The event was caused by a maintenance technician and was later reported in LER 50-267/85-023 (see MR0186). MR1285 indicated that the licensee had reported, in accordance with 10 CFR 50.72, on December 4, 1985, that a technician had opened the wrong electrical breaker; the error caused a loss of bearing water supply to the B helium circulator, which subsequently tripped. MR0186 reviewed LER 50-267/85-027 that the licensee had filed regarding this event. The other three events resulting in a helium circulator trip from a testing activity personnel error each occurred in a separate year. The first was reported in MR0386. The event was reported in LER 50-267/86-007; it happened during an inservice leak test when a tank was inadvertently drained. Draining the tank caused a high differential pressure trip of the A helium circulator. The second event resulted from a technician's error during testing which caused two helium circulators to trip. This event was first identified in MR0987 as a 10 CFR 50.72 report by the licensee. The last event was reported in the resident inspector's operating report and reviewed as part of MR1088. A technician pulled the wrong fuse during a surveillance and caused the A helium circulator to trip.

C. The five maintenance or repair activity errors were general in nature. For example, a technician performing a maintenance task inadvertently removed some electrical tape from the end of a wire, causing an instrumentation channel to become inoperable (see MR0386 or LER 50-267/86-009). Another time, post-maintenance inspections were not performed (see MR0585 or LER 50-267/85-006).

D. The first installation activity personnel error was an incorrect valve lineup that was found after a new valve had been installed (see MR0983 or LER 50-267/83-026). The second installation activity error occurred when a contract worker altering cable tray penetration fire seals failed to follow procedures (see MR1285). MR0886 indicated that the licensee had filed a report in accordance with 10 CFR 50.72 about the contamination of three individuals; this event was classified as a radiation protection activity event. Another radiation protection activity personnel error occurred when a grab sample was not taken and a TS surveillance was missed, resulting in a TS LCO violation (see MR0987 or LER 50-267/87-017). Finally, the ten other activity personnel errors were almost all reports of injuries to workers. For examples, see MR0589 and MR0689. Injuries accounted for nine of the ten other activity personnel errors, and all nine of these occurred after October 1988, before the plant permanently shutdown in 1989. Two of these nine events involved the deaths of individuals, one resulting from natural causes (see MR0989) and the other from a murder that occurred offsite (see MR0589). The last other activity personnel error event was a misplaced seismic qualification document (see MR1286).

4.5.1.2 Potential Safety Consequences

There were no extraordinary human factors issues or personnel errors uncovered or discovered as a result of analyzing the monthly reports for this project. Moreover, none of these errors involved potential safety concerns. Therefore, based on these assessments, these personnel error issues did not present a safety hazard at FSV.

4.5.2 FSV Operating Experience Pertaining to New Gas-Cooled Reactor Designs

Human error can never be eradicated completely; however, a dedicated and rigorous program to reduce human error can be expected to reduce the human error rate to an acceptably low level. For example, the commercial nuclear industry has a good track record over the past 10–15 years regarding testing and calibration of equipment. The testing and calibration of instruments and systems in this industry during this time has resulted in a very low human error rate. Inadvertent TS LCOs resulting from testing or surveillance, or accidental scrams or initiations during surveillance or testing, have become virtually zero. The techniques, training, and attention to detail exhibited or practiced in the commercial nuclear industry could be implemented or applied for all future GCR designs. This same sort of exemplary performance (and testing) record could be also accomplished for future GCRs.

4.6 OTHER EVENTS OR CONDITIONS (RELEVANT TO CURRENT GAS-COOLED REACTOR DESIGNS)

4.6.1 Summary of Monthly Operational Experience Reports

One hundred ninety–six events were found among the monthly reports for the earlier project that did not belong to any of the previous five areas: (1) water incursion events and failures of moisture detection systems [see Section 4.1 of this report], (2) air and other unwanted gas incursion events and failures of gas detection systems [see Section 4.2 of this report], (3) fuel failures or anomalies [see Section 4.3 of this report], (4) failures or cracks in graphite, pipes, and other reactor structural components [see Section 4.4 of this report], and (5) human factors and operator performance issues [see Section 4.5 of this report].

