4.0 TIME-LIMITED AGING ANALYSES

This section of the FCS License Renewal Application (LRA) deals with the identification and evaluation of Time-Limited Aging Analyses (TLAAs). TLAAs capture certain plant-specific aging analyses that are explicitly based on the current operating term of the plant. In 10 CFR 54.3, TLAAs are defined as noted below.

Time-limited aging analyses, for the purposes of this part, are those licensee calculations and analyses that:

(1) Involve systems, structures, and components within the scope of license renewal, as delineated in §54.4(a);

(2) Consider the effects of aging;

(3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;

(4) Were determined to be relevant by the licensee in making a safety determination;

(5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in § 54.4(b); and

(6) Are contained or incorporated by reference in the Current Licensing Basis (CLB).

The Statements of Consideration (SOC) accompanying 10 CFR 54 clarify the definition of a TLAA by explaining that an analysis is relevant if it “provides the basis for the licensee’s safety determination and, in the absence of the analysis, the licensee may have reached a different safety conclusion.” (60 FR 22480)

10 CFR 54 requires that a list of TLAAs (as defined above) be provided in the LRA, including a demonstration that one of the following resolutions (from § 54.21(c)(1)) is true for each TLAA:

(i) The analyses remain valid for the period of extended operation;

(ii) The analyses have been projected to the end of the period of extended operation; or

(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

In addition, 10 CFR 54 requires that license exemptions granted pursuant to §50.12, in effect, and based on TLAAs be identified and analyzed to confirm their validity for the period of extended operation.

4.1 IDENTIFICATION OF TIME LIMITED AGING ANALYSES

4.1.1 PROCESS OVERVIEW

Potential TLAAs were identified first through a search of regulatory and industry literature such as:

- The Statements of Consideration (SOC) for 10 CFR 54,
- NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, April 2001, and
- NEI 95-10, Industry Guidelines for Implementing the Requirements of 10 CFR 54 – The License Renewal Rule, Revision 3, March 2001.

Additional potential TLAAs were identified through reviews of other industry license renewal applications. Finally, a search of the FCS CLB (including licensing documents and the USAR), as well as Design Basis Documents, was performed using a full-text searchable electronic docket to identify any analyses that may contain additional, FCS-specific TLAAs. No new potential TLAAs were identified by this search. The search capabilities were also used to verify details of the applicability of the generic TLAAs to FCS and to support the conclusion that a particular TLAA did not apply to FCS.

These potential TLAAs were screened to determine if they met the definition presented in 10 CFR 54.3. Any that applied to FCS are addressed in Table 4.1-1.

4.1.2 IDENTIFICATION OF EXEMPTIONS

10 CFR 54.21(c)(2) also requires that an applicant for license renewal provide a list of all exemptions granted under 10 CFR 50.12 which are determined to be based on a TLAA. These TLAA-based exemptions must be evaluated and justification provided for the continuation of the exemption during the period of extended operation.

FCS exemptions were identified through a search of the FCS electronic docket. Each exemption was then reviewed for TLAA applicability. No TLAA-based exemptions were identified for FCS.

TABLE 4.1-1
TIME-LIMITED AGING ANALYSES APPLICABLE TO FCS

TLAA Category	Analysis	§ 54.21(c)(1) Resolution
Reactor Vessel Neutron Embrittlement	Pressure/Temperatures (P/T) Curves	(ii) The analyses will be projected to the end of the period of extended operation (4.2.1)
	Low Temperature Overpressure Protection (LTOP) PORV Setpoints	(ii) The analyses will be projected to the end of the period of extended operation. (4.2.2)
	Pressurized Thermal Shock (PTS)	(ii) The analyses have been projected to the end of the period of extended operation. (4.2.3)
	Reactor Vessel Upper Shelf Energy	(ii) The analyses will be projected to the end of the period of extended operation. (4.2.4)
Metal Fatigue	ASME III, Class 1 (vessels) RCS Piping	(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation. (4.3.1)
	Pressurizer Surge Line Thermal Stratification	(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation. (4.3.3)
	Fatigue of Class II and III Components (excluding NSSS Sampling)	(i) The analyses remain valid for the period of extended operation. (4.3.4)
	NSSS Sampling	(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation. (4.3.4)
Environmental Qualification	EQ of Electrical Equipment	(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation. (4.4)
Concrete Containment Pre-Stress	Containment Tendon Pre-stress	(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation. (4.5)

TABLE 4.1-1 (CONTINUED)
TIME-LIMITED AGING ANALYSES APPLICABLE TO FCS

TLAA Category	Analysis	§ 54.21(c)(1) Resolution
Containment Liner	Containment Liner Plate and Penetration Sleeve Fatigue	(ii) The analyses will be projected to the end of the period of extended operation. (4.6)
Other TLAAs	Reactor Coolant Pump Flywheel Fatigue	(i) The analyses remain valid for the period of extended operation. (4.7.1)
	Leak Before Break (LBB) Analysis for Resolution of USI A-2	(ii) The analyses will be projected to the end of the period of extended operation. (4.7.2)
	High Energy Line Break	(i) The analyses remain valid for the period of extended operation. (4.7.3)

4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT

There are four analyses affected by irradiation embrittlement that have been identified as TLAAs, and these issues are addressed specifically for FCS in Sections 4.2.1 through 4.2.4:

- Pressure/Temperatures (P/T) Curves
- Low Temperature Overpressure Protection (LTOP) PORV Setpoints
- Pressurized Thermal Shock (PTS)
- Reactor Vessel Upper Shelf Energy

This group of TLAAs concerns the effects of neutron embrittlement and how this mechanism affects analyses, limits, and programs that provide operating restrictions or support regulatory requirements for the reactor plant. Reactor pressure vessel embrittlement is generally greater for pressurized water reactors (PWRs) than for boiling water reactors (BWRs). BWR vessels experience less neutron irradiation and therefore less embrittlement. FCS uses a “low leakage” PWR core design that reduces the number of neutrons that reach the vessel wall and thus limits the vessel's embrittlement. However, the rate at which the vessel's steel embrittles also depends on its chemical composition. The amounts of two elements in the steel, copper and nickel, are the most important chemical components in determining how sensitive the steel is to neutron irradiation.

Neutron embrittlement is a significant aging mechanism for all ferritic materials that have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV) at the end of the period of extended operation. The relevant calculations use predictions of the cumulative damage to the reactor vessel from neutron embrittlement, and were originally based on the 40 year expected life of the plant. The reactor pressure vessel contains the core fuel assemblies and is made of thick steel plates that are welded together. Neutrons from the fuel in the reactor irradiate the vessel as the reactor is operated and change the material properties of the steel. The most pronounced and significant changes occur in the material property known as fracture toughness. Fracture toughness is a measure of the resistance to crack extension in response to stresses. A reduction in this material property due to irradiation is referred to as embrittlement. The largest amount of embrittlement usually occurs at the section of the vessel’s wall closest to the reactor fuel referred to as the vessel’s beltline.

