3.0 AGING MANAGEMENT REVIEW

For those structures and components identified as subject to an aging management review, 10 CFR 54.21(a)(3) requires demonstration that the effects of aging will be adequately managed so that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. This section describes the results of the aging management reviews of the components and structures identified in Section 2, *Scoping and Screening Methodology for Identifying Structures and Components Subject to Aging Management Review, and Implementation Results.*

The aging management reviews were conducted by:

1. Identifying the materials and environments of these structures and components;

2. Determining the applicable aging effect(s) requiring management; and

3. Assigning the appropriate aging management program to those components and structures with materials and environments that were determined to be subject to an aging effect requiring management.

The results of each mechanical and structural aging management review is documented as a unique set of component(s) or subcomponent(s), made of a material, exposed to an environment, with an Aging Effect Requiring Management (AERM), managed by an Aging Management Program (AMP). This unique set of

- component(s) or subcomponent(s)
- material
- environment
- AERM
- AMP

is defined as a FCS Aging Management Group (AMG). The aging management review results for systems, structures, or component groupings are made up of several AMGs.

Three types of aging management review results are discussed in this section of the application. The first of these are the FCS AMGs that credit AMPs evaluated in NUREG-1801. To identify those FCS AMGs that credit AMPs evaluated in NUREG-1801, each FCS AMG was compared to the NUREG-1801, Volume 2 aging management review results using the process documented below.

> FCS aging management review results were classified as being consistent with NUREG-1801 if the comparison between each FCS AMG and a single row from the tables in NUREG-1801, Volume 2 met the following criteria.

1. The FCS AMG environment and AERM are determined to be the same, using engineering judgment, as the environment and AERM documented in NUREG-1801, Volume 2.

2. The FCS AMG material is the same as or similar to (using engineering judgment) the material documented in NUREG-1801, Volume 2.

3. The FCS AMP is determined to be the same, using engineering judgment, as the AMP documented in NUREG-1801, Volume 2; or NUREG-1801, Volume 2 specifies a plant specific AMP.

FCS AMG aging management review results were classified as consistent with NUREG-1801 with deviation if the comparison between the FCS AMG and a single row from the tables in NUREG-1801, Volume 2 met criterion 1 and 2 above, and the FCS AMP deviates from one or more of the acceptance criteria for the AMP documented in Chapters 10 and 11 of NUREG-1801, Volume 2.

The Aging Management Review results for FCS AMGs that credit AMPs evaluated in NUREG-1801 are reported in Tables 3.x.1 of sections 3.1 through 3.6. The process used to develop these tables is described below.

The component, aging effect/mechanism, aging management programs and further evaluation recommended columns from Table 3.x.1 of NUREG-1800 were copied from NUREG-1800 for those rows applicable to a PWR.

A discussion column was added to the four columns. Where applicable, the following information was entered in the discussion column:

- A statement that the FCS AMGs are consistent with NUREG-1801, that the FCS AMGs are consistent with NUREG-1801 with deviation(s), or that the components, materials and environments identified in NUREG-1801 are not applicable to FCS
- Identification of the FCS AMP when NUREG-1801 specifies a plant specific program; the applicable Appendix B section is also identified
- A discussion of the materials and environments included in the FCS AMGs that are consistent with the materials and environments reported in NUREG-1801
- If necessary, a description of component(s) in the FCS AMGs that is not included in NUREG-1801
- If necessary, a description of material(s) in the FCS AMGs that is not included in NUREG-1801

In Table 3.1-1, additional detail on the comparison of components between the FCS AMGs and the line items from NUREG-1801, Volume 2 is included because of the level of component detail provided in Chapter IV of NUREG-1801, Volume 2. In Table 3.6-1, discussions of a FCS specific AMP and modification are included for electrical cables and connectors not subject to 10 CFR 50.49 EQ requirements.

The second type of aging management review result discussed in Sections 3.1, 3.2, 3.3, 3.4, and 3.5 of the application are the FCS AMGs that do not credit AMPs evaluated in NUREG-1801. These aging management review results are reported in Tables 3.x.2 of Sections 3.1, 3.2, 3.3, 3.4, and 3.5. The entries in Tables 3.x.2 were developed by identifying components with the same material, environment, AERM and AMP, and entering these results as a single row in the table. The AERM column of Tables 3.x.2 includes a discussion of the applicable aging mechanisms for the AERM. The applicable Appendix B section is also identified for each AMP.

The third type of Aging Management Review results discussed in this section of the application includes the components replaced on the basis of performance or condition. The performance or condition monitoring programs to ensure functionality during the period of extended operation are discussed in Section 3.3 of this application.

3.1 AGING MANAGEMENT OF REACTOR COOLANT SYSTEMS

The FCS reactor coolant systems evaluated in this section of the application consist of the Reactor Coolant System, the Reactor Vessel and the Reactor Vessel Internals and associated components.

The Reactor Coolant System consists of two heat transfer loops connected in parallel to the reactor vessel. Each loop contains one steam generator, two reactor coolant pumps, connecting piping and instrumentation. A pressurizer is connected to one of the reactor vessel outlet (hot leg) pipes by a surge line. All components of the Reactor Coolant System are located within the Containment Building.

The Reactor Vessel is a 140-inch beltline inner diameter two-loop vessel. This configuration has four coolant inlet nozzles and two coolant outlet nozzles. The vessel includes a removable head with multiple penetrations (control element drive mechanisms, in-core instrumentation nozzles, and the reactor vessel vent line). The vessel includes two leakage detection lines. The vessel is an all welded, manganese molybdenum-nickel steel plate and forging construction. The interior surfaces of the vessel in contact with reactor coolant are clad with austenitic stainless steel.

The Reactor Vessel Internals are designed to support and align the fuel assemblies, control element assemblies (CEAs), and in-core instrumentation (ICI) assemblies, and to guide reactor coolant through the reactor vessel. The components of the Reactor Vessel Internals consist of the upper guide structure, core support barrel, thermal shield, core shroud, CEA shroud assemblies, ICI assemblies, lower support structure, and flow skirt.

Operating Experience:

Site: A review of plant specific operating experience was conducted, including the review of Condition Reports and discussions with appropriate site personnel to identify AERM. These reviews concluded that the AERM identified by the FCS specific operating experience were consistent with those identified in NUREG-1801.

Industry: A review of industry-wide operating experience was conducted to identify aging effects requiring management. This included a review of operating experience issued during 2001. This review concluded that the AERM identified by industry operating experience were consistent with those identified in NUREG-1801.

On-Going: The on-going review of plant specific and industry-wide operating experience is conducted in accordance with the FCS Operating Experience Program.

3.1.1 AGING MANAGEMENT PROGRAMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

Table 3.1-1 shows the aging management programs evaluated in NUREG-1801 that are relied on for license renewal of the Reactor Coolant System at FCS. Note that this table only includes those components, materials and environments that are applicable to a PWR. Information on FCS specific components and materials, not listed in NUREG-1801 but included in the component group of this application, is included in the discussion column.

TABLE 3.1-1
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Reactor coolant pressure boundary components	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The metal fatigue time limited aging analyses are discussed in Section 4.3. 3. Consistent with NUREG-1801, this group includes the low alloy steel and carbon steel with stainless steel cladding, stainless steel, CASS, and nickel alloy in borated treated water; and low alloy steel in deoxygenated water and steam at FCS. 4. In addition to the components in NUREG-1801, this group includes pressure boundary components associated with the in core detectors, upper guide structure fasteners, and in-core instrument supports at FCS.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
				5. In addition to the materials in NUREG-1801, this group includes Alloy 600 control rod drive head penetrations, Alloy 600 cladding in areas of internal vessel attachments, Alloy 600 flow skirt, Alloy 600 cladding on the low alloy steel primary side head of the steam generators, and Combustion Engineering welded and mechanical Alloy 690 tube plugs at FCS. 6. This group does not include core shroud tie rods since they are not part of the FCS core shroud; fuel alignment pins since they are part of the FCS fuel assemblies; the pressurizer spray head and pressurizer quench tank because they are not within the scope of license renewal at FCS. 7. Cumulative fatigue damage is not an aging effect requiring management for control element assembly shroud bolts and core support barrel snubber assembly socket head cap screws. These components are preloaded to prevent fatigue cycles.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Steam generator shell assembly	Loss of material due to pitting and crevice corrosion	Inservice inspection; water chemistry	Yes, detection of aging effects is to be further evaluated	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. This aging effect is managed by the Inservice Inspection Program (B.1.5), the Chemistry Program (B.1.1) and the Steam Generator Program (B.1.7). The Steam Generator Program includes methods to detect general, crevice and pitting corrosion discussed in NUREG-1801, Volume 2, IV.D1.1-c. These programs are described in Appendix B of this application. 3. Consistent with NUREG-1801, this group includes carbon steel in deoxygenated treated water at FCS.
Pressure vessel ferritic materials that have a neutron fluence greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of fracture toughness due to neutron irradiation embrittlement	TLAA, evaluated in accordance with Appendix G of 10 CFR 50 and RG 1.99	Yes, TLAA	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The reactor vessel neutron embrittlement time limited aging analyses are discussed in Section 4.2. 3. Consistent with NUREG-1801, this group includes low alloy steel with stainless steel cladding in borated treated water at FCS. 4. The safety injection nozzles are not included in this group. These components are included with the RCS piping for FCS.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Reactor vessel beltline shell and welds	Loss of fracture toughness due to neutron irradiation embrittlement	Reactor vessel surveillance	Yes, plant specific	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. This aging effect is managed by the Reactor Vessel Integrity Program (B.1.6). This program is described in Appendix B of this application. 3. Consistent with NUREG-1801, this group includes low alloy steel with stainless steel cladding in borated treated water at FCS. 4. The safety injection nozzles are not included in this group at FCS. These components are included with the RCS piping.
Westinghouse and B&W baffle/former bolts	Loss of fracture toughness due to neutron irradiation embrittlement and void swelling	Plant specific	Yes, plant specific	This item is not applicable since FCS is a Combustion Engineering designed and manufactured reactor.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Small-bore reactor coolant system and connected systems piping	Crack initiation and growth due to SCC, intergranular SCC, and thermal and mechanical loading	Inservice inspection; water chemistry; one-time inspection	Yes, parameters monitored/inspected and detection of aging effects are to be further evaluated	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. This aging effect is managed by the Inservice Inspection Program (B.1.5), the Chemistry Program (B.1.1) and the One-Time Inspection Program (B.3.5). These programs are described in Appendix B of this application. The One-Time Inspection Program verifies that service-induced weld cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch connections. 3. Consistent with NUREG-1801, this group includes stainless steel in borated treated water at FCS.
Vessel shell	Crack growth due to cyclic loading	TLAA	Yes, TLAA	Underclad crack growth due to cyclic loading was not identified as a TLAA for FCS.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Reactor internals	Changes in dimension due to void swelling	Plant specific	Yes, plant specific	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801 with the exception noted in item 7 below. 2. This aging effect is managed by the Reactor Vessel Internals Inspection Program (B.2.9). This program is described in Appendix B of this application. 3. Consistent with NUREG-1801, this group includes the stainless steel in borated treated water at FCS. 4. In addition to the components in NUREG-1801, this group includes the thermal shield, the in-core instrument tubes and supports, and the flow skirt at FCS. 5. In addition to the materials in NUREG-1801, this group includes Alloy 600 at FCS. 6. This group does not include core shroud tie rods since they are not part of the FCS core shroud, and does not include the fuel alignment pins since they are part of the fuel assemblies.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
				7. Changes in dimension due to void swelling are not an aging effect requiring management for some reactor internals components because the intended function of the component is not affected. As noted in the Safety Evaluation Report Related to the License Renewal of Arkansas Nuclear One, Unit 1, the specific impacts of concern for void swelling are constriction of flow paths, interference with control rod insertion and excessive baffle bolt loading. This is consistent with NUREG-1705, *Safety Evaluation Report Related to the License Renewal of Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (Final Report)*, and NUREG-1723, *Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2 and 3*. Swelling of certain components does not impact the noted concerns.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
				These components are the core support barrel alignment key, core support barrel fasteners, core support barrel locking collar, core support barrel spacer, core support barrel upper flange, upper guide structure alignment lug (NUREG-1801 FAP guide lug), upper guide structure fasteners, upper guide structure guide pins, hold down ring, upper guide structure locking strip, upper guide structure plate (a support for an instrument tube), upper guide structure shim ring, upper guide structure tab, thermal shield positioning pins and screws, thermal shield pins, thermal shield shim, lower internals anchor block, lower internals fasteners, lower vessel internals dowel pins, core shroud fasteners, control element assembly shroud nuts and bolts, in-core instrumentation guide tubes (above instrumentation support plate), in-core instrumentation guide tube fasteners, incore instrumentation support plate and gussets.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
PWR core support pads, instrument tubes (bottom head penetrations), pressurizer spray heads, and nozzles for the steam generator instruments and drains	Crack initiation and growth due to SCC and/or primary water stress corrosion cracking (PWSCC)	Plant specific	Yes, plant specific	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. This aging effect is managed by the Alloy 600 Program (B.3.1). This program is described in Appendix B of this application. 3. Consistent with NUREG-1801, this group includes Alloy 600 in borated treated water at FCS. The vessel flange leak detection line at FCS is made of Alloy 600. 4. In addition to the components in NUREG-1801, this group includes surveillance capsule holders, pressurizer temperature nozzle, the pressurizer nozzle welds, the pressurizer thermal sleeves, and the pressurizer nozzle flanges flange leak detection line at FCS. 5. This group does not include bottom head instrument tubes because FCS does not have bottom head instrument tubes, and does not include the pressurizer spray head because it is not within the scope of license renewal at FCS.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Cast austenitic stainless steel (CASS) reactor coolant system piping	Crack initiation and growth due to SCC	Plant specific	Yes, plant specific	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. This aging effect is managed by the Chemistry Program (B.1.1), the Inservice Inspection Program (B.1.5) and the Thermal Embrittlement of Cast Austenitic Stainless Steel Program (B.2.11). These programs are described in Appendix B of this application. 3. Consistent with NUREG-1801, this group includes cast austenitic stainless steel (CASS) reactor coolant system piping at FCS. 4. The surge line is not included in this group at FCS because it is made of stainless steel (not CASS).

