Lessons Learned from Fast Flux Test Facility Experience

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Abstract. The Fast Flux Test Facility (FFTF) is the most recent liquid-metal reactor (LMR) to operate in the United States, having operated from 1982 to 1992 and played a key role in LMFBR development and testing activities. In addition to irradiation testing capabilities, FFTF provided long-term testing and evaluation of plant components and systems for LMFBRs. The FFTF was highly successful and demonstrated outstanding performance during its nearly 10 years of operation. The technology employed in designing and constructing this reactor, as well as information obtained from tests conducted during its operation, can significantly influence the development of new advanced reactor designs in the areas of plant system and component design, component fabrication, fuel design and performance, prototype testing, site construction, and reactor operations. Efforts have been made to preserve important lessons learned during the nearly 10 years of reactor operation, and a brief summary of these Lessons Learned will be discussed.

Key Words: FFTF; Lessons Learned; Liquid Metal Fast Reactor; reactor operations.

1. Introduction

The Fast Flux Test Facility (FFTF) is the most recent liquid metal reactor (LMR) to operate in the United States, having operated from 1982 to 1992 (FIG. 1). FFTF is located on the DOE Hanford Site near Richland, Washington. The 400-MWt sodium-cooled, low-pressure, high-temperature, fast-neutron flux, nuclear fission test reactor was designed specifically to irradiate Liquid Metal Fast Breeder Reactor (LMFBR) fuel and components in prototypical temperature and flux conditions. FFTF played a key role in LMFBR development and testing activities. The reactor provided extensive capability for in-core irradiation testing, including eight core positions that could be used with independent instrumentation for the test specimens. In addition to irradiation testing capabilities, FFTF provided long-term testing and evaluation of plant components and systems for LMFBRs.

The FFTF was highly successful and demonstrated outstanding performance during its nearly 10 years of reactor operation and nearly 20 years of plant systems (including the sodium systems) operation. The technology employed in designing and constructing this reactor, as well as information obtained from tests conducted during its operation, can significantly influence the development of new advanced reactor designs in the areas of plant system and component design, component fabrication, fuel design and performance, prototype testing, site construction, reactor operations, irradiated fuel storage, and plant deactivation [1][2][3][4].

The FFTF complex included the reactor, as well as equipment and structures for heat removal, containment, core component handling and examination, irradiated fuel storage, instrumentation and control, and for supplying utilities and other essential services. The FFTF Plant was designed using a "system" concept. All drawings, specifications and other engineering documentation were organized by these systems.