10 CFR 54.29(a) provides that a renewed license may be issued if “actions have been identified and have been or will be taken with respect to … (2) time-limited aging analyses that have been identified to require review under §54.21(c).” The analyses addressed in Sections 4.2.1 through 4.2.4 will be updated in a timely manner, either as indicated or as needed to continue plant operation in accordance with OPPD’s formal process for managing commitments.

OPPD will disposition the reactor vessel neutron embrittlement analyses for the period of extended operation in accordance with §54.21(c)(1)(ii). (References 4.2-1, 4.2-2, 4.2-3)

4.2.1 PLANT HEATUP/COOLDOWN (PRESSURE/TEMPERATURE) CURVES

The impact properties of all steel materials that form a part of the pressure boundary of the reactor coolant system were determined in accordance with the requirements of the ASME Code Section III. The operating stress limits for those materials in the reactor coolant system other than the reactor vessel are the same as those for the reactor vessel. Shortly after plant startup, the integrated neutron flux results in the reactor vessel being the controlling component for loss of fracture toughness.

Steel's fracture toughness also depends on its temperature, and this limits the pressure and temperature envelope in which the reactor can safely operate. During design, the impact properties of all steel materials that form a part of the pressure boundary of the reactor coolant system were determined in accordance with the requirements of the ASME Code Section III. After startup, the operating stress limits for the reactor vessel became the controlling component. Appendix G to 10 CFR 50 requires that P-T limits be established during all phases of reactor operation and that thermal stresses be limited by determining maximum heatup and cooldown rates. Heatup and cooldown rates are determined such that the resulting stress intensity does not exceed the material reference critical stress intensity factor K_{IC}. The material reference critical stress intensity factor is a function of the actual temperature minus RT_{NDT} (Reference nil ductility temperature). This temperature was calculated at the beginning of vessel life for the unirradiated state and it increases as fast neutrons irradiate the vessel. Since the reactor vessel’s steel is less susceptible to crack growth and is more ductile at higher temperatures, calculating a transition temperature guarantees a margin of fracture toughness at or above that temperature. Over the life of the reactor vessel, the transition temperature gradually increases, so it is necessary to reduce the allowable pressure to reduce the total stress.

The current pressure/temperature analyses are valid out to 40 effective full power years, which extends beyond the current operating license period but not to the end of the period of extended operation. The Technical Specifications will continue to be updated as required by either Appendices G or H of 10 CFR 50, or as operational needs dictate. This will assure that operational limits remain valid for current and projected cumulative neutron fluence levels. Therefore, the analyses will be projected to the end of the period of extended operation.

4.2.2 LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) PORV SETPOINTS

Low temperature overpressure protection limits are considered as part of the calculation of pressure/temperature curves. Loss of ductility at low temperatures due to irradiation must be evaluated during the period of extended operation, so LTOP considerations are included in the analyses described in Section 4.2.1. Therefore, the analyses will be projected to the end of the period of extended operation.

4.2.3 PRESSURIZED THERMAL SHOCK (PTS)

10 CFR 50.61 addresses another issue related to embrittlement and thermal stress called Pressurized Thermal Shock (PTS). During design transients, cold water injected into the vessel causes the vessel to cool rapidly and generates large thermal stresses in the steel. These stresses combine with the high internal pressure to create a fracture potential which could damage the pressure vessel. Irradiation makes the vessel's beltline more susceptible to cracking during a pressurized thermal shock event. The parameter describing this fracture potential is called the transition temperature or RT_{PTS} and it corresponds to the nil ductility reference temperature for the most limiting beltline material. It is a function of the projected fluence values and is calculated using guidance in Regulatory Guide 1.99, revision 2. Applicants are obligated to project the values of the increasing transition temperature into the period of extended operation.

OPPD has completed the projected calculation (Reference 4.2-4), and the NRC has concluded that RT_{PTS} for the FCS reactor vessel will remain below the 10 CFR 50.61 PTS screening criteria until 2033, the end of the period of extended operation (Reference 4.2-5). Therefore, the analyses have been projected to the end of the period of extended operation.

4.2.4 REACTOR VESSEL UPPER SHELF ENERGY

The NRC regulations that provide screening criteria for the increase in the transition temperature also address the decrease in a parameter called the “upper shelf energy.” Upper shelf energy is a measure of fracture toughness at temperatures above RT_{PTS} when the vessel is exposed to additional radiation. The screening criteria for the increase in transition temperature are found in 10 CFR 50.61. The screening criterion for the decrease in upper shelf energy is found in 10 CFR 50, Appendix G.

Preliminary calculations have shown that the vessel beltline Charpy upper-shelf energy for the limiting weld will be approximately 54.6 ft-lbs based on position 1.2 of RG 1.99. This value remains above the regulatory approved minimum of 50 ft-lbs through the period of extended operation. The existing Appendix G analysis will be finalized and formally revised to reflect that it bounds the minimum approved fluence value at the end of plant life. Therefore, the analyses will be projected to the end of the period of extended operation.

4.3 METAL FATIGUE

There are four distinct issues considered separately under the TLAA for Metal Fatigue and these issues are addressed for FCS in Sections 4.3.1 through 4.3.4:

- Reactor Coolant and associated systems thermal fatigue,
- Environmentally Assisted Fatigue
- Pressurizer Surge Line Thermal Stratification, and
- Fatigue of Class II and III components.

Fatigue is the gradual deterioration of a material that is subjected to repeated cyclic loads. Components have been designed or evaluated for fatigue according to the requirements of the codes listed in Table 4.3-1: (Reference 4.3-1, Section 4.2)

TABLE 4.3-1
FCS REACTOR COOLANT SYSTEM CODE REQUIREMENTS

Component	Code
Reactor Vessel	ASME III, Class A
Steam Generators Primary Side	ASME III, Class A
Steam Generator Secondary Side	ASME III, Class A
Pressurizer	ASME III, Class A
Coolant Pumps (Design Basis)	ASME III, Class A
Pressurizer Safety and Relief Valves	ASME III
RCS Loop Piping (Pressure Design)	USAS B31.1
RCS Loop Piping (Fatigue Design)	USAS B31.7 Draft

4.3.1 REACTOR COOLANT AND ASSOCIATED SYSTEM FATIGUE

The reactor coolant loop piping and fittings were designed and fabricated in accordance with the requirements of USAS B31.1, *Power Piping Code*, including all requirements of Code Cases N-9 and N-10. The exception is the centrifugally cast stainless steel pipe, which was supplied in accordance with ASTM A451-72 specifications in lieu of the ASTM A451-63 specifications listed in Case N-9. The reactor coolant loop attached piping was designed and fabricated in accordance with the requirements of USAS B31.7, *Draft Code for Nuclear Power Piping*. The fatigue analysis was performed for both the RCS loops and attached piping in accordance with the USAS B31.7, *Draft Code for Nuclear Power Piping,* using the design cyclic transients identified below for normal and abnormal transients.