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Pressurizer instrumentation penetrations and heater sheaths and sleeves made of Ni-alloys	Crack initiation and growth due to PWSCC	Inservice inspection; water chemistry	Yes, AMP for PWSCC of Inconel 182 weld is to be evaluated	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. This aging effect is managed by the Alloy 600 Program (B.3.1), Chemistry Program (B.1.1) and Inservice Inspection Program (B.1.5). These programs are described in Appendix B of this application. The Alloy 600 Program manages the AERM of PWSCC in Inconel 182 welds. 3. Consistent with NUREG-1801, this group includes Alloy 600 and nickel alloys in borated treated water at FCS. 4. In addition to the components in NUREG-1801, this group includes steam generator primary nozzle welds at FCS.
Westinghouse and B&W baffle former bolts	Crack initiation and growth due to SCC and IASCC	Plant specific	Yes, plant specific	This item is not applicable since FCS is a Combustion Engineering designed and manufactured reactor.
Westinghouse and B&W baffle former bolts	Loss of preload due to stress relaxation	Plant specific	Yes, plant specific	This item is not applicable since FCS is a Combustion Engineering designed and manufactured reactor.
Steam generator feedwater impingement plate and support	Loss of section thickness due to erosion	Plant specific	Yes, plant specific	The components identified in NUREG-1801, Volume 2, IV.D1.1-e are not applicable to FCS.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
(Alloy 600) Steam generator tubes, repair sleeves, and plugs	Crack initiation and growth due to PWSCC, outside diameter stress corrosion cracking (ODSCC), and/or intergranular attack (IGA) or loss of material due to wastage and pitting corrosion, and fretting and wear; or deformation due to corrosion at tube support plate intersections	Steam generator tubing integrity; water chemistry	Yes, effectiveness of a proposed AMP is to be evaluated	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. This aging effect is managed by the Steam Generator Program (B.1.7) and Chemistry Program (B.1.1). These programs are described in Appendix B of this application. The Technical Specifications had already incorporated NRC-approved basis for steam generator degradation management. 3. Consistent with NUREG-1801, this group includes Alloy 600 in borated treated and deoxygenated treated water at FCS. 4. In addition to the components discussed in NUREG-1801, Combustion Engineering mechanical and welded steam generator tube plugs are installed at FCS. 5. NUREG-1801, IV.D 1.2-f is not pertinent to FCS, as phosphate chemistry has never been used. Regarding NUREG-1801, IV.D 1.2-g, FCS did not require analysis in accordance with NRC Bulletin 88-02.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Tube support lattice bars made of carbon steel	Loss of section thickness due to FAC	Plant specific	Yes, plant specific	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The inspection of the tube support lattice bars for loss of thickness is included in the Steam Generator Program (B.1.7) described in Appendix B of this application. 3. Consistent with NUREG-1801, this group includes carbon steel in deoxygenated water at FCS.
Carbon steel tube support plate	Ligament cracking due to corrosion	Plant specific	Yes, effectiveness of a proposed AMP is to be evaluated	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. This aging effect is managed by the Steam Generator Program (B.1.7) and Chemistry Program (B.1.1). These programs are described in Appendix B of this application. The FCS Technical Specifications had already incorporated NRC-approved guidance for steam generator degradation management. 3. Consistent with NUREG-1801, this group includes carbon steel in deoxygenated water at FCS.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Steam generator feedwater inlet ring and supports	Loss of material due to flow-corrosion	Combustion engineering (CE) steam generator feedwater ring inspection	Yes, plant specific	As stated in NUREG-1801, Volume 2, VI.D1.3-a this effect is only applicable to certain CE System 80 steam generators. Because of difference in design between the FCS steam generators and the System 80 steam generators, this effect is not applicable to FCS.
Reactor vessel closure studs and stud assembly	Crack initiation and growth due to SCC and/or IGSCC	Reactor head closure studs	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes low alloy steel in air possibly exposed to borated treated water at FCS. 3. The Reactor Head Closure Studs Program is incorporated into the Bolting Integrity Program at FCS.
CASS pump casing and valve body	Loss of fracture toughness due to thermal aging embrittlement	Inservice inspection	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes CASS in borated treated water at FCS.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
CASS piping	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes CASS in borated treated water at FCS. 3. This group does not include the control element drive mechanism pressure housings and pressurizer surge line since they are stainless steel (not CASS) at FCS. This group also does not include the pressurizer spray head since it in not within the scope of license renewal for FCS.
BWR piping and fittings; steam generator components	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes carbon steel in deoxygenated treated water at FCS. 3. In addition to the materials included in NUREG-1801, this group includes low alloy steel at FCS.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Reactor coolant pressure boundary (RCPB) valve closure bolting, manway and holding bolting, and closure bolting in high pressure and high temperature systems	Loss of material due to wear; loss of preload due to stress relaxation; crack initiation and growth due to cyclic loading and/or SCC	Bolting integrity	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes stainless steel, low alloy steel and carbon steel in air possibly exposed to borated treated water at FCS. 3. In addition to the components in NUREG-1801, this group includes the in-core instrumentation penetration flange bolting at FCS.
CRD nozzle	Crack initiation and growth due to PWSCC	Ni-alloy nozzles and penetrations; water chemistry	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes Alloy 600 in borated treated water at FCS.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Reactor vessel nozzles safe ends and CRD housing; reactor coolant system components (except CASS and bolting) *Note: NUREG-1801, Volume 2, items IV.C2.3-b and IV.C2.4-b that include CASS are included in this group.*	Crack initiation and growth due to cyclic loading, and/or SCC, and PWSCC	Inservice inspection; water chemistry	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes stainless steel, austenitic stainless steel, Alloy 600 and carbon or low alloy steel clad with stainless steel in borated treated water at FCS. 3. In addition to the components in NUREG-1801, this group includes pressure boundary components associated with the in-core instrumentation, core support lugs and keyways, valves and the reactor vessel cladding at FCS.
Reactor vessel internals CASS components	Loss of fracture toughness due to thermal aging, neutron irradiation embrittlement, and void swelling	Thermal aging and neutron irradiation embrittlement	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes CASS in borated treated water at FCS.
External surfaces of carbon steel components in reactor coolant system pressure boundary	Loss of material due to boric acid corrosion	Boric acid corrosion	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes carbon steel in air possibly exposed to borated treated water at FCS.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Steam generator secondary manways and handholds (CS)	Loss of material due to erosion	Inservice inspection	No	This item is not applicable since FCS is a Combustion Engineering designed and manufactured Nuclear Steam Supply System.
Reactor internals, reactor vessel closure studs, and core support pads	Loss of material due to wear	Inservice inspection	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes low alloy steel and stainless steel in borated treated water at FCS. 3. This group does not include control element assembly shroud extension shaft guide since they are not a component at FCS, and fuel alignment pins since they are part of the fuel assembly at FCS.
Pressurizer integral support	Crack initiation and growth due to cyclic loading	Inservice inspection	No	The component identified in NUREG-1801 is not applicable to FCS.
Upper and lower internals assembly (Westinghouse)	Loss of preload due to stress relaxation	Inservice inspection; loose part and/or neutron noise monitoring	No	These items are not applicable since FCS is a Combustion Engineering designed and manufactured reactor.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Reactor vessel internals in fuel zone region (except Westinghouse and Babcock & Wilcox [B&W] baffle bolts)	Loss of fracture toughness due to neutron irradiation embrittlement, and void swelling	PWR vessel internals; water chemistry	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes stainless steel and nickel alloys in borated treated water at FCS. 3. In addition to the components in NUREG-1801, this group includes the thermal shield (its fasteners, positioning pins and shims), and in-core instrument guide tubes and supports which run along the control element assembly shrouds at FCS. 4. This group does not include core shroud tie rods since they are not part of the FCS core shroud and does not include fuel alignment pins since they are part of the fuel assembly at FCS.
Steam generator upper and lower heads; tubesheets; primary nozzles and safe ends	Crack initiation and growth due to SCC, PWSCC. IASCC	Inservice inspection; water chemistry	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes stainless steel in borated treated water at FCS. 3. In addition to the materials in NUREG-1801, this group includes the steam generator lower head manway nickel based alloy cladding at FCS.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Vessel internals (except Westinghouse and B&W baffle former bolts)	Crack initiation and growth due to SCC and IASCC	PWR vessel internals; water chemistry	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes stainless steel, nickel alloys and CASS in borated treated water at FCS. 3. In addition to the components in NUREG-1801, this group includes the thermal shield (its fasteners, positioning pins and shims), and in-core instrument guide tubes and supports at FCS. 4. This group does not include core shroud tie rods since they are not part of the FCS core shroud and does not include fuel alignment pins since they are part of the fuel assembly at FCS.
Reactor internals (B&W screws and bolts)	Loss of preload due to stress relaxation	Inservice inspection; loose part monitoring	No	These items are not applicable since FCS is a Combustion Engineering designed and manufactured reactor.
Reactor vessel closure studs and stud assembly	Loss of material due to wear	Reactor head closure studs	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes high strength steel in air possibly exposed to borated treated water at FCS. 3. The Reactor Head Closure Studs Program is incorporated into the Bolting Integrity Program at FCS.

TABLE 3.1-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Reactor internals (Westinghouse upper and lower internal assemblies; CE bolts and tie rods)	Loss of preload due to stress relaxation	Inservice inspection; loose part monitoring	No	1. The FCS aging management review results are consistent, with deviation, with those reviewed and approved in NUREG-1801. The Loose Part Monitoring Program is not credited at FCS. 2. This aging effect is managed by the Reactor Vessel Internals Inspection Program (B.2.9) as described in Appendix B of this application. 3. Consistent with NUREG-1801, this group includes nickel alloy and stainless steel in borated treated water at FCS. 4. In addition to the components in NUREG-1801, this group includes the socket head cap screws - snubber spacer block – core support barrel snubbers, and thermal shield positioning pins at FCS. 5. This group does not include core shroud tie rods since they are not part of the FCS core shroud.

3.1.2 COMPONENTS OR AGING EFFECTS THAT ARE NOT ADDRESSED IN NUREG-1801

Table 3.1-2 contains the Reactor Coolant Systems aging management review results that are not addressed in NUREG-1801. This table includes the component types, materials, environments, aging effects requiring management and the programs and activities for managing aging.

TABLE 3.1-2
FCS REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Flow element/orifice, pressurizer manway gasket retainer plate, nozzles, nozzle safe ends, pipes & fittings, reactor coolant pump casing, seal covers and reactor coolant pump seal bleed off flanges, pressurizer welds, and valve bodies	Stainless Steel	Ambient Air	None	Not Applicable
Electric heaters, steam generator - tubes, nozzles, nozzle safe ends, CEDM and incore instrument housings, reactor head vent pipe, pressurizer bottom head plate cladding, steam generator primary head cladding and shock suppressors & support	Nickel Based Alloy including Alloy 600	Borated Treated Water	Loss of Material Crevice corrosion in the presence of sufficient levels of oxygen, halogens, sulfates, or copper	Chemistry Program (B.1.1)
Nozzle, nozzle safe end and reactor coolant pipe alloy 182 welds	Nickel Based Alloy including Alloy 600	Ambient Air	None	Not Applicable
Steam generator lower head cladding	Nickel Based Alloy	Borated Treated Water	Cracking	Chemistry Program (B.1.1)

TABLE 3.1-2

FCS REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Reactor coolant pump thermal barrier	Cast Austenitic Stainless Steel (CASS)	Corrosion-Inhibited Treated Water	Cracking Exposure of stainless steel to halogens and sulfates	Chemistry Program (B.1.1) and Inservice Inspection Program (B.1.5)
Secondary side of the tubesheet	Low-Alloy Steel	Deoxygenated Treated Water (>200 deg F)	Loss of Material • General and crevice due to the exposure of low-alloy steel to dissolved oxygen • Pitting corrosion due to the exposure of low-alloy steel to halogens and sulfates	Chemistry Program (B.1.1) and Steam Generator Program (B.1.7)
Steam generator tube plugs	Nickel Based Alloy	Deoxygenated Treated Water (>200 deg F)	Loss of Material Crevice and pitting corrosion due to the exposure of nickel-based alloys to halogens and sulfates	Chemistry Program (B.1.1) and Steam Generator Program (B.1.7)

3.2 AGING MANAGEMENT OF ENGINEERED SAFETY FEATURES SYSTEMS

The Engineered Safety Features Systems are composed of the Safety Injection and Containment Spray System and the Mechanical Containment Penetrations Commodity Group at FCS.

The Safety Injection (SI) System injects borated water into the Reactor Coolant System to provide emergency core cooling. The major components of the SI system are the three high pressure safety injection (HPSI) pumps, two low pressure safety injection (LPSI) pumps, four safety-injection tanks, four safety-injection leakage coolers, eight HPSI control valves, four LPSI control valves and other various valves, instrumentation, and piping.

During normal plant operation the SI system is maintained in a standby mode with all of its components lined up for emergency injection. A safety injection actuation signal (SIAS) automatically starts the HPSI and LPSI pumps and automatically opens the high pressure and low pressure injection valves. During the injection mode of operation, the HPSI and LPSI pumps take suction from the Safety Injection and Refueling Water Tank (SIRWT) and inject borated water into the Reactor Coolant System (RCS) via the safety injection nozzles located on the RCS cold legs. The four safety injection tanks constitute a passive injection system.

The Containment Spray (CS) System consists of three spray pumps, two heat exchangers (shutdown cooling heat exchangers) and all necessary piping, valves, instruments, and accessories. The pumps discharge the borated water through the two heat exchangers, during recirculation, to a dual set of spray headers and spray nozzles in the containment. These spray headers are supported from the containment roof.