This group of other events or conditions was divided into seven sub-categories: (1) events related to secondary side systems or issues, (2) events related to electrical distribution concerns or systems, (3) events from I&C systems other than the previous systems listed, (4) events related to auxiliary systems or issues, (5) events related to primary reactor systems, (6) events related to waste management items, and (7) events that did not fit into any of the previous sub-categories. A brief description of each of these seven sub-categories is listed below.

The first sub-category is for, secondary side events and is best characterized as containing those events that resulted from the failure of some component, device, or item that was part of the secondary side. In this instance, the secondary side of FSV is taken to mean that portion of the plant that encompasses the power conversion systems that convert the energy produced by the GCR into energy used by the rest of the plant to produce electricity. This sub-category also includes those instances when a component or device failed to operate. These components, devices, or items are such things as valves, pumps, and piping associated with systems such as feedwater, main turbine generator, main steam, power conversion, main condenser, and support systems. The second sub-category is for, electrical distribution and includes systems, components, and items associated with such systems as alternating current (ac), direct current (dc), vital busses, essential electrical distribution such as 480 volts alternating current (Vac), emergency power generation, main generator power distribution, main transformers, and site distribution transformers. The third sub-category is for, I&C systems other than previous listed and includes those systems or subsystems associated with control and instrumentation of systems other than moisture detection (see Section 4.1), gas detection (see Section 4.2), and nuclear instrumentation. Systems and components associated with such systems as turbine generator control, feedwater control, emergency generator control, leak detection, and radiation monitoring would be included in this sub-category. The fourth sub-category is for auxiliary systems events and includes events occurring in fuel handling and storage systems, auxiliary handling equipment systems, helium purification systems, helium and nitrogen storage systems, reactor plant cooling water systems, circulating water systems, PCRV cooling water systems, and instrument and service air systems. The fifth sub-category is for primary reactor systems and includes events associated with the reactor core, including the reflector, reactor vessel, control rod drives, primary coolant, steam generators, orificing, and helium circulators. The sixth sub-category is waste management and deals primarily with events related to solid, liquid, or gaseous radwaste. The last sub-category includes events that did not pertain to any of the other generalizations. This sub-category included events concerning security violations or acts of nature that were noted but did not affect plant operations.

As noted earlier, there were 196 events in the broad category denoted "other events." The breakdown between the 7 sub-categories was 13 events related to secondary side systems, 24 events linked to electrical distribution systems, 50 events that affected "other" I&C systems, 39 events that concerned auxiliary systems, 40 events associated with the primary reactor systems, 12 events concerning waste management, and 18 events falling into the sub-category of "other."

Each of the sub-category distributions of events will be discussed separately and in greater detail in Section 4.6.1.1. No discernible pattern was noticed regarding the year-to-year distribution of events in this category, nor did any one sub-category contain a preponderance of events. Therefore, no specific comments can be made about this category, other than that more than 70% of all the events reported in the ORNL monthly reports were in this category.

4.6.1.1 Systems and Components that Failed

A. Three of the 13 events associated with the secondary side exemplified the type of event in this sub-category. MR0586 reviewed a licensee report filed in accordance with 10 CFR 50.72 indicating that a main steam isolation valve that had been tagged/closed had drifted open and caused a power transient. MR1087 first noted that the licensee had filed a report of a turbine building fire that was caused by a hydraulic oil leak. Eventually, the fire and oil leak caused a loop shutdown and a subsequent 12-minute loss of forced cooling . Subsequently, the licensee filed LER 560-267/87-023 in accordance with 10 CFR 50.73 as a 30-day root cause analysis of the turbine building fire. Finally, the event that led PSC management to decide to permanently shut down FSV was the discovery of cracks in the main steam ring header reported in LER 50-267/89-018 and reviewed as part of MR0989 (the last monthly report for the project). Subsequently, the cracks in the main stream ring header were determined by metallurgical examinations to be the result of thermal cycling of the header.