The following design cyclic transients include conservative estimates of the operational requirements for the components listed in Table 4.3-1, and were used in the fatigue analyses required by the applicable codes: (Reference 4.3-1, Section 4.2)

- 500 heatup and cooldown cycles at a heating and cooling rate of 100 deg F/hr.
- 15,000 power change cycles over the range of 10 percent to 100 percent of full load with a ramp load change of 10 percent of full load per minute increasing or decreasing.
- 2,000 cycles of 10 percent of full load step power changes increasing from 10 percent to 90 percent of full power and decreasing from 100 percent to 20 percent of full power.
- 10 cycles of hydrostatic testing the reactor coolant system at 3125 psia and at a temperature at least 60 deg F above the Nil Ductility Transition Temperature (NDTT) of the limiting component.
- 200 cycles of leak testing at 2100 psia and at a temperature at least 60 deg F greater than the NDTT of the reactor vessel.
- 1,000,000 cycles of operating variations of +100 psi and +6 deg F from the normal operating pressure and temperature.
- 400 reactor trips when at 100 percent power.

In addition to the above list of normal design transients the following abnormal transients were also considered when arriving at a satisfactory usage factor as defined in Section III of the ASME Boiler and Pressure Vessel Code.

- 40 cycles of loss of turbine load with delayed reactor trip from 100 percent power.
- 40 cycles of total loss of reactor coolant flow when at 100 percent power.
- 5 cycles of loss of secondary system pressure.

Each steam generator was also designed for the following conditions such that no component is stressed beyond the allowable limit as described in the ASME Boiler and Pressure Vessel Code, Section III:

- 4000 cycles (2,000 each direction) of transient pressure differentials of 85 psi across the primary head divider plate due to starting and stopping the reactor coolant pumps.
- 10 cycles of secondary side hydrostatic testing at 1235 psig while the primary side is at 0 psig.
- 200 cycles of secondary side leak testing at 985 psig while the primary side is at 0 psig.
- 5,000 cycles of adding 1000 gpm of 70 deg F feedwater with the plant in hot standby condition.
- 80 cycles of adding 300 gpm of 32 deg F feedwater with the plant in hot standby condition.

Certain additional design transients were also considered in arriving at a satisfactory usage factor as defined in Section III of the ASME Boiler and Pressure Vessel Code.

- 8 cycles of adding a maximum of 300 gpm of 32 deg F feedwater, with the steam generator secondary side dry and at 600 degrees F.

The following additional design cyclic and abnormal transients were used in the fatigue analysis required by the applicable design codes for certain components within the CVCS (Reference 4.3-1, Section 9.2.1.1):

- 1000 cycles of Maximum Purification.
- 8000 cycles of Boron Dilution.
- 80 cycles of Low Volume Control Tank Level.
- 500 cycles of Loss of Charging.
- 700 cycles of Loss of Letdown.
- 200 cycles of Long Term Letdown Isolation (in excess of 1 hour).
- 700 cycles of Short Term Letdown Isolation (up to 1 hour).
- 200 cycles of Intermittent Manual Charging (significant only for charging nozzles).

The unit is capable of withstanding these conditions for the prescribed numbers of cycles in addition to the prescribed operating conditions without exceeding the allowable cumulative usage factor.

The steam generators, pressurizer and reactor coolant pumps were designed and fabricated to the requirements of the 1965 edition of the ASME Boiler and Pressure Vessel Code Section III through and including the 1966 Summer Addenda. The reactor vessel was designed and fabricated to the requirements of Section III through and including the 1967 Winter Addenda. This code requires fatigue analyses and dictates design requirements with conservative design cycles that preclude the development of fatigue cracks during the design cycle life of the plant. Fatigue usage factors were derived for limiting critical components during the original plant design process that became the bases for the Technical Specifications.

Additionally, the thickness of the reactor coolant pipe and fittings met the requirements of ASME Section III through and including the Winter 1967 agenda, and a stress analysis similar to the requirements of ASME Section III was performed. Other reactor coolant pressure boundary piping and fittings including the pressurizer safety and relief valve discharge piping were designed and fabricated in accordance with the draft code for nuclear power piping (August 1968). In 1984, the safety and relief valve discharge piping was reclassified under USAS B31.1, *Power Piping Code*. Code Cases N-2 and N-10 to USAS B31.1 were applied to valves in the reactor coolant boundary.

Plant operating experience has shown that there are large margins between the magnitude and frequency of the actual and the design operating cycles. Design operating cycles are monitored and logged by plant staff. Many of the transients described above have very few or no recorded cycles. The following transients have no recorded cycles:

- 5 cycles of loss of secondary system pressure.
- 5,000 cycles of adding 1000 gpm of 70 deg F feedwater with the plant in hot standby condition.
- 8 cycles of adding a maximum of 300 gpm of 32 deg F feedwater, with the steam generator secondary side dry and at 600 deg F.

The following transients have only one cycle recorded corresponding to initial plant testing:

- 10 cycles of hydrostatic testing the reactor coolant system at 3125 psia and at a temperature at least 60 deg F above the Nil Ductility Transition Temperature (NDTT) of the limiting component.
- 200 cycles of secondary side leak testing at 985 psig while the primary side is at 0 psig.

For the cycles that are counted, the total count tabulation and count trends have been reviewed and none are projected to exceed design limits during the period of extended operation. Consequently, the use of the conservatism in the original design code permits the extension of code fatigue analyses into the period of extended operation. OPPD will establish a program to verify these conclusions. Therefore, the effects of aging will be adequately managed for the period of extended operation.

4.3.2 ENVIRONMENTALLY ASSISTED FATIGUE

Generic Safety Issue (GSI) 190 [Reference 4.3-2] was initiated by the NRC staff because of concerns about the potential effects of reactor water environments on reactor coolant system component fatigue life during the period of extended operation. GSI-190 was closed in December 1999 [Reference 4.3-3] and concluded that environmental effects have a negligible impact on core damage frequency, and as such, no generic regulatory action is required. However, as part of the closure of GSI-190, NRC concluded that licensees who apply for license renewal should address the effects of coolant environment on component fatigue life as part of their aging management programs [Reference 4.3-4].

Fatigue calculations that include consideration of environmental effects to establish cumulative usage factors could be treated as time-limited aging analyses (TLAAs) under 10 CFR Part 54, or they could be utilized to establish the need for an aging management program. In other words, the determination of whether a particular component location is to be included in a program for managing the effects of fatigue, and the characteristics of that program, should incorporate reactor water environmental effects.