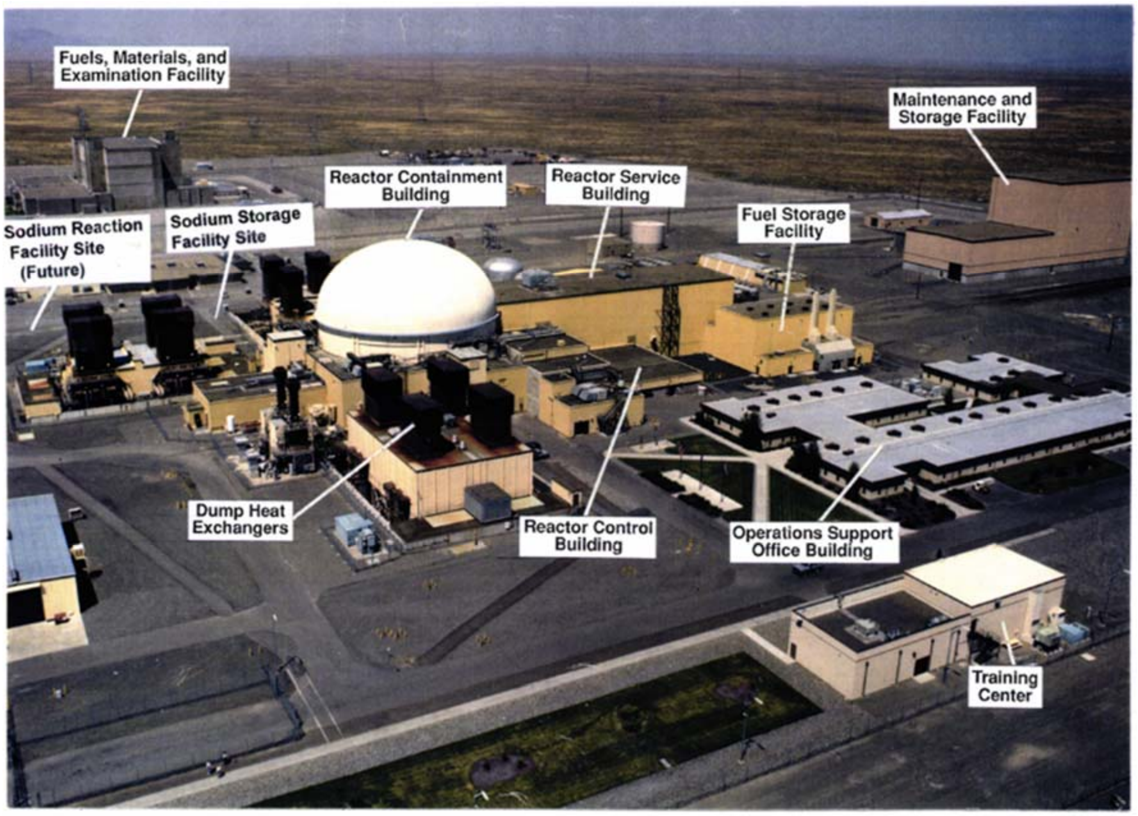

FIG. 1. FFTF Plant

2. Lessons Learned

The following sections summarize the key lessons learned from FFTF operation that have been documented so far.

2.1.FFTF Acceptance/Startup Test Program

Nuclear reactor power plants are extremely complex facilities which require extensive and well-coordinated testing programs to ensure a systematic, rigorous, and comprehensive startup of each plant system to verify that the design, documentation, installation, and operation conforms to the design and safety requirements specified in the System Design Descriptions (SDDs) and the Final Safety Analysis Report (FSAR).

Prior to the start of any acceptance testing at FFTF, construction testing was conducted to ensure that construction was completed in accordance with the drawings and specifications. Typically, construction testing did not require operation of any equipment or performance of any system operations. Successful construction testing was a prerequisite for acceptance of portions of the FFTF for turnover from the construction contractor. Following this turnover, further testing to provide confirmation of design, performance, and operating procedures was performed during the plant acceptance testing period.

FFTF performed and documented a rigorous and successful acceptance and startup testing program between 1978 (official start) and 1982 (official end), with 166 tests being completed by the end of the formal testing period. The testing was divided into five phases for test planning and scheduling purposes: 1) pre-operational tests determined whether components were ready to support subsequent system tests, 2) system startup tests demonstrated that the system design was satisfactory for the test conditions and that each system was ready to support subsequent phases of acceptance testing, 3) hot functional tests demonstrated that overall non-nuclear plant performance was acceptable and that FFTF was ready for the initial loading of fuel, 4) nuclear startup tests demonstrated satisfactory system performance and overall plant operation through initial criticality to reactor power levels of up to 10% of full power, 5) power ascension tests demonstrated satisfactory system performance and overall plant operation at reactor power levels above 10%, up to and including 100% power.

The FFTF acceptance test program is an example of a test program which could be adapted for startup of a future nuclear power plant:

- Acceptance testing must be adequately planned and controlled to be successful.
- Testing must be performed to well written procedures with testing goals and acceptance criteria adequately defined.
- Test results must be documented in a formal report.
- A hierarchy of review and approval of test results must be established to ensure requirements are met.