The Containment Penetrations and System Interface Components for Non-CQE Systems Commodity Group consists of isolation valves, piping, and mechanical penetrations into containment for the following mechanical systems: Compressed Air (CA-PA), Demineralized Water (DW), Blowpipe and Feedwater Blowdown (FW-BD). The safety related heat exchangers in the Demineralized Water System are included. The mechanical portions of all electrical penetrations (i.e., canister and header plate) are also included.

Operating Experience:

Site: A review of plant specific operating experience was conducted, including the review of Condition Reports and discussions with appropriate site personnel to identify AERM. These reviews concluded that the AERM identified by the FCS specific operating experience were consistent with those identified in NUREG-1801.

Industry: A review of industry-wide operating experience was conducted to

identify aging effects requiring management. This included a review of operating experience issued during 2001. This review concluded that the AERM identified by industry operating experience were consistent with those identified in NUREG-1801.

On-Going: The on-going review of plant specific and industry-wide operating experience is conducted in accordance with the FCS Operating Experience Program.

3.2.1 AGING MANAGEMENT PROGRAMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

Table 3.2-1 shows the aging management programs evaluated in NUREG-1801 that are relied on for license renewal of the Engineered Safeguards Features Systems at FCS. Note that this table only includes those components, materials and environments that are applicable to a PWR. Information on FCS specific components and materials, not listed in NUREG-1801 but included in the component group of this application, is included in the discussion column.

TABLE 3.2-1
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Piping, fittings, and valves in emergency core cooling system	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The metal fatigue time limited aging analyses are discussed in Section 4.3.
Components in containment spray (PWR only), standby gas treatment (BWR only), containment isolation, and emergency core cooling systems	Loss of material due to general corrosion	Plant specific	Yes, plant specific	The combinations of materials and environments identified in NUREG-1801 are not applicable to FCS.
Components in containment spray (PWR only), standby gas treatment (BWR only), containment isolation, and emergency core cooling systems	Loss of material due to pitting and crevice corrosion	Plant specific	Yes, plant specific	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The Chemistry Program (B.1.1) supplemented by the One Time Inspection Program (B.3.5) manages the aging effects of these components. These programs are described in Appendix B of this application. 3. Consistent with NUREG-1801, this group only includes stainless steel in oxygenated treated water for components in containment isolation at FCS.

TABLE 3.2-1
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Containment isolation valves and associated piping	Loss of material due to microbiologically influenced corrosion	Plant specific	Yes, plant specific	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The Chemistry Program (B.1.1) supplemented by the One Time Inspection Program (B.3.5) manages the aging effects of these components. These programs are described in Appendix B of this application. 3. Consistent with NUREG-1801, this group includes stainless steel in treated water at FCS.
High pressure safety injection (charging) pump miniflow orifice	Loss of material due to erosion	Plant specific	Yes, plant specific	The component identified in NUREG-1801 is not applicable to FCS.
External surface of carbon steel components *This row is only found in NUREG-1801, Volume1, Table 2. It is not found in Table 3.2-1 of NUREG-1800.*	Loss of material due to general corrosion	Plant specific	Yes, plant specific	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The General Corrosion of External Surfaces Program (B.3.3) manages the aging effects of these components. This program is described in Appendix B of this application. 3. Consistent with NUREG-1801, this group includes carbon steel components in ambient air at FCS. 4. In addition to materials included in NUREG-1801 this group includes galvanized carbon steel at FCS.

TABLE 3.2-1
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Piping and fittings of CASS in emergency core cooling system	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No	The material identified in NUREG-1801 is not applicable to FCS.
Components serviced by open-cycle cooling system	Local loss of material due to corrosion and/or buildup of deposit due to biofouling	Open-cycle cooling water system	No	The combinations of components, materials and environments identified in NUREG-1801 are not applicable to FCS. The Engineered Safety Features are not serviced by the open-cycle cooling system at FCS.
Components serviced by closed-cycle cooling system	Loss of material due to general, pitting, and crevice corrosion	Closed-cycle cooling water system	No	1. The aging management results are consistent with the results documented in NUREG-1801. 2. In addition to the programs identified in NUREG-1801, the Cooling Water Corrosion Program is supplemented by the Selective Leaching Program to manage the loss of material from cast iron components. 3. Consistent with NUREG-1801, this group includes stainless steel, carbon steel and cast iron in corrosion-inhibited treated water at FCS.

TABLE 3.2-1
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Pumps, valves, piping, and fittings in containment spray and emergency core cooling systems	Crack initiation and growth due to SCC	Water chemistry	No	1. The aging management results are consistent with the results documented in NUREG-1801. 2. Consistent with NUREG-1801, this group includes stainless steel and stainless steel clad carbon steel in chemically treated borated water at FCS. 3. In addition to the components in NUREG-1801, this group includes the safety injection tanks (accumulators), flow element and orifice bodies, orifice plate, and heat exchangers in the Engineered Safety Features System at FCS. 4. In addition to the components in NUREG-1801, this group includes pipes, fittings, valve bodies, filter casings, pump casings, ion exchangers and heat exchangers in the Spent Fuel Cooling System, which is one of the Auxiliary Systems at FCS.
Carbon steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No	1. The aging management results are consistent with the results documented in NUREG-1801. 2. Consistent with NUREG-1801, this group includes carbon and low alloy steel at FCS. 3. In addition to the materials discussed in NUREG-1801, this group includes cast iron and galvanized carbon steel at FCS.

TABLE 3.2-1
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Closure bolting in high pressure or high temperature systems	Loss of material due to general corrosion, loss of preload due to stress relaxation, and crack initiation and growth due to cyclic loading or SCC	Bolting integrity	No	1. The aging management results are consistent with the results documented in NUREG-1801. 2. Consistent with NUREG-1801, this group includes carbon and low alloy steel in ambient air at FCS.

3.2.2 COMPONENTS OR AGING EFFECTS THAT ARE NOT ADDRESSED IN NUREG-1801

Table 3.2-2 contains Engineered Safety Features Systems aging management review results that are not addressed in NUREG-1801. This table includes the component types, materials, environments, and aging effects requiring management, and the programs and activities for managing aging.

TABLE 3.2-2
FCS ENGINEERED SAFETY FEATURES COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Heat Exchanger - Tubes	Alloy 600	Treated Water - Borated	Loss of Material Crevice corrosion in the presence of sufficient levels of oxygen, halogens, sulfates, or copper	Chemistry Program (B.1.1)
Heat Exchanger - Tubes	Alloy 600	Treated Water - Borated	Cracking Stress Corrosion Cracking due to exposure to halogens or sulfates	Chemistry Program (B.1.1)
Heat Exchanger - Tubes	Alloy 600	Nitrite Corrosion-Inhibited Water	Loss of Material • Crevice and pitting corrosion in the presence of sufficient levels of oxygen, halogens, or sulfates • MIC due to exposure to microbiological activity	Chemistry Program (B.1.1) and Cooling Water Corrosion Program (B.2.3)
Pipes & Fittings	Brass	Ambient Air	None	Not Applicable
Pipes & Fittings, Bolting, Flow Element/Orifice Bodies, Tanks, Tubing, Valve Bodies, Canister for Electrical Containment Penetrations, Demineralized Water Heat Exchangers, Orifice Plate and Pump Casing	Stainless Steel	Ambient Air	None	Not Applicable

3.3 AGING MANAGEMENT OF AUXILIARY SYSTEMS

The Auxiliary Systems consist of the following systems and components:

- Spent Fuel Pool Cooling System
- Fuel Handling and Heavy Load Cranes (including New and Spent Fuel Storage Racks)
- Raw Water System (Open Cycle Cooling Water System in NUREG-1801)
- Component Cooling Water System (Closed Cycle Cooling Water System in NUREG-1801)
- Chemical and Volume Control System
- Instrument Air System
- Nitrogen Gas System
- Control Room HVAC and Toxic Gas Monitoring System
- Auxiliary Building HVAC System
- Containment HVAC System
- Ventilating Air System (includes Diesel Generator rooms)
- Fire Protection System including the Fire Protection Fuel Oil System
- Diesel Generator Fuel Oil System and Auxiliary Boiler Fuel Oil System
- Diesel Generator System including the Diesel Jacket Water System, the Diesel Generator Lube Oil System, and the Diesel Generators Starting Air System
- Primary Sampling System
- Liquid Waste Disposal System
- Gaseous Waste Disposal System
- Radiation Monitoring-Mechanical Components

The Spent Fuel Pool Cooling System consists of a stainless steel lined storage pool, two storage pool circulation pumps, a storage pool heat exchanger, a demineralizer and filter, two fuel transfer canal drain pumps, piping, and manual valves. The pool concrete and liner are evaluated with the Auxiliary Building.

The Fuel Handling and Heavy Load Cranes System consists of the refueling machine, tilting machines in the Auxiliary Building and in Containment, fuel transfer conveyor, fuel transfer carrier box, fuel transfer tube, new and spent fuel handling tools, new and spent fuel storage racks, thirty-six (36) cranes of varying types (i.e., polar crane, overhead crane, hoist with monorail, and jib crane) and three (3) elevators.

The Raw Water (RW) system is an open-cycle cooling water system which uses screened water from the Missouri River. The system includes four parallel vertical mixed-flow pumps installed in the Intake Structure pump house. The pumps discharge into an interconnected header which splits into two parallel supply headers. The two supply headers run underground from the Intake Structure to the Auxiliary Building, where they join in an interconnected inlet header to the four Component Cooling Water (CCW) heat exchangers. Downstream of the CCW heat exchangers, the Raw Water discharge header runs through the Turbine Building and discharges to the river.

The Component Cooling Water System is a closed loop system used to transfer heat from various components carrying radioactive or potentially radioactive fluids to the raw water. This system consists of three motor driven circulating pumps, four heat exchangers, a surge tank, valves, and piping. The water in the system is demineralized and deaerated and an inhibitor is added for protection against corrosion.

The Chemical and Volume Control System includes one regenerative heat exchanger, one letdown heat exchanger, five ion exchangers, one volume control tank, three positive-displacement charging pumps, one boric acid batching tank, two boric acid storage tanks, two centrifugal boric acid transfer pumps, and one chemical addition tank.

The Instrument Air System provides oil-free, filtered, and dried air for pneumatic controls, instrumentation, and the actuation of valves, dampers and similar devices. Instrument Air is distributed to the various pneumatic components it serves through a network of supply headers and distribution risers. The Instrument Air system also feeds the suction of the compressors for the Diesel Starting Air system. Backup accumulators containing instrument air or nitrogen are provided on selected pneumatic devices to ensure their operability if instrument air pressure drops.

The Nitrogen Gas System provides compressed nitrogen gas to the Safety Injection Tanks and provides a gas blanket to various vessels and contained areas of the plant.

The Control Room HVAC and Toxic Gas Monitoring System consists of two air conditioning units; two outside air filter units, each with its own supply fan; an outside air intake plenum; and distribution ductwork.

The Auxiliary Building HVAC System is a once-through, non-recirculating type using supply and exhaust fans. Portions of the Auxiliary Building HVAC System may be utilized to purge hydrogen from the containment.

The Containment HVAC System provides ventilation and cooling of the containment. Containment HVAC consists of four separate sub-systems. These sub-systems provide containment air re-circulation, cooling, nuclear detector well cooling, containment purge, and hydrogen purge.

The Ventilating Air System passive equipment is contained within the Emergency Diesel Generator rooms.

The Fire Protection System water supply system has two vertical turbine type fire pumps. One fire pump is driven by an electric motor and the other fire pump is driven by a diesel engine. Both pumps deliver screened and strained Missouri River water to the underground water distribution system, which in turn supply the automatic water fire

suppression systems, interior hose stations and fire hydrants in the yard. An independent underground looped yard main system capable of delivering sprinkler flow plus adequate hose flow to support manual fire fighting for a single fire is provided for the Fire Protection System.

Four safety related plant areas are provided with automatic halon 1301 extinguishing systems. These areas include the Cable Spreading Room, both Switchgear Rooms and the Control Room cabinets. The plant is divided into unique fire areas as required by Appendix A to NRC Branch Technical Position APCSB 9.5-1, and 10CFR 50, Appendix R. Walls enclosing separate fire areas utilize fire resistive construction. Openings in plant fire barriers are protected by rated fire doors, fire dampers, and fire barrier penetration seals. Portable fire extinguishers are identified in the Fire Hazards Analysis as being provided throughout the station, generally in accordance with NFPA 10. Fire extinguishers, fire hoses, and air packs are not subject to an aging management review because they are replaced based on condition in accordance with applicable NFPA standards and plant procedures for fire protection equipment. This position is consistent with the NRC Staff's guidance on consumables, which has been incorporated into NEI 95-10 Revision 2.

RCP lube oil collection neoprene hoses will be replaced on condition in accordance with the Period Surveillance and Preventive Maintenance Program. These hoses provide a gravity drain of RCP lube oil from the collection pans to the lube oil collection tanks. The hoses are not pressurized and do not normally contain fluid.

The Fire Protection Fuel Oil System supplies the sole source of fuel oil to the diesel engine fire pump. The unit is located at the north end of the Intake Structure. A 10-gallon fuel oil day tank for the diesel engine is located adjacent to the engine. Fuel is transferred from the 550-gallon diesel fire pump fuel oil tank to the day tank.