An analysis must satisfy all six criteria defined in 10 CFR 54.3 to qualify as a TLAA. Failure to satisfy any one of these criteria eliminates the analysis from further consideration as a TLAA. Fatigue design analysis for FCS has been determined to be a TLAA, even though the design limits are based on cycles rather than an explicit time period. However, reactor water environmental effects, as described in GSI-190, are not included in the FCS current licensing basis (CLB), such that the criterion specified in 10 CFR 54.3(a)(6) is not satisfied. Nevertheless, environmental effects on Class 1 component fatigue have been evaluated separately for FCS to determine if any additional actions are required for the period of extended operation.

The FCS approach to address reactor water environmental effects accomplishes two objectives, as illustrated in Figure 4.3.2-1. First, the TLAA on fatigue design has been resolved by confirming that the original transient design cycles remain valid for the 60-year operating period (See Section 4.3.1 on Class 1 Metal Fatigue). Confirmation by the Fatigue Monitoring Program will ensure these transient design cycles are not exceeded. Second, reactor water environmental effects on fatigue life are examined using the most recent data from laboratory simulation of the reactor coolant environment. These two aspects of fatigue design are kept separate, since fatigue design for FCS is part of the plant CLB and a TLAA, while the consideration of reactor water environmental effects on fatigue life, as described in GSI-190, is not considered part of the FCS CLB.

It is important to note that there are three areas of margin included in the FCS Fatigue Monitoring Program (B.2.5) that are worthy of consideration. These areas include margins resulting from actual cycle experience, cycle severity, and moderate environmental effects.

<u>Margin Due to Actual Cycles</u>: It has been concluded that the original 40-year design cycle set for Class 1 components is valid for the 60-year extended operating period. Conservative projections conclude that the design cycle limits will not be exceeded. Additional margin is available in the current Class 1 component fatigue analyses since the cumulative fatigue usage factors for all Class 1 components remain below the acceptance criteria of 1.0.

<u>Margin Due to Transient Severity:</u> Much of the conservatism in the fatigue analysis methodology is due to design cycle definitions. It has been concluded that the severity of the original FCS design cycles bound actual plant operation Additional industry fatigue

studies [References 4.3-5, 4.3-6, 4.3-7, 4.3-8] conclude that the fatigue impact of conservative design basis cycle definitions by themselves overwhelms the contributing impact of reactor water environmental effects.

Margin Due to Moderate Environmental Effects: A portion of the safety factors applied to the ASME Code Section III fatigue design curves includes moderate environmental effects. While there is debate over exactly the amount of margin this represents, it is noteworthy to recognize this safety factor in this qualitative discussion of margin.

Considering the three margins above, the FCS Fatigue Monitoring Program is conservative from an overall perspective. Nevertheless, specific assessments of potential environmental effects have been addressed.

Idaho National Engineering Laboratories (INEL) evaluated in NUREG/CR-6260 [Reference 4.3-9] fatigue-sensitive component locations at plants designed by all four U. S. nuclear steam supply system (NSSS) vendors, as a part of the industry effort to address environmental effects for operating nuclear power plants during the current 40-year licensing term. The pressurized water reactor (PWR) calculations included in NUREG/CR-6260, especially the "Older Vintage Combustion Engineering Plant," closely matches FCS with respect to the design codes used. Additionally, the evaluated design cycles considered in the evaluation match or bound the FCS design.

The fatigue-sensitive component locations chosen in NUREG/CR-6260 for the older vintage Combustion Engineering plant were:

- Reactor vessel shell and lower head
- Reactor vessel inlet nozzle
- Reactor vessel outlet nozzle
- Surge line elbow
- Charging system nozzle
- Safety Injection System nozzle
- Shutdown Cooling System Class 1 piping

NUREG/CR-6260 calculated fatigue usage factors for these locations utilizing the interim fatigue curves provided in NUREG/CR-5999 [Reference 4.3-10]. However, the data included in more recent industry studies [References 4.3-11 and 4.3-12] need to be considered in the evaluations of environmental effects. Environmental fatigue calculations have been performed for FCS for those component locations included in NUREG/CR-6260 using the appropriate methods contained in NUREG/CR-6583 for carbon/low alloy steel material, or NUREG/CR-5704 for stainless steel material, as appropriate. Based on these results, all component locations were determined to be acceptable for the period of extended operation, with the exception of the pressurizer surge line (specifically the surge line elbow below the pressurizer). The pressurizer surge line elbow requires further evaluation for the period of extended operation.

OPPD has selected aging management to address pressurizer surge line fatigue during the period of extended operation, in lieu of performing additional analyses to refine the fatigue usage factors for the pressurizer surge line. In particular, the potential for crack initiation and growth, including reactor water environmental effects, will be adequately managed during the extended period of operation by the continued performance of the FCS ASME Section XI, Inservice Inspection Program.

The FCS surge line is a 10-inch schedule 160 line connected to the pressurizer surge nozzle and to the hot leg surge nozzle. The surge line contains 18 welds. A sample of these surge line welds is currently examined every 10 years in accordance with the requirements of the ASME Section XI, Subsection IWB. Surge line welds selected for the inservice examinations, by nature of their size, require a volumetric examination, in addition to a surface examination. A number of the surge line welds have been examined ultrasonically during inservice examination intervals at FCS as part of the current ASME Section XI program, including inspections on the pressurizer surge line elbow welds. No indications have been identified.

The limiting pressurizer surge line welds will continue to be inspected during the third and fourth ISI intervals and prior to the license renewal period. The results of those inspections will be utilized to assess continuation of the current 10 year inspection interval for continued use throughout the remaining operating period.

The proposed aging management program to address fatigue of the FCS pressurizer surge lines during the period of extended operation is similar to the approach documented in the ASME Boiler and Pressure Vessel Code, Section XI - *Rules for Inservice Inspection of Nuclear Power Plant Components, Non-mandatory Appendix L*. However, OPPD recognizes that to date, the NRC has not endorsed the Appendix L approach. The primary NRC concerns with Appendix L include crack aspect ratio and acceptable fatigue crack growth rates (including environmental effects).