B. The electrical distribution sub-category contained 24 events; 12 of these events were related to either the emergency diesel generators (EDGs) or the plant's large power transformers. Five of the events were the result of so-called bad weather. For example, the severe blizzard that struck central Colorado in May 1983 caused a complete loss of offsite power to FSV. MR0583 reported on the effects the blizzard had on FSV operations and indicated that even though the A emergency diesel generator was out of service at the time, and the B EDG tripped following the transient, there was still at least 10 hours available before the decay heat load from the core could have begun to cause core damage. The licensee filed LER 50-267/83-018 regarding this event.

MR1283 reported that high winds at the plant on December 8, 1983, caused a fire detector on the reserve auxiliary transformer to come loose and impact a transformer deluge monitor. The deluge monitor malfunctioned and actuated the deluge system for the transformer, which caused the transformer to trip. The tripped transformer caused a loop shutdown followed by a reactor scram and a turbine trip. A grid distribution disturbance on April 4, 1988, resulted in a simultaneous manual reactor scram and a turbine trip. This event was initially reported in MR0488 as a 10 CFR 50.72 report. Subsequently, MR0588 extensively reviewed the final analysis for this event that is detailed in LER 50-267/88-004.

C. The 50 other I&C events were distributed among 4 general areas. First, 13 events were classified as inoperable instruments that were out of calibration or had drifted from their correct setpoints. The next classification was for events where instruments were moved, disturbed, or otherwise subjected to physical motion that produced an erroneous or false signal from the instrument. Eight events were in this group. Five events were then grouped together in which instruments failed, sent a false signal, or tripped because of a short between contacts or because the instrument had dirty contacts. The final grouping of events contained by far the largest number. There were 24 events in this last group, all of them resulting from instrument "noise" or a spike on an instrument's output signal. A few examples will help illustrate these classifications.

Instruments that had drifted out of calibration may be characterized by two LERs: LER 50-267/82-051 and LER 50-267/82-054. Each of these LERs reported an instrument drift resulting in inoperable ultrasonic monitors. MR0383 reviewed both of these LERs. An example of a short or dirty contacts was reported in MR0785 regarding LER 50-267/85-008, in which water in a junction box caused a short in an instrument, subsequently resulting in a scram. MR0789 reported a 10 CFR 50.72 event in which a circuit card was moved, causing an instrument signal transient. MR1287 reviewed LER 50-267/87-029, which reported that electrical noise had caused the control circuitry to generate a scram signal. A common result from the instrument noise problem was the generation of a rod withdrawal prohibit alarm. Most of the 24 events reporting instrument noise or a signal spike came from this type of alarm. For example, MR0186 reviewed LER 50-267/85-025 that reported rod withdrawal prohibit alarms from instrument noise. Also, MR0186 mentions several 10 CFR 50.72 reports by the licensee regarding rod withdrawal prohibits generated during the month. Eventually, the licensee issued LER 50-267/86-004 that examined these noise spikes on instrument signals.

D. The 39 events concerning auxiliary systems did not separate into readily discernible groups. So no simple breakdown by groups was done; rather some very general observations were noted. Seven events involving auxiliary systems were related to inoperable snubbers on auxiliary piping. For example, MR0383 reviewed LER 50-267/82-052, which reported eight instances of inoperable snubbers. Six events reported some failure associated with the failure of a valve. MR1283 examined leaking safety relief valves reported in LER 50-267/83-035. LER 50-267/83-035 listed 18 reports of leaking valves that had occurred since 1980. MR0488 reported flooding in a pump room that resulted from a failed expansion joint in the circulating water system. This event was initially reported in a 10 CFR 50.72 report, which the licensee later upgraded to an LER (see LER 50-267/88-006). Eight events were randomly distributed among various types of components. One example of a failed component was described in MR0983, which reviewed a preliminary notification of occurrence report regarding a compressor malfunction in the helium purification system. Another example was reported in MR0286. The licensee filed a report in accordance with 10 CFR 50.72 that it had found charred and burned cables on several components of support systems to the helium circulators, such as a circulator cold reheat drain valve, bottom head cooling system valves, and drain valves for the helium moisture separator. Several of the auxiliary systems occurrences involved some problem with the fire seal penetrations (see MR0186) or fire barriers (see MR0386). However, events associated with a "whole system" problem were the most numerous in this sub-category. For instance, in reporting the results of a meeting among NRC personnel, project personnel, and licensee personnel, MR0684 indicated that one of the subjects examined was the "Frequency of the unavailability of the emergency feedwater supply header to the helium circulator water turbine drives."