The Diesel Generator Fuel Oil System provides fuel to the emergency diesel generators in the proper amount to maintain engine speed and load. An 18,000 gallon underground storage tank serves both engines. Two transfer pumps for each diesel transfer fuel from the underground storage tank to the wall-mounted auxiliary tank. Fuel gravity drains from the wall mounted tank to the engine base tank. One engine-driven fuel oil pump and one motor driven fuel oil pump delivers fuel to the engine fuel injectors. Warehoused components include a portable hand pump, a rubber hose and hose couplings. These components will be replaced on performance or condition in accordance with the Periodic Surveillance and Preventive Maintenance Program. These components contribute to the first intended function listed above, involving the transfer of diesel fuel from the auxiliary boiler fuel oil storage tank to the diesel engine fuel oil storage tank. The components are normally not pressurized and normally do not contain fluid. The Auxiliary Boiler Fuel Oil System consists of a fuel oil transfer pump, piping, filters, instrumentation and warehoused equipment for delivery of fuel oil from the auxiliary boiler fuel oil storage tank to the diesel engine fuel oil storage tank.

The Diesel Generator System includes emergency diesel generators designed to furnish reliable in-plant ac power when power is not available from the 345 or 161-kV systems. Each emergency diesel generator is provided with an exhaust silencer and auxiliaries. Each emergency diesel generator interfaces with an integral cooling system, two air starting systems, a lubricating system, two fuel systems between the engine mounted fuel oil tanks and the engine fuel lines. Both emergency diesel generators are supplied fuel from a common, underground fuel oil storage tank by redundant transfer pumps. Immersion heaters are provided to maintain engine jacket water and lubricating oil temperatures at desirable temperatures for quick, reliable starting. The emergency diesel generators are located in separate rooms of the Auxiliary Building.

The Diesel Jacket Water System provides cooling to the engine. Each engine has its own self contained radiator type cooling system. Two different coolant mixtures are used in the diesels. For DG-1 a glycol based coolant mixture is used during the winter months with the coolant mixture being changed out to a nitrite based coolant mixture during the summer to ensure the rating of the generator. DG-2 uses a glycol based coolant mixture year round. The Diesel Generator Lube Oil System lubricates the diesel engine components and filters the engine lube oil. The Diesel Generators Starting Air System provides stored pressurized air for starting the emergency diesel generators. Each tank has the capacity for five starts of the diesel (combining for a total of ten emergency starts).

The Primary Sampling system includes the primary sampling panel, the CVCS panel, the steam generator blowdown analyzer rack, the instrument panel, steam generator blowdown sample chiller, and the manual sampling sink and hood.

The Liquid Waste Disposal system is used to collect, store, prepare for disposal, and dispose of liquid radioactive wastes. Radioactive liquid wastes are generated as a result of plant operation, repair, and maintenance activities.

The Gaseous Waste Disposal System includes the containment isolation valves that close on a Containment Isolation Actuation Signal (CIAS) and the piping between the containment penetrations and the containment isolation valves. Also included are the waste gas compressor seal water heat exchangers that receive cooling water from the Component Cooling Water System.

The Radiation Monitoring-Mechanical Components System consists of the mechanical portions of the radiation monitors and their supporting components.

Operating Experience:

Site: A review of plant specific operating experience was conducted, including the review of Condition Reports and discussions with appropriate site personnel to identify AERM. These reviews concluded that the AERM identified by the FCS specific operating experience

were consistent with those identified in NUREG-1801.

Industry: A review of industry-wide operating experience was conducted to identify aging effects requiring management. This included a review of operating experience issued during 2001. This review concluded that the AERM identified by industry operating experience were consistent with those identified in NUREG-1801.

On-Going: The on-going review of plant specific and industry-wide operating experience is conducted in accordance with the FCS Operating Experience Program.

3.3.1 AGING MANAGEMENT PROGRAMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

Table 3.3-1 shows the aging management programs evaluated in NUREG-1801 that are relied on for license renewal of the Auxiliary Systems at FCS. Note that this table only includes those components, materials and environments that are applicable to a PWR. Information on FCS-specific components and materials, not listed in NUREG-1801 but included in the component group of this application, is included in the discussion column.

TABLE 3.3-1
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Components in spent fuel pool cooling and cleanup	Loss of material due to general, pitting, and crevice corrosion	Water chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated	The combinations of components, materials and environments identified in NUREG-1801 are not applicable to FCS. These components are addressed in the tenth line of Table 3.2-1
Linings in spent fuel pool cooling and cleanup system; seals and collars in ventilation systems	Hardening, cracking and loss of strength due to elastomer degradation; loss of material due to wear	Plant specific	Yes, plant specific	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The General Corrosion of External Surfaces Program (B.3.3) manages this aging effect. This program is described in Appendix B of this application. 3. Consistent with NUREG-1801 this group only includes elastomer seals in the ventilation systems exposed to ambient air at FCS.
Components in load handling, chemical and volume control system (PWR), and reactor water cleanup and shutdown cooling systems (older BWR)	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801 for chemical and volume control and primary sampling systems. 2. The metal fatigue time limited aging analyses are discussed in Section 4.3.1 of this application.

TABLE 3.3-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Manage-ment Programs	Further Evalua-tion Recom-mended	Discussion
Heat exchangers in reactor water cleanup system (BWR); high pressure pumps in chemical and volume control system (PWR)	Crack initiation and growth due to SCC or cracking	Plant specific	Yes, plant specific	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The Chemistry (B.1.1) and One-Time Inspection Programs (B.3.5) manage this aging effect. A One Time Inspection will be conducted prior to the period of extended operation to confirm the effectiveness of the Chemistry Program. These programs are described in Appendix B of this application. 3. Consistent with NUREG-1801 this group includes stainless steel in chemically treated borated water at FCS.

TABLE 3.3-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Components in ventilation systems, diesel fuel oil system, and emergency diesel generator systems; external surfaces of carbon steel components	Loss of material due to general, pitting, and crevice corrosion, and MIC	Plant specific	Yes, plant specific	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The Periodic Surveillance and Preventive Maintenance (B.2.8), General Corrosion of External Surfaces (B.3.3) and Fire Protection Programs (B.2.6) manage this aging effect. These programs are described in Appendix B of this application. 3. Consistent with NUREG-1801 this group includes carbon steel, galvanized carbon steel, and copper in air, and carbon steel in diesel engine exhaust gases at FCS. 4. In addition to the materials identified in NUREG-1801, this group includes cast iron, low alloy steel, ductile iron and cadmium plated steel in ambient air; and stainless steel, galvanized carbon steel and coated carbon steel in diesel engine exhaust gases at FCS.

TABLE 3.3-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Components in reactor coolant pump oil collect system of fire protection	Loss of material due to galvanic, general, pitting, and crevice corrosion	One-time inspection	Yes, detection of aging effects is to be further evaluated	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The One-Time Inspection Program (B.3.5) and Periodic Surveillance and Preventive Maintenance (B.2.8) Programs manage this aging effect. These inspections will be conducted prior to the period of extended operation. These programs are described in Appendix B of this application. 3. Consistent with NUREG-1801 this group includes copper in lubricating oil at FCS. 4. In addition to the materials identified in NUREG-1801 this group includes stainless steel at FCS.

TABLE 3.3-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Diesel fuel oil tanks in diesel fuel oil system and emergency diesel generator system	Loss of material due to general, pitting, and crevice corrosion, MIC, and biofouling	Fuel oil chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The Diesel Fuel Monitoring and Storage Program (B.2.4) manages this aging effect. This program is described in Appendix B of this application. The Diesel Fuel Monitoring and Storage Program also includes the fuel oil chemistry program at FCS. The Diesel Fuel Monitoring and Storage Program includes measures to verify the effectiveness of the fuel oil chemistry control. These inspections will be conducted prior to the period of extended operation to confirm the effectiveness of the fuel oil chemistry program. 3, Consistent with NUREG-1801 this group includes carbon steel in fuel oil at FCS. 4. In addition to the materials identified in NUREG-1801 this group includes coated carbon steel, cast iron, stainless steel and bronze at FCS. 5. In addition to the components in NUREG-1801 this group includes filter housings, valve bodies, pump casings, pipes, fittings and tubing at FCS.

TABLE 3.3-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Heat exchangers in chemical and volume control system	Crack initiation and growth due to SCC and cyclic loading	Water chemistry and a plant-specific verification program	Yes, plant specific	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The Chemistry Program (B.1.1) verified by the One Time Inspection Program (B.3.5), Cooling Water Corrosion Program (B.2.3) and Periodic Surveillance and Preventive Maintenance Program (B.2.8) manage this aging effect. These inspections will be conducted prior to the period of extended operation to confirm the effectiveness of the Chemistry Program. These programs are described in Appendix B of this application. 3. Consistent with NUREG-1801 this group includes stainless steel in chemically treated borated water and corrosion inhibited treated water at FCS. 4. In addition to the components included in NUREG-1801 this group includes piping, fittings, valve bodies, and heat exchanger tubes in the Primary Sampling System; piping, fittings, and valves in the Liquid Waste Disposal System; and piping, fittings, valve bodies, transmitter housing, filter/ strainer housing, tanks, ion exchanger housing, and flow element housing in the Chemical and Volume Control System at FCS.

TABLE 3.3-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Neutron absorbing sheets in spent fuel storage racks	Reduction of neutron absorbing capacity and loss of material due to general corrosion (Boral, boron steel)	Plant specific	Yes, plant specific	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The Periodic Surveillance and Preventive Maintenance Program (B.2.8) manages this aging effect. This program is described in Appendix B of this application. The surveillance test evaluates the neutron absorbing samples for dimensional change, weight, neutron attenuation change and specific gravity change. 3. Consistent with NUREG-1801 this group includes Boral encapsulated in stainless steel in chemically treated borated water at FCS.
New fuel rack assembly	Loss of material due to general, pitting, and crevice corrosion	Structures monitoring	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801 this group includes carbon steel in ambient air at FCS.
Spent fuel storage racks and valves in spent fuel pool cooling and cleanup	Crack initiation and growth due to stress corrosion cracking	Water chemistry	No	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes stainless steel in borated treated water at FCS. 3. In addition to the components in NUREG-1801 this group includes fuel tilting machine, fuel transfer tube, fuel transfer conveyor, fuel transfer carrier box and miscellaneous fuel handling equipment at FCS.

TABLE 3.3-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Neutron absorbing sheets in spent fuel storage racks	Reduction of neutron absorbing capacity due to Boraflex degradation	Boraflex monitoring	No	The material identified in NUREG-1801 is not applicable to FCS.
Closure bolting and external surfaces of carbon steel and low-alloy steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes carbon and low alloy steel in air exposed to leaking and dripping borated treated water at FCS. 3. In addition to the materials identified in NUREG-1801 this group includes cast iron, galvanized carbon steel, coated carbon steel, and cadmium plated steel at FCS.
Components in or serviced by closed-cycle cooling water system	Loss of material due to general, pitting, and crevice corrosion, and MIC	Closed-cycle cooling water system	No	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801 this group includes carbon steel in chemically treated corrosion inhibited water at FCS. 3. In addition to the materials identified in NUREG-1801 this group includes stainless steel and copper alloy at FCS. 4. In addition to the components in NUREG-1801 this group includes flow element housing; indicator housing; orifice plate; and the piping, fittings, valves, pumps and heater sleeves of the Diesel Jacket Water closed cycle cooling loop at FCS.

TABLE 3.3-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Manage-ment Programs	Further Evalua-tion Recom-mended	Discussion
Cranes including bridge and trolleys and rail system in load handling system	Loss of material due to general corrosion and wear	Overhead heavy load and light load handling systems	No	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801 this group includes carbon steel in ambient air at FCS. 3. In addition to the components in NUREG-1801 this group includes the reactor vessel head lift rig at FCS.
Components in or serviced by open-cycle cooling water systems	Loss of material due to general, pitting, crevice, and galvanic corrosion, MIC, and biofouling; buildup of deposit due to biofouling	Open-cycle cooling water system	No	1. The aging management results are consistent with the results documented in NUREG-1801. 2. Consistent with NUREG-1801 this group includes carbon steel, bronze, cast iron and stainless steel in raw water at FCS. 3. In addition to the materials in NUREG-1801 this group includes low-alloy steel at FCS. 4. In addition to the components in NUREG-1801 this group includes the intake structure screens and associated components; indicator, flow element and strainer housings at FCS.

TABLE 3.3-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Buried piping and fittings	Loss of material due to general, pitting, and crevice corrosion, and MIC	Buried piping and tanks surveillance or Buried piping and tanks inspection	No Yes, detection of aging effects and operating experience are to be further evaluated	1. The aging management results are consistent with the results documented in NUREG-1801. 2. The aging effects are managed by the Buried Services External Corrosion Program (B.3.2) and the Fire Protection Program (B.2.6) described in Appendix B of this application. The aging management activities of the Fire Protection Program are the same as those of the Buried Services External Corrosion Program. 3. Consistent with NUREG-1801 this group includes carbon steel in soil at FCS. 4. In addition to the materials identified in NUREG-1801 this group includes galvanized carbon steel, cast iron, ductile iron, coated carbon steel, and zinc plated steel at FCS. 5. In addition to the components in NUREG-1801 this group includes valves, tanks and bolting at FCS.
Components in compressed air system	Loss of material due to general and pitting corrosion	Compressed air monitoring	No	The combinations of materials and environments identified in NUREG-1801 are not applicable to FCS.
Components (doors and barrier penetration seals) and concrete structures in fire protection	Loss of material due to wear; hardening and shrinkage due to weathering	Fire protection	No	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801 this group includes carbon steel and sealant in ambient air at FCS.

TABLE 3.3-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Components in water-based fire protection	Loss of material due to general, pitting, crevice, and galvanic corrosion, MIC, and biofouling	Fire water system	No	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801 this group includes carbon steel, cast iron, stainless steel and bronze in raw water at FCS. 3. In addition to the materials identified in NUREG-1801 this group includes galvanized carbon steel, low alloy steel, brass, copper alloy, and ductile iron at FCS. 4. In addition to the components included in NUREG-1801 this group includes bolting and flow element housing at FCS.
Components in diesel fire system	Loss of material due to galvanic, general, pitting, and crevice corrosion	Fire protection and fuel oil chemistry	No	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801 this group includes carbon steel in fuel oil at FCS. 3. In addition to the materials identified in NUREG-1801 this group includes galvanized carbon steel and cast iron at FCS.
Tanks in diesel fuel oil system	Loss of material due to general, pitting, and crevice corrosion	Above ground carbon steel tanks	No	The components identified in NUREG-1801 are not applicable to FCS.