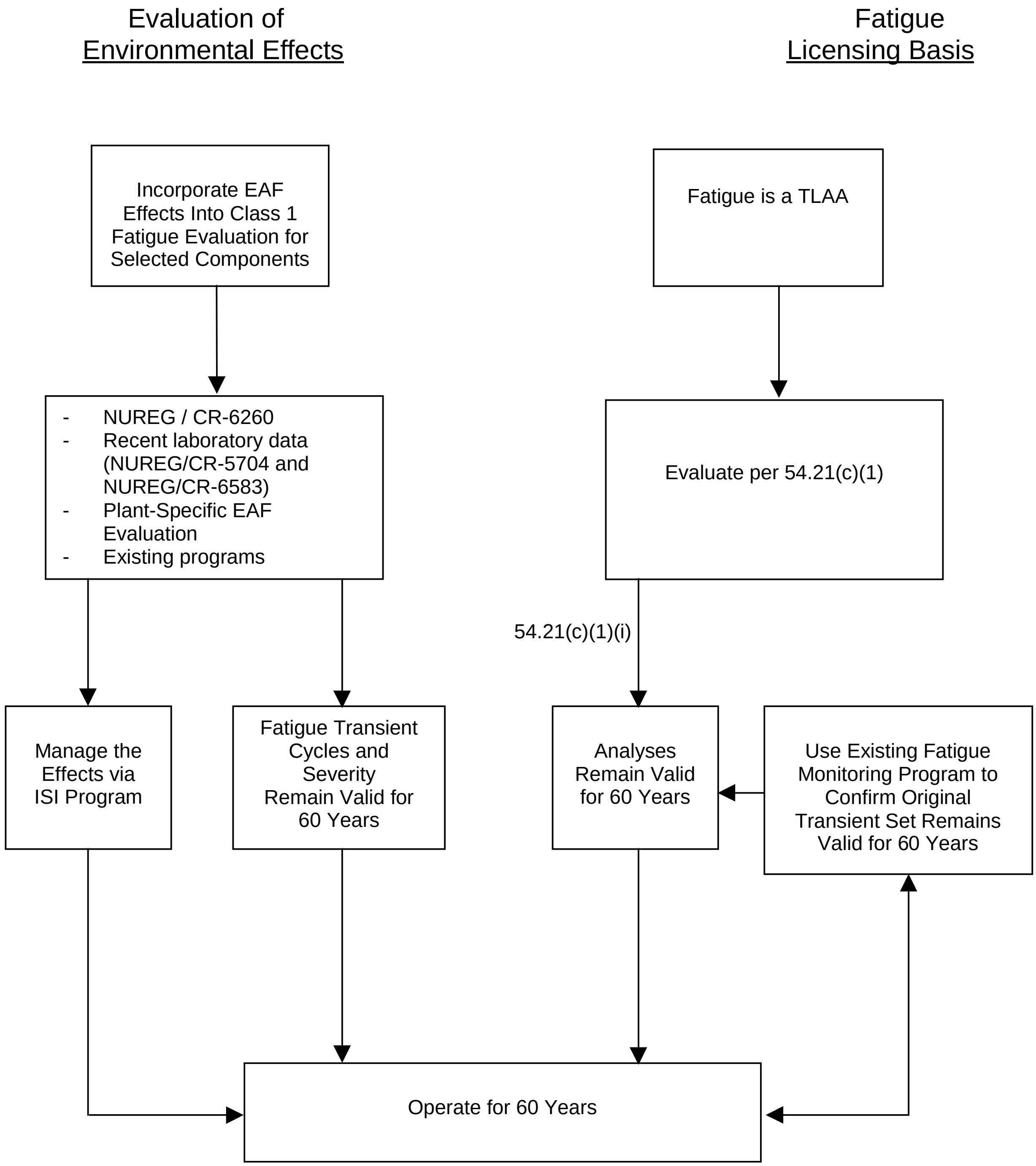
As noted above, several pressurizer surge line welds have been ultrasonically examined. No reportable indications have been identified. In addition, OPPD plans to inspect the limiting surge line welds during the third and fourth inservice inspection interval, and prior to entering the extended period of operation. The results of these inspections will be utilized to assess the appropriate approach for addressing environmentally-assisted fatigue of the surge lines. The approach developed could include one or more of the following:

- Further refinement of the fatigue analysis to lower the CUF(s) to below 1.0, or
- Repair of the affected locations, or
- Replacement of the affected locations, or
- Manage the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC).

Should OPPD select Option 4 (i.e., inspection) to manage environmentally-assisted fatigue during the period of extended operation, inspection details such as scope, qualification, method, and frequency will be provided to the NRC prior to entering the period of extended operation.

The OPPD position to address the effects of environmentally assisted fatigue meets the requirements specified in the NRC closure of GSI-190. The position takes a proactive approach by performing volumetric and surface examinations of the most fatigue sensitive locations and the pressurizer surge line elbow welds, during both the current period of operation and the period of extended operation. The commitment to inspect the fatigue sensitive surge line locations in accordance with the ASME Section XI, Inservice Inspection Program provides reasonable assurance that potential environmental effects of fatigue will be managed such that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

FIGURE 4.3.2-1
GSI-190 EVALUATION PROCESS



4.3.3 PRESSURIZER SURGE LINE THERMAL STRATIFICATION

Pressurizer surge line thermal stratification is an issue raised by NRC Bulletin 88-11. One of the requirements of this Bulletin was to analyze the effects of this mechanism on the stress and fatigue calculations for the surge line. The generic and bounding analysis for all CE plants was performed by CE and submitted to the NRC. The fatigue portion of this analysis calculated a. 937 usage factor for the surge line after the 40 year design life. This value is based on the use of the most limiting configuration of the surge line for a CE-designed plant and as a result is very conservative for FCS. To address this issue for the purposes of license renewal, the pressurizer surge line bounding locations will be included in the Fatigue Monitoring Program (B.2.5). This program will compile realistic usage factors for the critical areas which are expected to be lower than those predicted by the generic evaluation. This usage factor will be determined from actual plant operating data to include the effects of thermal stratification. This reevaluation will take place prior to the period of extended operation. Therefore, realistic fatigue usage for the surge line will be tracked, and actions will be taken to reevaluate, repair, or replace the surge line before a fatigue-induced failure occurs. The effects of aging will be adequately managed for the period of extended operation.

4.3.4 FATIGUE OF CLASS II AND III COMPONENTS

The design code for Class II and III Components at FCS is the Draft Code for Nuclear Power Piping USAS B31.7. USAS B31.7 requires the design for Class II and III piping to meet the requirements of USAS B31.1 1965. The USAS B31.1 requires that a conservatively determined stress range reduction factor of 1.0 be used during the original plant design for up to 7000 equivalent full power cycles. While no calculations or analyses meeting the definition of a TLAA were identified for this issue, the fatigue of Class II and III components will conservatively be treated as a TLAA. The 7000 cycle limit could only be reached if the piping system endured the equivalent of a full temperature cycle approximately once every 3 days. Practical experience with plant operation has demonstrated that the design cycle limit will not be reached during the period of extended operation. The existing analyses will remain valid through the period of extended operation for all Class II and III systems except one.

The only exception to the 7000 cycle limit is the NSSS sampling system. Normal sampling from the RCS hot leg results in cyclical thermal stresses whenever the RCS is above ambient conditions. Over a 60-year period of operation, the 7000 full cycle limit would be reached with only an average of 2 cycles per week. Samples at FCS are typically taken approximately three times per week. The affected portions of the NSSS Sampling System will be included in the Fatigue Monitoring Program (B.2.5). Therefore, the effects of aging will be adequately managed for the period of extended operation.