Beginning in July 1986, the safety-grade emergency firewater cooldown process for the plant came under scrutiny. MR0786 reported that the licensee had filed a 10 CFR 50.72 report on July 11, 1986, concerning the preliminary analysis of the emergency firewater cooldown using the reheater section of the SG. The preliminary analysis indicated that there may not have been sufficient flow from the emergency firewater system to account for all the decay heat from a shutdown from 100% power. The single failure criteria for firewater were also noted in MR0786 in reference to another 10 CFR 50.72 report regarding the firewater safe shutdown cooling flow path. Following this, MR0886 reviewed LER 50-267/88-020, which discussed the safety analysis associated with the environmental qualification program at FSV as part of 10 CFR 50.49. In essence, the safe shutdown analysis performed by the licensee depended on the emergency firewater system to provide adequate decay heat removal following a shutdown; however, the system as configured may have been unable to perform its intended safety function. The following monthly report, MR0986, reviewed another 10 CFR 50.72 report that discussed the re-analysis of the firewater cooldown following shutdown from a design-basis earthquake potentially occurring at full power. This 10 CFR 50.72 report was upgraded to an LER in accordance with 10 CFR 50.73 (see LER 50-267/86-026). This re-analysis of firewater cooldown capabilities was further reviewed as part of MR1186 and MR0187.

E. Twenty-nine of the 40 events associated with the primary reactor systems came from 1 of 3 occurrences: helium circulator(s) trip, loop(s) shutdown, or a direct reactor scram. Five events involved the CRDs. One event was an inadvertent reserve shutdown system initiation and injection into the core that was reported in LER 50-267/88-017 and reviewed as Part of MR0189. The remaining five events were either TS LCOs or leaking PCRV safety valves.

   The first of the five events associated with the CRDs was the two control rods fail-to-scram event that is discussed in Section 4.1.1.1.e of this report (also see MR0482 which reviewed LER 50-267/82-007). MR1085 indicated that the licensee had filed a report in accordance with 10 CFR 50.72 reporting a control rod that failed to move when signaled to do so. Another 10 CFR 50.72 report, reviewed as part of MR0486, indicated that a control rod had been dropped into the core. LER 50-267/89-015 was reviewed as part of MR0889 and detailed the circumstances surrounding an inoperable control rod pair. A control rod that was unable to scram or be inserted into the core was reported by the licensee in LER 50-267/89-009, which was reviewed as part of MR0589.

F. Small liquid releases accounted for 7 of the 12 events related to waste management issues. Two such examples were reported in two LERs, LER 50-267/83-036 (reviewed as part of MR0184) and LER 50-267/85-004 (reviewed as part of MR0685). The other five waste management events involved gaseous effluents from radwaste streams released from the plant. For example, two minor releases were reported in accordance with 10 CFR 50.72 and reviewed as part of MR0687.

G. Ten of the 18 other events involved some sort of security violation at the plant. Three events reported a document had been misplaced or stolen. An earthquake in Wyoming that was barely felt at FSV was reported in MR1084. MR0687 indicated that a tornado had touched down somewhere near the plant, which was reported by the licensee in a 10 CFR 50.72 report. On January 30, 1988, a natural gas well located about 5 miles from the plant exploded and resulted in a 10 CFR 50.72 report by the licensee, noted in MR0288.