2.2.Decay Heat Removal and Sodium Natural Circulation

Because of the relative simplicity of the sodium loops, including the DHXs, they were not only used to remove the reactor heat during normal plant operation and shutdown, but they also constituted the designated safety grade reactor decay heat removal system. Small “pony motors” were included on the sodium pumps to provide approximately 10% of the full power sodium flow rate through both the primary and secondary loops during reactor shutdown. The pony motors could be powered by the on-site diesel and gas turbine electrical generators, but these generators were not safety grade (1E) equipment. Instead, the FFTF decay heat removal safety case relied on natural circulation of sodium in both the primary and secondary sodium loops (as well as natural draft air flow through the DHXs).

As a DOE facility, the FFTF did not require and did not receive a license from the Nuclear Regulatory Commission (NRC). However, the NRC and the Advisory Committee on Reactor Safeguards (ACRS) did perform a review of the FFTF and prepared a Safety Evaluation Report (SER) just as they would for a commercial reactor seeking an operating license [5]. The NRC stated that the design features important to safety provided for the FFTF comply with the intent of existing rules and design criteria for licensed reactors. However, this conclusion was based on an assumption of demonstration of the adequacy of natural circulation decay heat removal in the primary and secondary heat transport systems.

The adequacy of natural circulation decay heat removal was verified during the FFTF reactor startup test program which was completed in the early 1980s. Testing began with verification of the natural circulation performance of the secondary sodium loops and DHXs, prior to the start of reactor nuclear operation. Included in these tests was a demonstration of the recovery of natural circulation after it was interrupted by inverting the thermal driving head. After the reactor had achieved criticality, a series of scram to natural circulation tests was performed during the initial power ascent to confirm the adequacy of natural circulation at specified power levels prior to proceeding to higher power. This provided a demonstration of the long term natural circulation behavior and allowed verification of the long term decay heat removal control process under which DHX modules were sequentially removed from service in order to balance plant heat loss and decay heat.

During this natural circulation test series, two Fuels Open Test Assemblies (FOTAs) were included in the core. The FOTAs were highly instrumented test fuel assemblies that included thermocouples incorporated into the wire wrap along essentially the entire length of many of the fuel pins.. One of the FOTAs was located near the center of the core and the other was located closer to the periphery. Together, the FOTAs (along with other plant instrumentation) allowed excellent characterization of the plant performance during the natural circulation testing.

In the early 1980s, inherent (sometimes called passive) safety became a major focus of the nuclear power industry, to a large extent due to the core melt accident at the Three Mile Island plant in 1979. As a result of this new focus on inherent safety, a device called the GEM (Gas Expansion Module) was invented at FFTF to passively insert negative reactivity during a loss of flow event and a series of Loss of Flow Without Scram (LOFWOS) tests was performed to demonstrate the performance of the GEMs. In addition, two precursor natural circulation tests were performed to further characterize the natural circulation performance of the plant.

The primary lesson learned was that because of the importance of the decay heat removal function and the complexity of the thermal hydraulics in some components, the validity of the analytical models needed to be confirmed by tests. While non-reactor tests could have been used to provide high confidence in the predictions, in-reactor tests were required to provide the ultimate demonstration of acceptable performance.

2.3.Secondary Sodium Flow Oscillations

Overall, the main FFTF sodium systems performed with no major problems during the nearly twenty years that they were in operation. One exception was the occurrence of periodic flow and pressure oscillations in the secondary main heat transport system loops. Following an extensive and structured investigation, it was concluded that the oscillations were caused by periodic vortex formation and release from piping tees near the inlets of the DHXs. Several potential concerns associated with these oscillations were evaluated. Potential means of eliminating the oscillations were also investigated. However, it was concluded that the oscillations were acceptable if changes were made to some plant control systems and operating procedures.

There are several lessons learned associated with this occurrence:

- Even what appear to be fairly standard and straight forward system design configurations can result in unexpected behavior and it is important to carefully monitor system performance during initial operation.
- A very thorough and structured evaluation process must be undertaken to assess the impacts of any unexpected operating condition.
- Other plants that use piping tee arrangements, especially those with reducers immediately downstream, should assess the potential for a similar phenomenon during development of the system design.