4.4 ENVIRONMENTAL QUALIFICATION (EQ)

4.4.1 BACKGROUND

10 CFR 50.49, *Environmental qualification of electric equipment important to safety for nuclear power plants*, requires that safety related electrical equipment, important to safety, that is relied upon to remain functional during and following a design basis event be environmentally qualified to perform its' intended function. Additionally, any non-safety related electrical equipment whose failure, under postulated environmental conditions, could prevent satisfactory accomplishment of safety functions should be qualified. Post accident monitoring equipment relied upon by the operators to take actions to mitigate the consequences of a postulated event should also be qualified to ensure that the operators have reliable data and are not misled. For the period of extended operation, EQ is a TLAA affecting all equipment in the scope of the EQ program having a qualified life value of 40 years or greater, but less than 60 years, whether active or passive.

To establish reasonable assurance that the safety related electrical equipment will perform its safety function when exposed to postulated harsh environmental conditions, licensees are required to develop an environmental qualification program. The program must demonstrate that the safety related electrical equipment required to perform the various safety related functions, identified in 10 CFR 50.49, are qualified to perform as intended. The program must maintain the environmental qualification of the equipment for its installed life. Periodic replacement and/or refurbishment of equipment are performed in order to maintain the qualified life of the device. The qualified life of an equipment type is that period of time the equipment is installed, under normal and abnormal plant operating conditions (thermal and radiation exposure), and still be expected to perform its intended function following a postulated design basis event. The qualified life of an equipment type is determined utilizing the ambient environmental conditions to which it is exposed for the predicted installation period as well as any internal heat rise and cyclic stresses. The qualified life of an equipment type can be affected by changes in plant design and operating conditions; on this basis, the qualified life of an equipment type is frequently revisited to determine if any changes have occurred which could potentially affect the life of the equipment. Recalculations of qualified life as well as updates to equipment performance characteristics are performed under the current EQ program. Activities, such as equipment upgrade and qualified life, will continue during the period of extended operation and appropriate changes will be implemented as required by evolving regulatory requirements.

4.4.2 PROGRAM DESCRIPTION

The FCS Electrical Equipment Qualification (EEQ) Program has been established to implement the requirements of the EQ Rule, 10 CFR 50.49. The program provides for necessary procedural controls, ensuring that appropriate and timely changes are reflected. The FCS EEQ Program addresses the effects of aging to ensure that the required electrical equipment function is maintained and qualified throughout its' installed life.

The FCS EEQ Program accomplishes the following to meet the requirements delineated in 10 CFR 50.49:

- Reviews original qualified life bases;
- Establishes margin/uncertainty limits for qualified life;
- Reviews available aged specimen test data for impact on and validation of margin/ uncertainty;
- Reviews any data for impact on and validation of margin/uncertainty;
- Adjusts qualified life based on consideration of analytical and test data, and refurbishment without violating the qualification margin/uncertainty limits;
- Establishes new replacement dates for qualified equipment based on emergent issues, new data, industry experience, etc., as appropriate in and accordance with plant and 10 CFR 50.49 program procedures.

4.4.3 EQ CALCULATIONS AND CONSIDERATIONS FOR LICENSE RENEWAL

10 CFR 50.49 requires that all significant effects from normal service conditions be considered. This would include the expected thermal aging effects from normal temperature exposure, any radiation effects during normal plant operation, and cycle aging. The evaluation of the environmental service conditions for the period of extended operation requires a re-evaluation of the aging effects to determine whether the equipment or item can continue to support the intended pre-accident service (40 to 60 years) while continuing to maintain the capability to perform its post-accident intended function.

4.4.3.1 THERMAL AGING CONSIDERATIONS

Existing analyses for thermal aging of all equipment within the FCS EQ Program will be reviewed to determine if the existing calculations remain valid for the period of extended operation, or if additional analysis will be required to demonstrate qualification through the period of extended operation.

4.4.3.2 RADIATION CONSIDERATIONS

The total integrated dose (TID) for the 60 year period will be established by making the assumption that it is equal to 1.5 times the normal operating dose for 40 years (i.e., 60 years / 40 years = 1.5). The 60-year TID will then be compared to the qualification level to ensure that the required TID was met or enveloped. If the required TID calculated by this methodology is higher than the qualification value, the component group or part will require assessment, prior to the “end of life date,” in accordance with EQ program requirements.

4.4.3.3 MECHANICAL CYCLE CONSIDERATIONS

The evaluation of the period of extended operation will address mechanical cycle-aging requirements for EQ equipment. In the absence of more specific information, an assumption will be made that the multiplier for normal cycles for the license renewal period would be 1.5 times the cycles assumed in the current 40 year analysis (i.e., 60 years / 40 years = 1.5). If the device was previously qualified for this number of cycles no additional review was required. If the number of normal cycles by this methodology is higher than the qualification value then the component group or part will require assessment prior to its “end of life” date in accordance with EQ program requirements.

4.4.3.4 EQ GENERIC SAFETY ISSUE (GSI) 168 FOR ELECTRIC COMPONENTS

Since Environmental Qualification is a TLAA for license renewal, outstanding GSIs that could affect the validity of any credited analyses must be dispositioned as part of the application process. GSI-168 remains unresolved and states,

> … the staff reviewed significant license renewal issues and found that several related to environmental qualification (EQ). A key aspect of these issues was whether the licensing bases, particularly for older plants whose licensing bases differ from newer plants, should be reassessed or enhanced in connection with license renewal or whether they should be reassessed for the current license term. The staff concluded that differences in EQ requirements constituted a potential generic issue which should be evaluated for back-fit independent of license renewal.
>
> …the staff reviewed tests of qualified cables … to determine the effects of aging on cable products used in nuclear power plants. After accelerated aging, some of the environmentally-qualified cables either failed or exhibited marginal insulation resistance during accident testing, indicating that qualification of some electric cables may have been non-conservative. … [T]he test results raised questions with respect to the EQ and accident performance capability of certain artificially-aged cables. Depending on the application, failure of these cables during or following design basis events could affect the performance of safety functions in nuclear power plants.

For the purpose of license renewal, there are three options for resolving issues associated with a GSI:

- If the issue is resolved before the renewal application is submitted, the applicant can incorporate the resolution into the application.
- An applicant can submit a technical rationale which demonstrates that the CLB will be maintained until some later point in the period of extended operation, at which time one or more reasonable options would be available to adequately manage the effects of aging.

- An applicant could develop a plant-specific aging management program that incorporates a resolution to the aging issue.