TABLE 3.3-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Closure bolting	Loss of material due to general corrosion; crack initiation and growth due to cyclic loading and SCC	Bolting integrity	No	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801 this group includes carbon steel and low alloy steel in ambient air at FCS.
Components (aluminum bronze, brass, cast iron, cast steel) in open-cycle and closed-cycle cooling water systems, and ultimate heat sink	Loss of material due to selective leaching	Selective leaching of materials	No	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes cast iron and bronze in raw water and soil at FCS.
Fire barriers, walls, ceilings and floors in fire protection	Concrete cracking and spelling due to freeze-thaw, aggressive chemical attack, and reaction with aggregates; loss of material due to corrosion of embedded steel	Fire protection and structures monitoring	No	1. The aging management results are consistent with the results documented in NUREG-1801. 2. Consistent with NUREG-1801 this group includes concrete in ambient air at FCS.

3.3.2 COMPONENTS OR AGING EFFECTS THAT ARE NOT ADDRESSED IN NUREG-1801

Table 3.3-2 contains Auxiliary Systems aging management review results that are not addressed in NUREG-1801. This table includes the component types, materials, environments, and aging effects requiring management, and the programs and activities for managing aging.

TABLE 3.3-2
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Blowers & fans, ventilation damper, filter/strainer housing, valve bodies, valve operators, new fuel storage rack -aluminum, halon system nozzle, switch/bistable housing, transmitter element housing	Aluminum	Ambient Air	None	Not Applicable
Filter/strainer housing, valve bodies	Aluminum	Fuel Oil	Loss of Material • MIC due to the potential for microorganism introduction and moisture contamination during bulk fuel oil supply and delivery • Pitting/Crevice/General Corrosion due to potential for water contamination and water pooling in a fuel oil system	Diesel Fuel Monitoring and Storage Program (B.2.4)
Filter/strainer housing	Aluminum	Fuel Oil	Loss of Material • MIC due to the potential for microorganism introduction and moisture contamination during bulk fuel oil supply and delivery • Pitting/Crevice/General Corrosion due to potential for water contamination and water pooling in a fuel oil system	Fire Protection Program (B.2.6)

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Filter/strainer housing, Valve Operators, Valve bodies	Aluminum	Instrument Air	None	Not Applicable
Valve bodies	Aluminum	Gas - Nitrogen	None	Not Applicable
Switch/bistable housing	Aluminum	Raw Water	Loss of Material Crevice and pitting corrosion and MIC due to stagnant conditions	Fire Protection Program (B.2.6)
New and spent fuel handling tools	Aluminum	Occasionally exposed to Treated Water - Borated	Cracking Due to stress corrosion cracking (SCC) due to the exposure of aluminum to halogens and stress	Chemistry Program (B.1.1)
New and spent fuel handling tools	Aluminum	Occasionally exposed to Treated Water - Borated	Loss of Material • Pitting corrosion due to the exposure of aluminum to halogens and sulfates • Galvanic corrosion due to aluminum in contact with stainless steel and exposed to halogens • Exfoliation due to the exposure of aluminum to halogens	Chemistry Program (B.1.1)
Subcomponent - new fuel storage rack -boral sheets	Boral	Ambient Air	None	Not Applicable

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Valve bodies, bolting, filters/strainer housing, flow element/orifice housing, fire protection sprinkler/spray nozzle, switch/bistable housing, heat exchangers, pump casings	Brass or Bronze	Ambient Air	None	Not Applicable
Valve bodies, filters/strainer housing, pump casings	Brass or Bronze	Fuel Oil	Loss of Material • MIC due to the potential for microorganism introduction and moisture contamination during bulk fuel oil supply and delivery • Pitting/Crevice/General Corrosion due to potential for water contamination and water pooling in a fuel oil system	Diesel Fuel Monitoring and Storage Program (B.2.4)
Valve bodies	Brass	Gas - Halon	None	Not Applicable
Valve bodies	Brass or Bronze	Gas - Instrument Air	None	Not Applicable
Valve bodies	Brass	Gas - Nitrogen	None	Not Applicable
Valve bodies	Brass	Gas - Refrigerant (Liquid)	None	Not Applicable

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Heat exchanger	Brass or Bronze	Lubricating Oil	Loss of Material General corrosion due to the possibility for water contamination and water pooling	Cooling Water Corrosion Program (B.2.3)
Heat exchanger	Brass	Nitrite Corrosion-Inhibited Treated Water	Cracking Due to SCC because of the ammonium compounds present in the water due to the nitrite corrosion inhibitor	Cooling Water Corrosion Program (B.2.3) and Chemistry Program (B.1.1)
Heat exchanger	Brass	Nitrite Corrosion-Inhibited Treated Water	Loss of Material • Crevice and pitting corrosion due to potential stagnant or low flow conditions • Galvanic corrosion due to the high conductivity of the process fluid and the presence of dissimilar metals in contact • MIC due to the exposure of copper alloy to microbiological activity	Cooling Water Corrosion Program (B.2.3) and Chemistry Program (B.1.1)

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Switch/bistable housing	Brass	Raw Water	Loss of Material • Crevice and pitting corrosion and MIC due to stagnant conditions • Galvanic corrosion due to the conductivity of the process fluid and the presence of dissimilar metals in contact	Fire Protection Program (B.2.6)
Valve bodies	Cadmium Plated Steel	Gas - Instrument Air	None	Not Applicable
Pipes & fittings	Carbon Steel	Above ground, buried in gravel and protected from the elements	Loss of Material Due to external surface corrosion due to the potential for the existence of sufficient oxygen, moisture levels, and/or soil contaminants	Diesel Fuel Monitoring and Storage Program (B.2.4)
Pipes & fittings	Carbon Steel	Concrete	None	Not Applicable
Filter strainer housing, heat exchangers, lubricator motors, pipes & fittings, tanks, valve bodies, accumulators, valve operators	Carbon Steel	Gas - Instrument Air	None	Not Applicable
Pipes and fittings	Carbon Steel	Gas - Hydrogen	None	Not Applicable

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Valve bodies, accumulators, pipes & fittings	Carbon Steel	Gas - Nitrogen	None	Not Applicable
Heat exchangers - shell and tube sheet	Carbon Steel	Lubricating Oil	Loss of Material General corrosion due to the possibility for water contamination and water pooling	Periodic Surveillance and Preventive Maintenance Program (B.2.8)
Pipes & fittings, valves	Carbon Steel or Cast Iron	Concrete	None	Not Applicable
Heat exchanger - shell	Carbon Steel	Oxygenated Treated Water <200 deg F	Loss of Material • General and crevice corrosion due to dissolved oxygen • Pitting corrosion due to halogens • Galvanic corrosion due to the conductivity of the process fluid and the presence of dissimilar metals in contact	Chemistry Program (B.1.1) and Cooling Water Corrosion Program (B.2.3)
Valve bodies	Cast Iron	Gas - Refrigerant (Liquid)	None	Not Applicable

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Pump casings, valve bodies, pipes & fittings, heat exchanger - channel/channel head	Cast Iron	Glycol and Nitrite Corrosion-Inhibited Treated Water	Loss of Material • General and crevice corrosion due to the exposure of cast iron to dissolved oxygen • Pitting corrosion due to exposure to halogens	Chemistry Program (B.1.1) and Cooling Water Corrosion Program (B.2.3)
Pump casings, valve bodies, pipes & fittings	Cast Iron	Glycol and Nitrite Corrosion-Inhibited Treated Water	Loss of Material Selective leaching due to the exposure of cast iron to dissolved oxygen	Selective Leaching Program (B.3.6)
Valve bodies, pipes & fittings	Cast Iron	Buried in Ground	Loss of Material • General corrosion due to exposure to dissolved oxygen • Selective leaching due to the exposure of cast iron to dissolved oxygen	Fire Protection Program (B.2.6)

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Tanks	Coated Carbon Steel	Fuel Oil	Loss of Material • MIC due to the potential for microorganism introduction and moisture contamination during bulk fuel oil supply and delivery • Pitting/Crevice/General Corrosion due to potential for water contamination and water pooling in a fuel oil system	Diesel Fuel Monitoring and Storage Program (B.2.4)
Tank	Coated Carbon Steel	Above ground, buried in gravel and protected from weather	Loss of Material General corrosion and pitting due to exposure to the potential for the existence of sufficient oxygen, moisture levels and/or soil contaminants	Diesel Fuel Monitoring and Storage Program (B.2.4)
Pressure vessels	Coated Carbon Steel	Gas - Halon (Liquid)	None	Not Applicable
Pipes & fittings	Concrete	Buried in Ground	None	Not Applicable
Pipes & fittings	Concrete	Raw Water	None	Not Applicable

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Safety Injection Refueling Water Tank	Concrete with coated carbon steel liner	Treated Water - Borated	Loss of Material Due to exposure of the material to moisture, contaminants, dissolved oxygen, and boric acid (i.e., general corrosion, crevice corrosion, pitting corrosion, boric acid corrosion and galvanic corrosion)	Structures Monitoring Program (B.2.10)
Pipes & fittings, tubing	Copper, Copper Alloy, Copper-Zinc Alloy	Gas - Instrument Air	None	Not Applicable
Valve bodies, pipes & fittings, heat exchanger tubes	Copper, Copper Alloy	Gas - Refrigerant	None	Not Applicable
Heat exchangers, valves	Copper, Copper alloy	Nitrite Corrosion-Inhibited Treated Water	Loss of Material • Crevice and pitting corrosion due to potential stagnant or low flow conditions • Galvanic corrosion due to the high conductivity of the process fluid and the presence of dissimilar metals in contact • MIC due to the exposure of copper alloy to microbiological activity	Cooling Water Corrosion Program (B.2.3) and Chemistry Program (B.1.1)

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Tubing, pipes & fittings, heat exchanger - tubes	Brass, Bronze, Copper, Copper Alloy, Copper-Zinc Alloy	Ambient Air	None	Not Applicable
Heat exchangers	Copper Alloy	Nitrite Corrosion-Inhibited Treated Water	Cracking Due to SCC because of the ammonium compounds present in the water due to the nitrite corrosion inhibitor	Cooling Water Corrosion Program (B.2.3) and Chemistry Program (B.1.1)
Tubing	Copper-Zinc Alloy	Buried in Ground	Loss of Material General and pitting corrosion due to the potential for the existence of sufficient oxygen, moisture levels, and/or soil contaminants	Buried Services External Corrosion Program (B.3.2)
Tubing	Copper-Zinc Alloy	Buried in Ground	Loss of Material Due to dezincification	Selective Leaching Program (B.3.6)
Tubing	Copper-Zinc Alloy	Above ground, buried in gravel and protected from weather	Loss of Material General and pitting corrosion due to the potential for the existence of sufficient oxygen, moisture levels, and/or soil contaminants	Diesel Fuel Monitoring and Storage Program (B.2.4)

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Tubing	Copper-Zinc Alloy	Fuel Oil	Loss of Material • MIC due to the potential for microorganism introduction and moisture contamination during bulk fuel oil supply and delivery • Pitting/Crevice/General Corrosion due to potential for water contamination and water pooling in a fuel oil system	Diesel Fuel Monitoring and Storage Program (B.2.4)
Pipes & fittings	Galvanized and Carbon Steel	Gas - Diesel Exhaust	Cracking Due to embrittlement at elevated temperatures	Periodic Surveillance and Preventive Maintenance Program (B.2.8)
Pipes & fittings	Galvanized Steel	Above ground, buried in gravel and protected from the elements	Loss of Material Due to general and pitting corrosion due to the potential for the existence of sufficient oxygen, moisture levels, and/or soil contaminants	Diesel Fuel Monitoring and Storage Program (B.2.4)
Galvanized duct-work for ventilating systems	Galvanized Steel	Exposed to Weather	Loss of Material • Crevice corrosion due to crevices existing that allow a corrosive environment to develop • General corrosion due to presence of both oxygen and moisture	General Corrosion of External Surfaces Program (B.3.3)

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Sight glass	Glass	Fuel Oil, Lubricating Oil, Nitrite Corrosion-Inhibited Treated Water, Air	None	Not Applicable
Flow element/orifice body, pipes & fittings, pump casings, valve bodies	Heat-Traced Stainless Steel	Plant Indoor Air	Cracking Due to possible leachables in heat-tracing adhesive (cement) combined with component temperatures exceeding 160 deg F due to the heat tracing	One Time Inspection Program (B.3.5)
Fire barriers	Mineral Fiber	Ambient Air	Separation Due to vibration, movement, and shrinkage	Fire Protection Program (B.2.6)
Fire barriers	Mineral Fiber Board	Ambient Air	Cracking Due to vibration and movement	Fire Protection Program (B.2.6)
Fire barriers	Mineral Fiber Board	Ambient Air	Loss of Material Due to abrasion	Fire Protection Program (B.2.6)
Fire barriers	Mineral Fiber Board	Ambient Air	Separation Due to vibration, movement, and shrinkage	Fire Protection Program (B.2.6)