2.4.Cesium Release from Failed Fuel Pins and Transport within the Plant

Although fuel cladding breaches have been relatively infrequent in sodium cooled reactors, when they do occur, radioactive cesium (^{134}Cs and ^{137}Cs) can be released to the coolant and become a significant source of radiological dose in the plant. Cesium can be effectively removed from the sodium using cesium traps. Because of the relatively high vapor pressure of cesium at reactor operating temperatures, cesium tends to be transported to and throughout the reactor cover gas system. Through proper system design and operation, the levels of cesium contamination and resultant worker dose can be limited to relatively low values.

Although the FFTF fuel performed extremely well, one driver fuel pin and several test pins did experience cladding breaches during the more than ten years of reactor operation. These breaches released relatively small amounts of radioactive cesium into the primary sodium and cover gas systems and this led to a number of operational complications and concerns. A cesium trap was installed in the FFTF primary sodium processing system in 1987 following nearly seven years of reactor operation, including a number of fuel pin failure events. The storage of failed fuel in the two fuel storage vessels [Interim Decay Storage (IDS) and Fuel Storage Facility (FSF)] for many years also resulted in the release of cesium to the sodium in those facilities.

Two key items should be considered in future reactor designs:

- Include a cesium trap to "immediately" remove cesium from the sodium. This certainly applies to the primary (reactor) system and possibly also to any systems that will store failed fuel.
- Include a vapor trap(s) designed to reduce the transport of cesium vapor into the cover gas system.

2.5.Gas Entrainment/Accumulation in Sodium/NaK Systems

The main concern is that gas entrainment and accumulation in the primary coolant could result in a positive reactivity insertion in the reactor if a significant volume were to pass through the core. Although there was no evidence of any significant gas passing through the core at FFTF, there was clear evidence that there was some gas entrained in the primary sodium. This gas entrainment led to a few minor operational concerns and inconveniences.

During the plant design, the possibility for the release of entrained gas in the low velocity primary side of the Intermediate Heat Exchanger (IHX) was recognized. This was a major concern because natural circulation of sodium through the main HTS provided the means of emergency decay heat removal. Gas accumulation in the IHX could potentially interrupt the natural circulation. Therefore, a small sodium line was included to recirculate a small fraction of the total primary sodium flow (and any entrained gas) from the top of the IHX back to the primary pump, where the gas would be released to the cover gas space. During a planned electrical outage during a refueling/maintenance shutdown, one of the IHX recirculation lines froze. This was not expected because the pumps continued to operate on the pony motors and this should have provided sufficient sodium flow to prevent freezing. However, a detailed evaluation of the IHX recirculation line geometry showed that the line was not connected at the highest point of the IHX and that sufficient gas could collect to block the recirculation line at low flow.

Freeze vents were incorporated at the high points of the sodium systems to allow the venting of argon gas during sodium fill and the addition of argon gas during sodium drain. The freeze vents included sodium proximity detectors for use during sodium fill. When sodium was detected at the desired elevation in the freeze vent, the argon valve was closed and electrical heaters on the freeze vent were de-energized to form a very reliable leak tight seal. The freeze vent sodium proximity detectors started alarming after a relatively short time of plant operation, indicating the presence of gas, rather than sodium, in the lower section of the freeze vents.

While this had essentially no impact on plant operation, it was somewhat of a surprise and was another clear indication of gas entrainment in the primary main HTS sodium.

The Fuel Storage Facility (FSF) NaK heat removal loop consisted of a heat exchanger located in the FSF fuel storage vessel, a Natural Draft Heat Exchanger (NDHX) to dump the heat from the NaK loop to the atmosphere, an expansion tank, and the interconnecting piping. During acceptance testing of the first NaK heat removal loop, temporary thermocouples strapped to the outlet of each of the NDHX finned tubes showed that several of the tubes had much lower outlet temperatures than expected and much lower than the other tubes. Flow through the cold tubes was being restricted because of gas trapped in them. Some of the tubes had relatively minor high points in them and these were sufficient to trap enough gas during NaK fill to significantly affect the flow under the low flow natural circulation condition. This problem was resolved by using a small electromagnetic pump in an auxiliary loop to pump the NaK backwards in the loop, thus flushing the gas out of the NDHX to the expansion tank following loop fill.