OPPD has chosen to pursue the second approach, so until GSI-168 is resolved, aging management of such cables will continue to be addressed through plant-specific programs. At that time, one or more reasonable options should be available to adequately manage the effects of aging.

4.4.4 CONCLUSION

The FCS EEQ Program is consistent with X.E1 Environmental Qualification (EQ) of Electrical Components as identified in NUREG-1801.

The FCS EEQ Program has been demonstrated to be capable of programmatically managing the qualified lives of EQ components within the scope of license renewal. The NRC has determined that the EEQ Program is an acceptable program to address environmental qualification in accordance with 10 CFR 54 (Reference 4.4-1). As part of the CLB, the FCS EEQ Program will provide for extension of the qualification to the end of the period of extended operation. Therefore, the effects of aging on the intended functions will be adequately managed for the period of extended operation. Program revisions will be made as appropriate to accommodate changes to the licensing basis, regulatory requirements, and resolutions of generic safety issues.

4.5 CONCRETE CONTAINMENT TENDON PRE-STRESS

The pre-stress on the containment tendons decreases over plant life as a result of elastic deformation, creep and shrinkage of concrete, anchorage seating losses, tendon wire friction, stress relaxation and corrosion. The cylindrical walls and dome are post-tensioned to the extent that the internal pressure produced by the applicable DBE would be more than balanced by the pre-stress forces. In addition, conventional, bonded reinforcing steel was provided to resist local moments and shears at penetrations and discontinuities, and to distribute strain due to shrinkage of concrete and temperature effects. The containment wall and dome were pre-stressed by means of unbonded post-tensioned tendons. Pre-stressing tendon integrity is monitored and confirmed by the Containment Inservice Inspection Program (B.1.2). The program provides for tendon inspection 1, 2 and 4 years after initial pre-tensioning, and every five years thereafter for the remaining life of the plant. The pre-stressing tendon surveillances are performed in accordance with NRC Regulatory Guide 1.35 revision 3 (Reference 4.5-1), as implemented in Amendment 139 to the FCS operating license. Curves showing anticipated variation of tendon force with time, together with the lower limit curves to be applied to surveillance readings are shown in the FCS USAR. The curves are given in terms of net force in the tendon and as a percentage of the initial tendon load. The calculated pre-stress at end of plant life exceeds by a reasonable margin the intensity required to meet the design criteria. This margin is the basis of the limits set for deviation with time of the tendon forces as measured by the periodic lift-off readings. If at any time surveillance testing indicates a decrease in the tendon force below the given limit line, corrective action will be taken in accordance with the Technical Specifications.

The USAR curves will be extended to 60 years of plant life to cover the period of extended operation. This will also show that the pre-stressing force is acceptable for continued service at the end of the period of extended operation considering the assumed time dependent nature of pre-stress losses. The tendon surveillance program will be continued into the period of extended operation using the updated curves. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.6 CONTAINMENT LINER PLATE AND PENETRATION SLEEVE FATIGUE

The containment liner and penetration sleeves were designed to be essentially leak-tight under all postulated loading combinations by limiting strains to those values that have been shown to result in leak-tight pressure vessels. The ASME Boiler and Pressure Vessel Code, Section III, *Nuclear Vessels*, was employed as a guide in the determination of acceptable strains. At penetrations, the liner is thickened to minimize stress concentrations and to reduce the possibility of local welding distortions. The liner reinforcement at all penetrations meets the requirements of the ASME Boiler and Pressure Vessel Code, Section III, *Class B Vessels*. Penetration design and materials conform to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, *Nuclear Vessels*. The piping is anchored at the penetration sleeves, and the anchorage restraint forces and moments were included in the design of the sleeve anchorage. The temperature of the penetration sleeve at its attachment weld to the liner does not exceed 150 deg F under operating conditions. Sleeve radiation fins, thermal sleeves, and pipe insulation were employed to maintain this temperature requirement.

Fatigue considerations were of prime importance and the fatigue loadings assumed for the design of the liner and attachments were:

- Thermal cycling caused by one loss-of-coolant accident;
- Thermal cycling caused by variation of annual outdoor temperatures (40 cycles for the plant life of 40 years); and
- Thermal cycling caused by variation of internal temperature between shutdown and operating conditions (500 cycles for the plant life of 40 years).

Liner plate/penetration sleeve fatigue is therefore a TLAA for license renewal. The results of the containment fatigue analysis indicated that when the maximum compressive strain in the liner was reached under operating conditions and subsequent cyclic temperature variations were applied to the liner, there was no significant change in stress and strain in concrete or steel for the second cyclic load indicating that shakedown had occurred during the first cycle of loading. Also, the investigation for 500 cycles of loading for the liner steel, anchor steel and anchor welds resulted in a computed cumulative usage factor of 0.05 as compared with an allowable usage factor of 1.0 (Reference 4.3-1, Section 5.6). Consideration of 60 years of operation as opposed to 40 would have no relevant impact on these results. However, the observed buckling of the liner is slightly larger than was assumed in the original analyses. The original analysis for the liner assumed a 1/16 inch inward curvature between stiffeners. The actual bulges are estimated to be from 1/4 inch to 3/4 inch. Strains resulting from thermal cycling may be greater than the original analysis resulting in an increased fatigue usage factor. This condition has been evaluated and found adequate for the current term. OPPD will complete an analysis considering the actual bulges for a 60-year life. The analysis will be completed before the beginning of the period of extended operation. Therefore, the analysis will be projected to the end of the period of extended operation.

4.7 OTHER TLAAS

4.7.1 REACTOR COOLANT PUMP FLYWHEEL FATIGUE

4.7.1.1 GENERAL ELECTRIC RCP FLYWHEELS

General Electric manufactured the original RCP motors. Each GE pump motor is provided with a flywheel that reduces the rate of flow decay upon loss of pump power. Conservative design bases and stringent quality control measures have been taken to preclude failure of the flywheel. The following design features ensure that the requirements for structural soundness were met:

- Division of the mass into three separate discs;
- A keyway fillet radius not less than 1/8 inch was used to minimize stress concentrations;
- Fabrication of the discs using forged carbon steel plate having different tensile strengths.

The resistance to rupture of the reactor coolant pump flywheels has been examined at 120 percent overspeed. Using fracture mechanics data furnished by the motor vendor, the critical crack length for the disc most susceptible to crack propagation was found to be 3 inches assuming the crack extended radially outward from the keyway and penetrated completely through the thickness of the disc. Using the crack growth prediction techniques described in Reference 4.7-1, the conclusion was that over 185,000 complete cycles from zero to 120 percent overspeed would be required to cause a 1/2 inch long crack extending radially from the keyway to grow to critical size.

This number of cycles will not be exceeded if the licensing period is extended to 60 years. To do so would require in excess of 8 pump starts per day, which far exceeds actual and projected pump use. Since the cycle limit will not be exceeded, the analysis for the General Electric produced RCP flywheels remains valid for the period of extended operation.