4.6.1.2 Potential Safety Consequences

The potential safety consequences from the 196 other events were collectively representative of routine operational events at FSV. That is, none of the seven sub-categories contributed an inordinate number of events to the overall total. In almost 8 years of operational events assessment, four of the sub-categories contributed on the average about three to five events per year, and the remaining three sub-categories contributed about one to two events per year. This sort of "even" distribution of events indicates a generally flat profile. Moreover, within each sub-category, there were no features that dominated or any individual items that were clearly outside the norm or exhibited anything other than a normal relationship. A detailed analysis of the 196 events may reveal a hidden component or cause that was not apparent from a cursory study and consequently could be a dominant factor. A more comprehensive study might uncover a relationship between cause and effect that was not discernible from this simple review and analysis. Further, the experiences and lessons-learned from the commercial nuclear industry from the past 10–12 years can be applied to these other types of events seen in this category. That application may reduce both the frequency and relative importance of any future events.

4.6.2 FSV Operating Experience Pertaining to New Gas-Cooled Reactor Designs

The nuclear power industry currently has a good record regarding secondary side, auxiliary, electrical distribution, I&C, primary reactor, and waste management systems over the past 10–12 years. The lessons-learned, experience, and training developed throughout the years by the commercial nuclear industry can be applied to any future endeavors or designs for GCRs. Almost all of the events detailed in this report were characterized by their normalcy, with no readily identifiable trait to set any one apart from the rest. That is, most of these events were what normally could be expected during normal plant operation throughout the lifetime of the plant. That is, the events occurring at FSV during 1981–1989 concerning electrical distribution could be expected to occur in any electrical distribution system at any power plant. The electrical distribution system is not unique to a particular type of power plant, and the electrical distribution system for a coal-fired power plant is very similar to the electrical distribution system for an HTGR like FSV. However, some of the secondary systems, and certainly the primary reactor systems, are unique to an HTGR or a GCR. Therefore, any useful information or lessons-learned that may be gleaned from the operational events analysis of this group of "other events" at FSV must be acquired from a more detailed review.

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APPENDIX A

OBSERVATIONS OF DR. DAVID L. MOSES ON THE FORT SAINT VRAIN (FSV) LICENSING PROCESS

APPENDIX A

OBSERVATIONS OF DR. DAVID L. MOSES ON THE FORT SAINT VRAIN (FSV) LICENSING PROCESS

When reviewing the operational experience at FSV as it may apply to the future licensing 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 LWRs. This appendix attempts to explain some of the differences that existed and how these differences can impact subsequent licensing and regulatory analyses that would pose 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 Act. 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, the regulatory requirement to convert to a Class 103 license was dropped. Thus, the NRC allowed a rather wide 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.

The period of FSV operations 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 the following:

1. 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 Prestressed Concrete Reactor Vessel (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.

2. 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 Technical Specifications Upgrade Program (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.

3. 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 review of the FSV Preliminary Safety Analysis Report (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 now plans to develop a chapter on reactor physics in the RTDP [Reactor Technology Development Plan], as described in Section 4.3.4. 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."

4. 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 reactor 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.

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11. ABSTRACT *(200 words or less)*

Oak Ridge National Laboratory (ORNL) performed an operational events assessment of Fort St. Vrain (FSV) nuclear power plant during 1981–1989. As part of that assessment, the ORNL principal investigator filed 96 monthly reports detailing the results of each month's analysis. This project reviewed those 96 monthly reports and identified pertinent operating experiences that occurred at FSV which may be applicable to current or future gas cooler reactor (GCR) designs.

Two hundred seventy–nine events were catalogued into seven general categories: (1) water incursion events or failures of moisture detection systems, (2) air or other unwanted gas incursion events and failures of gas detection systems, (3) fuel failures or anomalies, (4) failures or cracks in graphite, pipes, and other reactor structural components, (5) failures of nuclear instrumentation systems, (6) human factors and operator performance issues, and (7) other events or conditions relevant to current GCR designs. Several events were reviewed over a period of years and appeared in more than one monthly report.

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