The main lesson learned was that when designing future LMRs, the potential for and effects of gas entrainment should be minimized.

2.6.Sodium Spills and Fires

Because of the additional hazard associated with the spill and burning of radioactive sodium, the FFTF systems containing radioactive sodium were all contained in cells that were fully lined with steel and normally inerted with nitrogen. Therefore a sodium leak to these cells would result in a very limited sodium reaction (no fire).

The first level of defense is to prevent the occurrence of liquid metal spills. Like most sodium cooled reactor systems, the FFTF systems were constructed primarily of various grades of stainless steel that have been shown to be compatible with liquid metal. All liquid metal systems at FFTF were designed to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, generally either Class I or Class II. The second level of defense against liquid metal leaks is to construct the systems to a very high level of quality. FFTF liquid metal systems were constructed to the requirements of Class I or II of the ASME Boiler and Pressure Vessel Code which required stringent requirements on material certifications, welding procedures, welder qualifications, and post weld inspections. All welds in the liquid metal system boundaries were subject to radiographic inspection. The final level of defense against sodium leaks is operation of the systems within the bounds of the system design and analyses. The FFTF was operated by highly trained personnel using procedures developed to assure that the plant conditions remained within the limits of the design and analysis.

The FFTF sodium systems operated for over twenty years, including more than ten years of reactor power operation. During this time, only one sodium leak (approximately 75 gallons of primary sodium) occurred from a small electromagnetic (EM) pump during a refueling outage. The leak was caused by a combination of freeze and thaw cycles that caused localized deformation of the EM pump duct wall and subsequent operation at cavitation conditions. This led to erosion and eventual failure of the pump duct wall at the location of one of the deformations. Plant operating procedures were subsequently revised to tightly control pump freeze and thaw cycles and to prevent pump cavitation. The leaked sodium was cleaned up and the pump replaced with a spare unit with virtually no impact on the planned duration of the refueling outage. There were no injuries to plant personnel and no impacts to public health and safety.

The experience at the FFTF demonstrated that through proper design, construction and operation of the plant systems, the probability and consequences of liquid metal spills can be limited to acceptably low values.

2.7.Primary System Pressure Drop Increase

FFTF experienced a significant increase in the primary system sodium pressure drop during its early operation. The increase in pressure drop is believed to have been caused by the deposition of silicon based crystals in the fuel assembly inlet orifices, although this was never entirely proven. As the original fuel assemblies were replaced with new assemblies during subsequent refueling outages, the pressure drop returned to near the "normal" value. Fortunately, the FFTF design incorporated sufficient margins to accommodate the pressure drop increase with relatively little impact on plant operations. There are several valuable lessons learned from this occurrence:

- It is important that personnel involved directly in plant operations be familiar with all aspects of laboratory work related to important plant operational characteristics.
- All operating parameters must be monitored very closely during the early operation of a new facility to ensure that the systems are performing as expected.
- Reasonable margins should be incorporated in the design of plant systems and equipment to accommodate unexpected phenomena.
- The introduction of undesirable materials into the plant sodium systems must be minimized to the extent possible.

2.8.Swelling of FFTF Reflector Assemblies

FFTF outer row assemblies consisted of a stack of InconelR-600 blocks penetrated by stainless steel coolant tubes that acted as a radial neutron reflector and as a straight but flexible core boundary. During design, these assemblies were assumed to exhibit low swelling behavior in a neutron flux based on a collection of high nickel alloy data available at the inception of FFTF. However, during an FFTF refueling outage, it was observed that the degree of difficulty in withdrawing an outer row driver fuel assembly was a function of the peak fast fluence of neighboring reflector assemblies. Post irradiation examinations showed that the reflector assemblies were both bowed and stiff. The Inconel-600 blocks had distorted into a trapezoidal cross section due to differential swelling in a steep radial flux gradient. Greater irradiation induced volumetric swelling was found than any previously reported data or correlation for Inconel-600 at equivalent fast fluence.