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Heat exchanger - tubes, heat exchanger - shell	Nickel-Base Alloy	Deoxygenated Treated Water (>200 deg F)	Cracking Stress Corrosion Cracking due to potential exposure to halogens or sulfates	Chemistry Program (B.1.1) and Cooling Water Corrosion Program (B.2.3)
Heat exchanger - tubes, heat exchanger - shell	Nickel-Base Alloy	Deoxygenated Treated Water (>200 deg F)	Loss of Material • Crevice corrosion due to potential exposure to dissolved oxygen • MIC due to the potential for microbiological activity • Pitting corrosion due to potential exposure to halogens and sulfates	Chemistry Program (B.1.1), Cooling Water Corrosion Program (B.2.3) and One Time Inspection Program (B.3.5)
Heat exchanger - tubes, heat exchanger - shell	Nickel-Base Alloy	Nitrite Corrosion-Inhibited Treated Water	Loss of Material Crevice and pitting corrosion due to the exposure of nickel-based alloys to halogens and sulfates	Chemistry Program (B.1.1) and Cooling Water Corrosion Program (B.2.3)
Heat exchanger - shell	Nickel-Base Alloy	Plant Indoor Air	None	Not Applicable
Fire barriers	Pyrocrete	Ambient Air	Cracking Due to vibration, movement, and shrinkage	Fire Protection Program (B.2.6)

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Fire barriers	Pyrocrete	Ambient Air	Loss of Material • Due to vibration that may cause delamination • Due to movement that may cause separation	Fire Protection Program (B.2.6)
Fire barriers	Pyrocrete	Ambient Air	Separation • Due to contact with pipe surfaces • Due to hydration	Fire Protection Program (B.2.6)
Indicator/recorder body	Polysulfone	Plant Indoor Air	None	Not Applicable
Indicator/recorder body	Polysulfone	Raw Water	None	Not Applicable
Flow element/orifice body, pipes & fittings, valve bodies, heat exchanger-channel/ channel head, heat exchanger - shell, piping spray shield, pressure vessels, filter strainer housing, pump casings, bolting, new Fuel storage racks	Stainless Steel	Ambient Air	None	Not Applicable
Pipes & fittings	Stainless Steel	Concrete	None	Not Applicable
Pipes & fittings, valve bodies	Stainless Steel	Deoxygenated Treated Water (>200 deg F)	Cracking Due to exposure of stainless steel to halogens and sulfates	Chemistry Program (B.1.1) and One Time Inspection Program (B.3.5)

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Pipes & fittings, valve bodies	Stainless Steel	Deoxygenated Treated Water (>200 deg F)	Loss of Material • Crevice corrosion due exposure of stainless steel to dissolved oxygen • Pitting corrosion due to the exposure of stainless steel to halogens and sulfates	Chemistry Program (B.1.1) and One Time Inspection Program (B.3.5)
Filter strainer housing, valve bodies	Stainless Steel	Fuel Oil	Loss of Material • MIC due to the potential for microorganism introduction and moisture contamination during bulk fuel oil supply and delivery • Pitting/Crevice/General Corrosion due to potential for water contamination and water pooling in a fuel oil system	Diesel Fuel Monitoring and Storage Program (B.2.4)
Pipes & fittings	Stainless Steel	Gas - Diesel Exhaust	Cracking Due to moisture-containing contaminants concentrate, resulting in an environment conducive to SCC/IGA	Periodic Surveillance and Preventive Maintenance Program (B.2.8)
Pipes & fittings, valve bodies	Stainless Steel	Gas - Hydrogen	None	Not Applicable
Pipes & fittings, valve bodies, tubing	Stainless Steel	Gas - Instrument Air	None	Not Applicable

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Pipes & fittings, valve bodies, tubing	Stainless Steel	Gas - Nitrogen	None	Not Applicable
Valve bodies, piping spray shield	Stainless Steel	Lubricating Oil	Loss of Material General corrosion due to the possibility for water contamination and water pooling	Fire Protection Program (B.2.6)
Flow element/orifice body, Indicator/recorder housing, orifice plate, pipes & fittings, valve bodies, heat exchanger - tubes	Stainless Steel	Nitrite Corrosion-Inhibited Treated Water	Cracking Due to exposure to halogens and sulfates	Chemistry Program (B.1.1) and Cooling Water Corrosion Program (B.2.3)
Valve bodies, indicator/recorder body, orifice plate, bolting, pipes & fittings, filter strainer housing, tubing, heat exchanger - channel/channel head, ION exchangers, pump casings, tanks, transmitter/element housing, new fuel storage rack, fire blocking damper	Stainless Steel	Ambient Air	None	Not Applicable
Valve bodies, heat exchanger - tubes	Stainless Steel	Oxygenated Treated Water <200 deg F	Loss of Material • Crevice corrosion due to an oxygenated treated water environment • Pitting corrosion due to exposure to halogens and sulfates	Chemistry Program (B.1.1) and Cooling Water Corrosion Program (B.2.3)

TABLE 3.3-2 (CONTINUED)
FCS AUXILIARY SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Filter strainer housing	Stainless Steel	Raw Water	Loss of Material • Crevice corrosion due to the presence of dissolved oxygen and impurities • MIC due to exposure to microbiological activity • Pitting corrosion due to exposure to halide ions	Periodic Surveillance and Preventive Maintenance Program (B.2.8)
Bolting	Zinc plated steel	Buried in ground	Loss of Material General corrosion due to exposure to dissolved oxygen	Fire Protection Program (B.2.6)
Glass in metal fire penetration barriers	Glass	Plant Indoor Air	None	Not Applicable

3.4 AGING MANAGEMENT OF STEAM AND POWER CONVERSION SYSTEMS

The Steam and Power Conversion Systems consist of the Main Steam System, the Main and Auxiliary Feedwater Systems, Steam Generator Blowdown System and associated components at FCS.

The Main Steam System consists of piping from each steam generator that penetrates the containment wall to the main steam isolation valves that are located in each pipe just outside containment. Also included in the Main Steam System boundary is the piping to the turbine-driven auxiliary feedwater pump and the associated drains and vents.

The Feedwater System consists of a supply line to each steam generator. A feedwater isolation valve in each steam generator supply line is located just outside the containment penetration.

The Auxiliary Feedwater (AFW) System supplies feedwater to the steam generators whenever the reactor coolant system temperature is above 300 deg F and the main feedwater system is not in operation. The AFW System contains the emergency feedwater storage tank (EFWST), two pumps, plus related piping, valves, and instrumentation. One pump is electric motor driven, and the other is steam turbine driven. The AFW System can supply the steam generators through two different flow paths. One flow path is through an interconnection with the main feedwater piping upstream of the feedwater regulating valves, after which the water enters the each steam generator through the normal feed ring. This flow path is typically used during normal plant heatup and cooldown evolutions. The other flow path connects to the AFW nozzles on the steam generators. Either AFW pump can pump water from the EFWST to the steam generators.

Operating Experience:

Site: A review of plant specific operating experience was conducted, including the review of Condition Reports and discussions with appropriate site personnel to identify AERM. These reviews concluded that the AERM identified by the FCS specific operating experience were consistent with those identified in NUREG-1801.

Industry: A review of industry-wide operating experience was conducted to identify aging effects requiring management. This included a review of operating experience issued during 2001. This review concluded that the AERM identified by industry operating experience were consistent with those identified in NUREG-1801.

On-Going: The on-going review of plant specific and industry-wide operating experience is conducted in accordance with the FCS Operating Experience Program.

3.4.1 AGING MANAGEMENT PROGRAMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

Table 3.4-1 shows the aging management programs evaluated in NUREG-1801 that are relied on for license renewal of the Steam and Power Conversion Systems at FCS. Note that this table only includes those components, materials and environments that are applicable to a PWR. Information on FCS specific components and materials, not listed in NUREG-1801 but included in the group described in a particular line of the table, is included in the discussion column.

TABLE 3.4-1
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STEAM AND POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
Piping and fittings in main feedwater line, steam line and auxiliary feedwater (AFW) piping	Cumulative fatigue damage	Fatigue Monitoring	Yes, TLAA	1. The TLAA is applicable to Class II and III piping at FCS. See Section 4.3.4 for the TLAA discussion of Class II and III Piping. 2. Consistent with NUREG-1801, this group includes piping, fittings, and valve bodies at FCS.
Piping and fittings, valve bodies and bonnets, pump casings, tanks, tubes, tubesheets, channel head, and shell (except main steam system)	Loss of material	Water chemistry and One-Time Inspection	Yes, detection of aging effects should be further evaluated	1. The Chemistry Program (B.1.1), supplemented by the One-Time Inspection Program (B.3.5) and Preventive Maintenance and Surveillance Program (B.2.8), manages the aging effects of these components. The programs are described in Appendix B of this application. NUREG-1801 indicates that the verification of the effectiveness of the water chemistry program should be conducted with an inspection of stagnant flow locations within the systems. These inspections will be conducted in accordance with either the One-Time Inspection Program or Preventive Maintenance and Surveillance Program. 2. Consistent with NUREG-1801, this group includes carbon steel and stainless steel in treated water at FCS. 3. In addition to the materials identified in NUREG-1801, this group includes low alloy steel components at FCS.

TABLE 3.4-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STEAM AND POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
				4. In addition to the components included in the Steam and Power Conversion Systems in NUREG-1801, this group includes pump casings, valve bodies, piping and fittings, tubes, tubesheets, channel head, flow element bodies, filters/strainer and transmitter bodies at FCS. 5. In addition to the components identified in NUREG-1801, this group includes the stainless steel pipes, fittings, and valve bodies in the Chemical and Volume Control System, which is one of the Auxiliary Systems at FCS.
AFW piping	Loss of material due to general, pitting, and crevice corrosion, MIC, and biofouling	Plant specific	Yes, plant specific	The combinations of components, materials and environments identified in NUREG-1801 are not applicable to FCS. The AFW piping at FCS is not exposed to untreated water from backup water supply

TABLE 3.4-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STEAM AND POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
Oil coolers in AFW system (lubricating oil side possibly contaminated with water)	Loss of material due to general (carbon steel only), pitting, and crevice corrosion, and MIC	Plant Specific Periodic Surveillance and Preventive Maintenance	Yes, plant specific	1. The Periodic Surveillance and Preventive Maintenance Program (B.2.8) manages this aging effect by ensuring water is not present in lubricating oil and that the oil is changed on a refueling frequency. This program is described in Appendix B of this application. 2. Consistent with NUREG-1801, this group includes carbon steel and stainless steel in lubricating oil possibly contaminated with water at FCS. 3. In addition to the materials identified in NUREG-1801, this group includes cadmium plated steel and cast iron at FCS. 4. In addition to the components included in the Steam and Power Conversion Systems in NUREG-1801, this group includes carbon and stainless steel filter/strainer bodies at FCS. 5. In addition to the components identified in NUREG-1801, this group includes piping and fittings, valve bodies, flow element bodies, filter/strainer housings, and tanks in the Diesel Generator Lubricating Oil System, which is one of the Auxiliary Systems at FCS.

TABLE 3.4-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STEAM AND POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
External surface of carbon steel components	Loss of material due to general corrosion	Plant Specific - General Corrosion of External Surfaces	Yes, plant specific	1. The General Corrosion of External Surfaces Program (B.3.3) manages this aging effect. This program is described in Appendix B of this application. 2. Consistent with NUREG-1801, this group includes carbon and low alloy steel in ambient air at FCS.
Carbon steel piping, valve bodies, and pump casings	Wall thinning from flow-accelerated corrosion	Flow Accelerated Corrosion	No	1. The aging management results are consistent with the results reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes carbon steel in treated water and saturated steam at FCS. 3. In addition to the materials identified in NUREG-1801, this group includes low alloy steel components at FCS. 4. In addition to the components included in NUREG-1801, this group includes filter/ strainer bodies at FCS.

TABLE 3.4-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STEAM AND POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
Carbon steel piping and valve bodies in main steam system	Loss of material from crevice and pitting corrosion	Water Chemistry	No	1. The aging management results are consistent with the results documented in NUREG-1801. 2. Consistent with NUREG-1801, this group includes carbon steel in saturated steam at FCS. 3. In addition to the materials identified in NUREG-1801, this group includes low alloy steel at FCS. 4. In addition to the components included in NUREG-1801, this group includes filter/ strainer bodies at FCS.
Closure bolting in high-pressure or high-temperature systems	Loss of material from atmospheric corrosion and crack initiation and growth from cyclic loading, stress corrosion cracking.	Bolting Integrity	No	1. The aging management results are consistent with the results documented in NUREG-1801. 2. Consistent with NUREG-1801, this group includes carbon and low alloy steel bolting in ambient air in high pressure or high temperature systems at FCS.

TABLE 3.4-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STEAM AND POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
Heat exchangers and coolers/ condensers serviced by open-cycle cooling water	Loss of material due to general (carbon steel only), pitting, and crevice corrosion, MIC, and biofouling; buildup of deposit due to biofouling	Open-cycle Cooling Water System	No	The combinations of materials and environment identified in NUREG-1801 are not applicable to FCS.
Heat exchangers and coolers/ condensers serviced by closed-cycle cooling water	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Closed-cycle Cooling Water System	No	1. The aging management results are consistent with the results documented in NUREG-1801. 2. Consistent with NUREG-1801, this group includes stainless steel components in corrosion inhibited treated water at FCS. 3. In addition to the components identified in NUREG-1801, this group includes heat exchanger tubes in the Primary Sampling System and the Spent Fuel Pool Cooling System. It also includes heat exchanger tubes and tubesheets in the Chemical and Volume Control System associated with the vacuum deaerator pumps.
External surface of above ground condensate storage tank	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Above Ground Carbon Steel Tanks	No	The component identified in NUREG-1801 is not applicable to FCS.

TABLE 3.4-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STEAM AND POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
External surface of buried condensate storage tank and AFW piping	Loss of material due to general, pitting, and crevice corrosion, and MIC	Buried piping and tanks surveillance or Buried piping and tanks inspection	No Yes, detection of aging effects and operating experience are to be further evaluated	The component identified in NUREG-1801 is not applicable to FCS.
External surface of carbon steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No	1. The aging management results are consistent with the results documented in NUREG-1801. 2. Consistent with NUREG-1801, this group includes carbon and low alloy steel components in ambient air and leaking and dripping chemically treated borated water at FCS.