4.7.1.2 ABB MOTOR FLYWHEEL

During the 1996 refueling outage, the reactor coolant pump RC-3B motor was replaced with a motor manufactured by ABB Industries. The replacement motor was designed, manufactured and tested per the guidance of RG 1.14, Rev. 1, *Reactor Coolant Pump Flywheel Integrity*. The flywheel is a single piece design made from forged ASTM A508 4/5 steel and shrink fitted to the shaft collar. The flywheel is conservatively designed and made with closely controlled quality material such that the probability of a flywheel failure is sufficiently small; therefore, a steel shroud was not included in the flywheel design. A crack growth analysis was performed by ABB, which demonstrated that critical flaw growth would not occur with fewer than 10,000 complete cycles from zero to 120 percent overspeed.

This number of cycles will not be exceeded if the licensing period is extended to 60 years. To do so would require approximately 1 pump start every 2 days, which far exceeds actual and projected pump use. Since the cycle limit will not be exceeded, the analysis for the ABB produced RCP flywheel remains valid for the period of extended operation.

4.7.2 LEAK BEFORE BREAK (LBB) ANALYSIS FOR RESOLUTION OF USI A-2

In response to USI A-2, Westinghouse attempted to eliminate consideration of primary loop pipe breaks from plant design bases. In 1981, OPPD participated in the Westinghouse Owners Group effort since the material similarity of the Reactor Coolant System at FCS is closer to plants of Westinghouse design than it is to other CE plants. The focus of the evaluation was whether or not a postulated crack which is assumed to appear instantaneously in plant piping will become unstable and lead to a full circumferential break when subjected to the worst possible combinations of plant loading. This evaluation showed that double-ended breaks of reactor coolant pipes are unrealistic and, as a result, large LOCA loads on primary system components will not occur. The resulting report was issued before the NRC began requiring LBB analyses to consider thermal aging of piping.

There are two TLAA aspects to LBB, crack growth and thermal aging. While transient cycle fatigue crack growth is a TLAA for FCS and also a design consideration, thermal aging was not evaluated for FCS by either the original design code or the LBB analysis. Consequently, OPPD will perform a plant specific LBB analysis prior to the period of extended operation. This analysis will consider a 60-year life and thermal aging effects of the CASS RCS and will be competed before the beginning of the period of extended operation. Therefore, the analysis will be projected to the end of the period of extended operation.

4.7.3 HIGH ENERGY LINE BREAK (HELB)

The High Energy Line Break (HELB) analysis (Reference 4.3-1, Appendix M) is a potential TLAA because postulated fatigue cumulative usage factors (CUFs) based on

40 years of operation may be used as screening criteria to determine piping locations that require further analysis regarding the effects of pipe ruptures outside Containment. For FCS, the Main Steam (MS) and Main Feedwater (MFW) systems contain piping for which CUFs have been evaluated for screening.

Fatigue analyses were previously performed, for the B31.7 Class I portions of MS and MFW outside Containment, to identify locations with cumulative usage factors greater than 0.1 as one of the criteria for selecting postulated break locations. Reevaluating these CUFs for the period of extended operation could impact the assumed number of cycles used in those analyses and potentially cause additional locations to exceed a usage factor of 0.1, which could require postulating additional break locations. The Class I portions encompass the piping from the Containment penetrations to the first isolation valves outside Containment. Since these Class I portions of the MS and MFW systems are the only piping runs at FCS for which the CUF screening criteria could be applicable, they are the only piping potentially affected by this TLAA.

For the Class I MFW piping, all locations with CUFs greater than 0.1 for 40 years were also selected as break locations based on stresses exceeding the other selection criteria of $2S_m$ (Reference 4.3-1, Appendix M, Attachment B). These locations occur at each end of each segment, bounding the nodes for which CUFs were less than 0.1. The Class I portions of MFW outside Containment are wrapped in steel “barrel slat” enclosures to prevent lateral pipe movement and the formation of longitudinal and axial jets, which could impact nearby structures and equipment. Pipe whip restraints are installed to limit pipe movement due to circumferential breaks within these segments. The consequences of a break at an intermediate node, not previously selected as a break location, are bounded by the consequences of the breaks assumed at the ends. Therefore, projection of the CUFs for the period of extended operation will not require either any additional pipe break analysis to be performed or hardware to be installed on the Class I portions of MFW outside Containment.

For the Class I MS piping, there were no locations with CUFs greater than 0.1 for 40 years, but there were many locations where stresses exceeded $2S_m$ (Reference 4.3-1, Appendix M, Attachment A). The Class I portions of MS outside Containment are wrapped in steel “barrel slat” enclosures to prevent lateral pipe movement and the formation of longitudinal and axial jets, which could impact nearby structures and equipment. Pipe whip restraints are installed to limit pipe movement due to circumferential breaks within these segments. The consequences of a break at any node, not previously selected as a break location, are bounded by the consequences of the breaks previously assumed. A potential exception considered were the piping connections to the isolation valves, as it is has not been evaluated whether the slats extend far enough beyond these nodes to prevent movement induced by a circumferential break at those locations from pulling free of the slat enclosures. However, the CUFs at these nodes are less than 0.001 and projecting them to account for the period of extended operation will not result in exceeding the 0.1 screening criteria. Therefore, projection of the CUFs for the period of extended operation will not require any additional pipe break analysis to be performed or hardware to be installed on the Class I portions of MS outside Containment.

The circumferential breaks, already postulated, are bounding for all nodes with respect to direction and magnitude of force. Consideration of the period of extended operation will not impact the selection of the bounding locations. The barrel slats, which cover the piping segments, restrain longitudinal movements and jets along the length of the Class I pipe, not just at the postulated break points. In summary, projection of the CUFs used as HELB screening criteria for the period of extended operation will not require any additional pipe break analysis to be performed or hardware to be installed on the Class I piping. The CUFs are in fact not part of the actual analysis, but only represent screening criteria used to select bounding locations. Therefore, the analysis remains valid for the period of extended operation.

4.8 REFERENCES

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4.2-4 Fort Calhoun Operating License DPR-40 and Technical Specifications.

4.2-5 Alan B. Wang (USNRC) to S.K. Gambhir (OPPD), “Fort Calhoun Station, Unit No. 1 – Issuance of Amendment – Deletion of Section 3.D, “License Term” (TAC No. MA9690)”, dated June 7, 2001

4.3-1 Fort Calhoun Station Updated Safety Analysis Report

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4.3-4 Letter from Dana A. Powers (Chairman, ACRS) to Dr. William D. Travers (Executive Director for Operations, USNRC), "Proposed Resolution of Generic Safety Issue-190, Fatigue Evaluation of Metal Components for 60-Year Plant Life," December 10, 1999.

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