The main lesson learned from this experience is that this sequence of events highlights the perils of completing a design without sufficient material properties data. In this case, swelling data on the various types of Inconel-600, at varying flux and temperature conditions, were not available when the reflector assemblies were designed and not available to warn of subsequent refueling operations difficulty.

2.9.Thermal Transient Usage

Key plant systems, especially those that play a role in the protection of the health and safety of the public, will typically be designed to the requirements of the ASME Code, including the identification and analysis of transient events which can affect the integrity and lifetime of the associated system components and piping. Selection of the specific events and number of each event (called the "Duty Cycle") is very important. The Duty Cycle must be established to ensure conservatism in the design, yet not so conservative such that the design is driven to be overly complex and expensive.

The only event for which the entire plant accumulated a significant fraction of the design Duty Cycle is the Normal Shutdown transient. The fraction of design transient usage for this event ranges from ~44% for the primary cold legs to ~58% for the Dump Heat Exchanger (DHX) modules. The actual number of Normal Reactor Startup and Reactor Scram transients was much lower than specified in the design (~10% of the design allowance). The FFTF experience should be considered when assigning a reactor scram frequency for future plant designs. Improvements in electronics in general and control systems in particular should make future plants even more reliable than FFTF. The DHX modules generally accumulated thermal transient usage at a higher rate than the rest of the plant. This is largely because FFTF experienced a significant number of DHX control system problems early in plant life. Although the reliability of control systems has certainly improved over the twenty five years since FFTF operated, it is likely that control system failures, such as those associated with the steam generators in a nuclear plant producing electricity, will continue to be a major source of thermal transient usage. The percentage of “Loss of DHX Airflow through All Modules in Loop” was also quite high for the secondary cold legs. Nearly the entire design allotment of Dry Preheat and Dry Cooling transients was consumed for the limiting DHX module (and was fairly high for several other modules). Essentially all of these transients occurred during initial preheat in preparation for sodium fill of the plant. Thermal striping (thermal stresses caused by rapid local temperature fluctuations resulting from the mixing of two flow streams of differing temperature) of the mixing tees at the outlets of the DHX modules was a significant concern and damage to these tees was also tracked by the periodic thermal transient reports. The cumulative damage to the limiting tee (between DHX modules E-6 and E-7) was only 1.1% of the allowable.

The main lesson learned from this topic was that while tracking of the actual thermal transient usage in FFTF showed that the design Duty Cycle was generally selected appropriately, the actual number of some events was approaching the design allowance on some components and was well below it on others.

2.10. Sodium Thermal Stratification

In spite of its very high thermal conductivity, under low flow conditions sodium can stratify, resulting in large vertical thermal gradients and the potential for undesirable thermal stresses. Several instances of thermal stratification were experienced at FFTF. Two types of FFTF pre-operational tests were used to observe mixing patterns in the reactor outlet plenum. These tests were reactor scram from 100% power to pony motor flow and reactor scram from 100% power to natural circulation flow. The temperature profile in the outlet plenum during these tests was measured with thermocouples located at several elevations near the outlet plenum wall. The thermal response of the FFTF to a reactor scram was the subject of extensive analyses using computer design codes which treated thermal transport in the heat transport system piping as one-dimensional uniform flow. The deviation of actual pipe flows from uniform to stratified flows was a concern because it implied non-uniform piping thermal stresses and it could affect thermal head development in low flow (especially natural circulation) transients. During pre-operational testing, transient data for reactor scram to pony motor and natural circulation flow showed that at pony motor flow, flow in the piping was fast (turbulent) enough that stratification did not occur. However, data for the scram to natural circulation showed clear evidence of stratification in the reactor outlet piping. The key Lessons Learned is that sodium thermal stratification has the potential cause significant thermal gradients, thermal transients and resulting stresses. The main lesson learned was that these effects need to be considered in the design and operation of plant systems and components.