3.4.2 COMPONENTS OR AGING EFFECTS THAT ARE NOT ADDRESSED IN NUREG-1801

Table 3.4-2 contains Steam and Power Conversion Systems aging management review results that are not addressed in NUREG-1801. This table includes the component types, materials, environments, aging effects requiring management, and the programs and activities for managing aging.

TABLE 3.4-2
FCS STEAM AND POWER CONVERSION SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Pumps	Aluminum	Lubricating Oil	Loss of Material General corrosion due to the possibility for water contamination and water pooling	Periodic Surveillance and Preventive Maintenance Program (B.2.8)
Pumps	Aluminum	Plant Indoor Air	None	Not Applicable
Heat exchanger (channel, channel head, tubes) and valves	Copper Alloy	Deoxygenated Treated Water (<200 deg F)	Loss of Material • Crevice and pitting corrosion due to potential stagnant or low flow conditions • Wear due to flow induced vibration	One Time Inspection Program (B.3.5)
Heat exchanger (channel, channel head, tubes) and valves	Copper Alloy	Deoxygenated Treated Water (<200 deg F)	Loss of Material Selective leaching	Selective Leaching Program (B.3.6)
Filters/Strainers, heat exchanger (shell and tubes), indicator/ recorder body, pipes, fittings and valves	Copper Alloy	Lubricating Oil	Loss of Material General corrosion due to the possibility for water contamination and water pooling	Periodic Surveillance and Preventive Maintenance Program (B.2.8)
Pipes, fittings, valves filter/strainer, heat exchanger shell	Copper Alloy	Ambient Air	None	Not Applicable

TABLE 3.4-2 (CONTINUED)
FCS STEAM AND POWER CONVERSION SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW NOT EVALUATED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Indicator/Recorder (sightglass)	Glass	Deoxygenated Treated Water <200 deg F Lubricating Oil Ambient Air	None	Not Applicable
Pipes, fittings, valves, filter/strainer, heat exchanger, flow element/orifice, transmitter element, bolting, pump casing	Stainless Steel	Ambient Air	None	Not Applicable
Pipes, fittings, valves, filter/strainer, flow element/orifice, transmitter element, pump casing	Stainless Steel	Oxygenated or Deoxygenated Treated Water (<200 deg F)	Cracking Due to exposure of stainless steel to halogens and sulfates	Water Chemistry (B.1.1) and One-Time Inspection (B.3.5) Programs
Pipes, fittings, and valves	Stainless Steel	Deoxygenated Treated Water (>200 deg F) or Saturated Steam	Cracking Due to exposure of stainless steel to halogens and sulfates	Water Chemistry (B.1.1) and One-Time Inspection (B.3.5) Programs
Pipes, fittings, and valves	Stainless Steel	Deoxygenated Treated Water (>200 deg F) or Saturated Steam	Loss of Material • Crevice corrosion due exposure of stainless steel to dissolved oxygen • Pitting corrosion due to the exposure of stainless steel to halogens and sulfates	Water Chemistry (B.1.1) and One-Time Inspection (B.3.5) Programs

3.5 AGING MANAGEMENT OF CONTAINMENT, STRUCTURES AND COMPONENT SUPPORTS

The Containment, Structures and Component Supports are comprised of the Containment, Auxiliary Building, Turbine and Service Building, Intake Structure, Building Piles and associated component supports at FCS.

The Containment structure is a partially prestressed, reinforced concrete Class I structure composed of cylindrical walls, domed roof and a bottom mat. The mat is common to both the Containment structure and the Auxiliary Building and is supported on steel piles driven to bedrock. The mat incorporates a depressed center portion for the reactor vessel. The Containment has a 1/4-inch internal carbon steel liner. The unbonded tendons are in conduits filled with waterproof grease. The tendon anchors are accessible for inspection, testing, and re-tensioning via the tendon access gallery located directly beneath the cylinder walls and at the dome roof.

The Auxiliary Building is a multi-floored, reinforced concrete, Class I structure. From the bottom of the foundation mat to the roof, the structure is of box-type construction with internal bracing provided by vertical concrete walls and horizontal floor slabs. The spent fuel pool is contained within the Auxiliary Building and consists of a stainless steel lined concrete structure. The control room is located within the Auxiliary Building. The Auxiliary Building masonry walls in the area of safety-related equipment have been reinforced to provide protection for Class I equipment and components located nearby.

The Turbine and Service Building is a multi-floored Class II structure. From the basement floor to the operating floor, the structure is a box-type, reinforced concrete structure with internal bracing provided by concrete walls, floor slabs and structural steel. The mat foundation is supported on steel piles driven to bedrock. From the operating floor to the roof, the structure is braced steel frame clad with aggregate resin panels. The multi-layered built-up roof is supported by metal decking spanning between open web steel joists. The turbine generator is located on the operating floor. It is supported by a mass concrete structure referred to as the turbine pedestal.

The Intake Structure is a multi-floored Class I structure. From the bottom of the foundation mat to seven feet above the operating floor, the structure is a box-type reinforced concrete structure with internal bracing provided by concrete walls and floor slabs. The mat foundation is supported on steel pipe piles driven to bedrock. Above the reinforced concrete structure to the roof the structure is a braced steel frame clad with aggregate resin panels. The multi-layered built-up roof is supported by metal decking spanning between open web steel joists. The diesel-driven fire pump fuel tank enclosure is included in the Intake Structure.

The Building Piles commodity group consists of four types of piles: Class A steel pipe piles, Class B steel pipe piles, concrete caissons, and steel H-piles. Class A piles are 20-inch OD open-end pipe piles with 1.031-inch thick walls driven to bedrock. The piles are filled with sand to the point four feet below the top of the pile. The remaining top four feet is filled with concrete. Class A piles are capped with a 2-inch thick steel plate end closure. Class B piles are 12.75-inch OD closed-end pipe piles with 0.25-inch thick walls and filled with concrete. Class B piles are capped with a 1.25-inch steel plate end closure. Concrete caissons are 3-foot diameter reinforced concrete cylinders that extend 10 feet into bedrock. Steel H-piles are used in the foundations of the diesel engine fuel oil storage tank.

Duct banks are comprised of conduits encased in concrete and are located below grade. Duct banks are used to route electrical power cables between buildings. Electrical manholes are reinforced concrete box-type structures which allow for inspection and routing of the cables. Duct banks and electrical manholes contain both CQE and Non-CQE cables. Only the duct banks and electrical manholes of Class I design that contain safety-related cables are within the scope of license renewal.

Operating Experience:

Site: A review of plant specific operating experience was conducted, including the review of Condition Reports and discussions with appropriate site personnel to identify AERM. These reviews concluded that the AERM identified by the FCS specific operating experience were consistent with those identified in NUREG-1801.

Industry: A review of industry-wide operating experience was conducted to identify aging effects requiring management. This included a review of operating experience issued during 2001. This review concluded that the AERM identified by industry operating experience were consistent with those identified in NUREG-1801.

On-Going: The on-going review of plant specific and industry-wide operating experience is conducted in accordance with the FCS Operating Experience Program.

3.5.1 AGING MANAGEMENT PROGRAMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

Table 3.5-1 shows the aging management programs evaluated in NUREG-1801 that are relied on for license renewal of Structures, and Component Supports at FCS. Note that this table only includes those components, materials and environments that are applicable to a PWR. Information on FCS specific components and materials, not listed in NUREG-1801 but included in the component group of this application, is included in the discussion column.

TABLE 3.5-1
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Common Components to all Types of PWR and BWR Containments				
Penetration sleeves, penetration bellows, and dissimilar metal welds	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	1. The metal fatigue time limited aging analyses are discussed in Section 4.6 of this application. 2. Consistent with NUREG-1801, this group includes penetration sleeves, penetration bellows, and dissimilar metal welds at FCS.
Penetration sleeves, bellows, and dissimilar metal welds.	Cracking due to cyclic loading, or crack initiation and growth due to SCC	Containment ISI and Containment leak rate test	Yes, detection of aging effects is to be evaluated	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The Containment Inservice Inspection Program (B.1.2) and the Containment Leak Rate Program (B.1.3) manage these aging effects. These programs are described in Appendix B of this application. 3. Consistent with NUREG-1801, this group includes stainless steel in ambient air at FCS. 4. Stress corrosion cracking for stainless steel bellows with dissimilar metal welds is applicable only if the susceptible material is exposed to a corrosive environment. The bellows at FCS are not exposed to a corrosive environment; therefore, Stress Corrosion Cracking is not an aging effect requiring management.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Penetration sleeves, penetration bellows, and dissimilar metal welds	Loss of material due to corrosion	Containment ISI and Containment leak rate test	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes carbon steel in ambient air at FCS.
Personnel airlock and equipment hatch	Loss of material due to corrosion	Containment ISI and Containment leak rate test	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-180, this group includes carbon steel in ambient air at FCS.
Personnel airlock and equipment hatch	Loss of leak tightness in closed position due to mechanical wear of locks, hinges and closure mechanism	Containment leak rate test and Plant Technical Specifications	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes carbon steel in ambient air at FCS.
Seals, gaskets, and moisture barriers	Loss of sealant and leakage through containment due to deterioration of joint seals, gaskets, and moisture barriers	Containment ISI and Containment leak rate test	No	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The equipment hatch gasket, made of neoprene, is the only component included in this component group at FCS.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
PWR Concrete (Reinforced and Prestressed) and Steel Containment				
Concrete elements: foundation, walls, dome.	Aging of accessible and inaccessible concrete areas due to leaching of calcium hydroxide, aggressive chemical attack, and corrosion of embedded steel	Containment ISI	Yes, if aging mechanism is significant for inaccessible areas	1. The FCS aging management review results are consistent with those reviewed and approved in NUREG-1801. 2. The Containment Inservice Inspection Program (B.1.2) manages the aging effects for these components. This program is described in Appendix B of this application. 3. Leaching of calcium hydroxide from reinforced concrete becomes significant only if the concrete is exposed to flowing water. The reinforced concrete at FCS is not exposed to flowing water. Even if reinforced concrete is exposed to flowing water, such leaching is not significant if the concrete is constructed to ensure that it is dense, well-cured, has low permeability, and that cracking is well controlled. Cracking is controlled through proper arrangement and distribution of reinforcing bars. The concrete at FCS was designed in accordance with ACI 318-63 (per USAR Section 5.3.1 and USAR Section 5.11.3.1) and has these characteristics. Therefore, a plant specific program for below-grade inaccessible areas is not required.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
				4. Below-grade exterior reinforced concrete at FCS is not exposed to an aggressive environment (pH less than 5.5), or to chloride or sulfate solutions beyond defined limits (greater than 500 ppm chloride, or greater than 1500 ppm sulfate). Periodic monitoring of below-grade water chemistry will be conducted during the period of extended operation to demonstrate that the below-grade environment is not aggressive. Therefore, a plant specific aging management program for below-grade inaccessible areas is not required.
Concrete elements: foundation	Cracks, distortion, and increases in component stress level due to settlement	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	1. The aging management results are consistent with those reviewed and approved in NUREG 1801. Applicable components are within the scope of the Structures Monitoring Program (B.2.10) described in Appendix B of this application. 2. The structures at FCS are supported on end-bearing steel pipe piles driven to bedrock. Settlement of the concrete subfoundation is not a plausible aging mechanism. A de-watering system is not relied upon for control of settlement at FCS.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Concrete elements: foundation	Reduction in foundation strength due to erosion of porous concrete subfoundation	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. Applicable components are within the scope of the Structures Monitoring Program (B.2.10) described in Appendix B of this application. 2. The reinforced concrete at FCS is not exposed to flowing water and a de-watering system is not relied upon for control of erosion of cement from porous concrete subfoundations.
Concrete elements: foundation, dome, and wall	Reduction of strength and modulus due to elevated temperature	Plant specific	Yes, for any portions of concrete containment that exceed specified temperature limits	Subsection CC-3400 of ASME Section III, Division 2, specifies the concrete temperature limits for normal operation or any other long-term period. The temperatures shall not exceed 150 deg F except for local areas that are allowed to have increased temperatures not to exceed 200 deg F. Per USAR Section 2.5.2.3, the record high temperature in the vicinity of FCS was 114 deg F in July 1936. This is below the temperature limit of 150 deg F. USAR Table 9.10-1 provides maximum building/room temperatures for the Auxiliary Building, Turbine Building, Containment, Control Room, Engine Driven Auxiliary Feedwater Pump Room, Radioactive Waste Processing Building, Chemistry and Radiation Protection Building, and Office/Cafeteria Addition.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
				The maximum indoor plant temperature in Table 9.10-1 is 120 deg F inside the main area of containment. This is below the temperature limit of 150 deg F. Per USAR Section 5.5.4, sleeve radiation fins and thermal sleeves (in conjunction with pipe insulation) are used to limit maximum temperature at the containment penetration sleeves to 150 deg F under operating conditions. The nuclear detector well cooling system cools the out-of-core neutron detectors, which are located in tubes or wells in the reactor compartment annulus between the lower portion of the reactor vessel and the biological shield, and maintains the shield concrete temperature below 150 deg F. Technical Specification Limiting Condition for Operation 2.13 requires that the annulus exit temperature from the nuclear detector cooling system shall not exceed a temperature found to correlate to 150 deg F concrete temperature. Therefore, no portions of concrete containment exceed specified temperature limits and no aging management is required.
Prestressed containment: tendons and anchorage components	Loss of prestress due to relaxation, shrinkage, creep, and elevated temperature	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	1. See Section 4.5 for the TLAA discussion of containment tendons. 2. Consistent with NUREG-1801 this group includes containment tendons and anchorage components at FCS.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Steel elements: liner plate, containment shell	Loss of material due to corrosion in accessible and inaccessible areas	Containment ISI and Containment leak rate test	Yes, if corrosion is significant for inaccessible areas	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. 2. Corrosion for inaccessible areas (embedded containment liner) is not significant because: a. Concrete meeting the requirements of ACI 318 or 349 and the guidance of 201.2R was used for the containment concrete in contact with the embedded containment liner. b. The concrete is monitored to ensure that it is free of penetrating cracks that provide a path for water seepage to the surface of the containment shell or liner. c. The moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with IWE requirements. d. Borated water spills and water ponding on the containment concrete floor are not common and when detected are cleaned up in a timely manner.
Steel elements: protected by coating	Loss of material due to corrosion in accessible areas only	Protective coating monitoring and maintenance	No	The combinations of components, materials and environments identified in NUREG-1801 are not applicable to FCS. Protective coatings are not relied on to manage the effects of aging at FCS.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Prestressed containment: tendons and anchorage components	Loss of material due to corrosion of prestressing tendons and anchorage components	Containment ISI	No	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes containment tendons and anchorage components at FCS.
Concrete elements: foundation, dome, and wall	Scaling, cracking, and spalling due to freeze-thaw; expansion and cracking due to reaction with aggregate	Containment ISI	No	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes concrete exposed to ambient air and below grade concrete at FCS.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Class I Structures				
All Groups except Group 6: accessible interior/ exterior concrete & steel components	All types of aging effects	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	1. The aging management results are consistent with those reviewed and approved in NUREG 1801. Applicable components are within the scope of the Structures Monitoring Program (B.2.10) described in Appendix B of this Application 2. As described in NUREG-1557, freeze/thaw does not cause loss of material from reinforced concrete in foundations, and in above and below grade exterior concrete, for plants located in a geographic region of negligible weathering conditions (weathering index <100 day-inch/yr). Loss of material from such concrete is not significant at plants located in areas in which weathering conditions are severe (weathering index >500 day-inch/yr) or moderate (100-500 day-inch/yr), provided that the concrete mix design meets the air content (entrained air 3-6%) and water-to-cement ratio (0.35-0.45) specified in ACI 318-63 or ACI 349-85. The weathering index for FCS is >500 day-inch/yr. The concrete mix design specified a water-to-cement ratio of 0.38 and air entrainment of 4.75% + 0.75% for Class A concrete for FCS. It specified a water-to-cement ratio of 0.44 and air entrainment of 5.00% + 1.00% for Class B concrete. Therefore, the conditions of NUREG-1801, Volume 2, Chapter III are satisfied.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
				3. Leaching of calcium hydroxide from reinforced concrete becomes significant only if the concrete is exposed to flowing water. Leaching is not significant if the concrete is constructed to ensure that it is dense, well-cured, has low permeability, and that cracking is well controlled. Cracking is controlled through proper arrangement and distribution of reinforcing bars. The concrete at FCS was designed in accordance with ACI 318-63 (per USAR Section 5.3.1 and USAR Section 5.11.3.1) and has these characteristics. Therefore, the conditions of NUREG-1801 Volume 2 Chapter III are satisfied. 4. Investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295-54 or ASTM C227-50 demonstrated that the aggregates used in the construction of FCS do not react within reinforced concrete. Concrete for FCS was constructed in accordance with ACI 201.2R-77. C. Therefore, the conditions of NUREG-1801, Volume 2, Chapter III are satisfied.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
				5. Per NUREG-1557, corrosion of embedded steel is not significant for concrete structures above or below grade that are exposed to a non-aggressive environment. A non-aggressive environment, as defined by NUREG-1557, is one with a pH greater than 11.5 or chlorides less than 500 ppm. NUREG-1557 also concludes that corrosion of embedded steel is not significant for concrete structures exposed to an aggressive environment but have a low water-to-cement ratio, adequate air entrainment, and designed in accordance with ACI 318-63 or ACI 349-85. A low water-to-cement ratio is defined as 0.35 to 0.45 and adequate air entrainment is defined as 3 to 6 percent. The concrete at FCS is not exposed to aggressive river water or groundwater. There is no heavy industry in the area whose emissions would cause degradation to concrete or steel. The concrete that surrounds the embedded steel has a pH greater than or equal to 12.5. The concrete mix design specified a water-to-cement ratio of 0.38 and air entrainment of 4.75% + 0.75% for Class A concrete. It specified a water-to-cement ratio of 0.44 and air entrainment of 5.00% + 1.00% for Class B concrete.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
				Class C concrete was only used for radiation shields; therefore, would not be exposed to an environment that would promote corrosion of embedded steel. The concrete at FCS was designed in accordance with ACI 318-63 (per USAR Section 5.3.1, Revision 0 and USAR Section 5.11.3.1, Revision 2). Therefore, the conditions of NUREG-1801, Volume 2, Chapter III are satisfied and aging management is not required. Below-grade exterior reinforced concrete at FCS is not exposed to an aggressive environment (pH less than 5.5), or to chloride or sulfate solutions beyond defined limits (greater than 500 ppm chloride, or greater than 1500 ppm sulfate). Periodic monitoring of below-grade water chemistry will be conducted during the period of extended operation to demonstrate that the below-grade environment is not aggressive. Therefore, the conditions of NUREG-1801, Volume 2, Chapter III are satisfied.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
				6. Aggressive chemical attack on reinforced concrete is not significant if the concrete is exposed to a nonaggressive environment. A non-aggressive environment, as defined by GALL, is one with a pH greater than 5.5, chlorides less than 500 ppm, or sulfates less than 1500 ppm. The concrete at FCS is not exposed to aggressive river water or groundwater. There is no heavy industry in the area whose emissions would cause degradation to concrete or steel. Therefore, the conditions of NUREG-1801, Volume 2, Chapter III are satisfied. 7. The structures at FCS are supported on end-bearing steel pipe piles driven to bedrock. Settlement of the concrete subfoundation are not plausible aging mechanisms. A de-watering system is not relied upon for control of settlement at FCS. 8. The reinforced concrete at FCS is not exposed to flowing water and a de-watering system is not relied upon for control of erosion of cement from porous concrete subfoundations,