2.11. Sodium Pump Flooding/Shaft Bowing/Seizure

During the FFTF Acceptance Test Program the sodium level in two of the three primary pumps was inadvertently raised above the maximum level normally allowed. The elevated pump sodium level caused permanent shaft deformation in one pump due to the hot sodium preferentially traveling up one side of the shaft-to-support cylinder annulus and this pump was replaced shortly after the event occurred. The pump replacement activity was performed without major difficulty (a spare pump was available), the damaged pump shaft was straightened and the refurbished pump was returned to the plant for use as a spare. The second pump was restored to operation with no indication of significant adverse effects. However, this pump seized approximately three years later while operating at pony motor speed. It was concluded that the seizure was caused by the relocation of sodium deposits that had existed in the pump shaft annulus since the time of the overfilling event. Following extensive evaluation and exercising of the pump, it was again restored to service and it continued to operate without further problems throughout the remainder of plant life (more than 20 years). This series of events provides several valuable Lessons Learned:

- Great care must be taken to avoid moving sodium into undesirable locations during manipulation of plant systems and equipment. Sodium level changes are particuarly sensitive to changes in cover gas pressure.
- While some impacts of inadvertent sodium movement may be immediately obvious, others may remain hidden for an extended period of time.
- The pump replacement and repair experience provide confidence in the ability to replace and repair major sodium wetted plant equipment

2.12. Sodium Vapor Trap Design and Operation

An inert gas, typically argon, is used as a cover gas over the sodium in the reactor vessel and other components in Liquid Metal Reactors (LMRs). Due to the high sodium temperature, the cover gas may become saturated with sodium vapor and contain significant quantities of sodium aerosol. Sodium vapor traps are installed in the cover gas discharge piping to remove most of this sodium before the gas is sent to the downstream processing and analysis systems and eventually discharged. This sodium removal is especially important in the reactor cover gas where the sodium is radioactive. Although operation of the sodium vapor traps at FFTF was mostly uneventful, some problems were experienced with one trap during early operation. Following extensive analysis and testing, the operating parameters and philosophy for this trap were modified and all of the traps operated without further problems throughout the life of the plant. The key Lessons Learned are:

- The design and operation of the vapor traps need to recognize the potential for and impacts of significant quantities of sodium aerosol (not just vapor) in the cover gas.
- It can be advantageous to use alternate (perhaps portable/temporary) cover gas monitoring equipment that bypasses the vapor traps during periods of reactor shutdown.
- As historically designed and operated, sodium vapor traps are not very effective at removing cesium from the reactor cover gas. Since cesium from failed fuel can be transported into and contaminate the cover gas systems, changes to vapor trap operating

conditions should be evaluated to improve their cesium removal efficiency or consider the addition of separate cesium vapor traps.

2.13. Deactivation of Primary Loop Isolation Valves

The FFTF included large sodium isolation valves in the hot and cold legs of each of the three primary Reactor Heat Transport System loops. These valves were originally included in the plant design primarily to allow the plant to be operated at nominally two thirds power with only two loops available. However, concerns related to the potential loss of natural circulation capability in the loops due to gas in-leakage in case of melt out of the freeze seals on the valve stems led to deactivation of the valves. The key lesson learned related to this topic are:

- The potential impacts of equipment failures on plant operation, and especially on the performance of safety related functions, must be fully understood and carefully evaluated during the design of the plant.
- Incorporation of features intended to allow plant operation to continue with a loop completely out of service will likely be very expensive and have limited value.

3. Conclusions

The FFTF was highly successful and demonstrated outstanding performance during the 10 years of reactor operation and nearly 20 years of plant system (including the sodium systems) operation. The technology employed in designing and constructing this reactor, as well as information obtained from tests conducted during its operation, can significantly influence the development of new advanced reactor designs in the areas of plant system and component design, component fabrication, fuel design and performance, prototype testing, site construction, reactor operations, irradiated fuel storage, and plant deactivation.

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