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
				9. Subsection CC-3400 of ASME Section III, Division 2, specifies the concrete temperature limits for normal operation or any other long-term period. The temperatures shall not exceed 150 deg F except for local areas that are allowed to have increased temperatures not to exceed 200 deg F. Per USAR Section 2.5.2.3, the record high temperature in the vicinity of FCS was 114 deg F in July 1936. This is below the temperature limit of 150 deg F. USAR Table 9.10-1 provides maximum building/room temperatures for the Auxiliary Building, Turbine Building, Containment, Control Room, Engine Driven Auxiliary Feedwater Pump Room, Radioactive Waste Processing Building, Chemistry and Radiation Protection Building, and Office/Cafeteria Addition. The maximum indoor plant temperature in Table 9.10-1 is 120 deg F inside the main area of Containment. This is below the temperature limit of 150 deg F. Per USAR Section 5.5.4, sleeve radiation fins and thermal sleeves (in conjunction with pipe insulation) are used to limit maximum temperature at the containment penetration sleeves to 150 deg F under operating conditions.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
				The nuclear detector well cooling system cools the out-of-core neutron detectors, which are located in tubes or wells in the reactor compartment annulus between the lower portion of the reactor vessel and the biological shield, and maintains the shield concrete temperature below 150 deg F. Technical Specification Limiting Condition for Operation 2.13 requires that the annulus exit temperature from the nuclear detector cooling system shall not exceed a temperature found to correlate to 150 deg F concrete temperature. Therefore, no portions of concrete containment exceed specified temperature limits and no aging management is required.
Groups 1-3, 5, 7-9: inaccessible concrete components, such as exterior walls below grade and foundation	Aging of inaccessible concrete areas due to aggressive chemical attack, and corrosion of embedded steel	Plant-specific	Yes, if an aggressive below-grade environment exists	Below-grade exterior reinforced concrete at FCS is not exposed to an aggressive environment (pH less than 5.5), or to chloride or sulfate solutions beyond defined limits (greater than 500 ppm chloride, or greater than 1500 ppm sulfate). Periodic monitoring of below grade water chemistry will be conducted during the period of extended operation to demonstrate that the below-grade environment is not aggressive. Therefore, the conditions of NUREG-1801, Volume 2, Chapter III are satisfied and aging management is not required.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Group 6: all accessible/inaccessible concrete, steel, and earthen components	All types of aging effects, including loss of material due to abrasion, cavitation, and corrosion	Inspection of Water-Control Structures or FERC/US Army Corps of Engineers dam inspections and maintenance	No	The components identified in NUREG-1801 are not applicable to FCS.
Group 5: liners	Crack initiation and growth from SCC and loss of material due to crevice corrosion	Water Chemistry Program and Monitoring of spent fuel pool water level	No	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes stainless steel in water at FCS. 3. In addition to the components identified in NUREG-1801, this group includes the reactor cavity liner, the reactor cavity seal ring and the fuel transfer penetration at FCS. 4. In addition to monitoring of spent fuel pool level the Periodic Surveillance and Preventive Maintenance Program (B.2.8) performs a leak rate analysis of the refueling canal liner.
Groups 1-3, 5, 6: all masonry block walls	Cracking due to restraint, shrinkage, creep, and aggressive environment	Masonry Wall	No	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. The masonry wall program is included in the FCS Structures Monitoring Program (B.2.10). 2. Consistent with NUREG-1801, this group includes concrete block in ambient air at FCS.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Groups 1-3, 5, 7-9: foundation	Cracks, distortion, and increases in component stress level due to settlement	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. Applicable components are within the scope of the Structures Monitoring Program (B.2.10) described in Appendix B of this application 2. The structures at FCS are supported on end-bearing steel pipe piles driven to bedrock. Settlement of the concrete subfoundation are not plausible aging mechanisms. A de-watering system is not relied upon for control of settlement at FCS. 3. Consistent with NUREG-1801, this group includes reinforced concrete at FCS.
Groups 1-3, 5-9: foundation	Reduction in foundation strength due to erosion of porous concrete subfoundation	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. Applicable components are within the scope of the Structures Monitoring Program (B.2.10) described in Appendix B of this application 2. The reinforced concrete at FCS is not exposed to flowing water and a de-watering system is not relied upon for control of erosion of cement from porous concrete subfoundations. 3. Consistent with NUREG-1801, this group includes reinforced concrete at FCS.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Groups 1-5: concrete	Reduction of strength and modulus due to elevated temperature	Plant-specific	Yes, for any portions of concrete that exceed specified temperature limits	Subsection CC-3400 of ASME Section III, Division 2, specifies the concrete temperature limits for normal operation or any other long-term period. The temperatures shall not exceed 150 deg F except for local areas that are allowed to have increased temperatures not to exceed 200 deg F. Per USAR Section 2.5.2.3, the record high temperature in the vicinity of FCS was 114 deg F in July 1936. This is below the temperature limit of 150 deg F. USAR Table 9.10-1 provides maximum building/room temperatures for the Auxiliary Building, Turbine Building, Containment, Control Room, Engine Driven Auxiliary Feedwater Pump Room, Radioactive Waste Processing Building, Chemistry and Radiation Protection Building, and Office/Cafeteria Addition. The maximum indoor plant temperature in Table 9.10-1 is 120 deg F inside the main area of Containment. This is below the temperature limit of 150 deg F. Per USAR Section 5.5.4, sleeve radiation fins and thermal sleeves (in conjunction with pipe insulation) are used to limit maximum temperature at the containment penetration sleeves to 150 deg F under operating conditions. The nuclear detector well cooling system cools the out-of-core neutron detectors, which are located in tubes or wells in the reactor compartment annulus between the lower portion of the reactor vessel and the biological shield, and maintains the shield concrete temperature below 150 deg F.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
				Technical Specification Limiting Condition for Operation 2.13 requires that the annulus exit temperature from the nuclear detector cooling system shall not exceed a temperature found to correlate to 150 deg F concrete temperature. Therefore, no portions of concrete containment exceed specified temperature limits and no aging management is required.
Groups 7, 8: liners	Crack Initiation and growth due to SCC; Loss of material due to crevice corrosion	Plant-specific	Yes	The combinations of components, materials and environments identified in NUREG-1801 are not applicable to FCS.
Component Supports				
All Groups: support members: anchor bolts, concrete surrounding anchor bolts, welds, grout pad, bolted connections, etc.	Aging of component supports	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. Applicable components are within the scope of the Structures Monitoring Program (B.2.10) described in Appendix B of this application. 2. Consistent with NUREG-1801, this group includes carbon steel and reinforced concrete exposed to ambient air at FCS. 3. In addition to the materials identified in NUREG-1801, this group includes stainless steel at FCS. 4. In addition to the components identified in NUREG-1801, this group includes carbon steel vibration isolators at FCS.

TABLE 3.5-1 (Continued)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	A CLB fatigue analysis does not exist at FCS Station.
All Groups: support members: anchor bolts, welds	Loss of material due to boric acid corrosion	Boric acid corrosion	No	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes carbon steel exposed to ambient air at FCS.
Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds, spring hangers, guides, stops, and vibration isolators	Loss of material due to environmental corrosion; loss of mechanical function due to corrosion, distortion, dirt, overload, etc.	ISI	No	1. The aging management results are consistent with those reviewed and approved in NUREG-1801. 2. Consistent with NUREG-1801, this group includes carbon steel exposed to ambient air at FCS. 3 In addition to the materials identified in NUREG-1801, this group includes stainless steel at FCS.
Group B1.1: high strength low-alloy bolts	Crack initiation and growth due to SCC	Bolting integrity	No	1. The aging management results are consistent (with deviation) with the results documented in NUREG-1801. 2. The Bolting Integrity Program discussed in Appendix B of this application includes an alternative means of managing cracking due to SCC. 3. Consistent with NUREG-1801, this group includes high strength low-alloy bolts exposed to ambient air at FCS.

3.5.2 COMPONENTS OR AGING EFFECTS THAT ARE NOT ADDRESSED IN NUREG-1801

Table 3.5-2 contains Containment, Structures and Component Supports aging management review results that are not addressed in NUREG-1801. This table includes the component types, materials, environments, aging effects requiring management, and the programs and activities for managing aging.

TABLE 3.5-2
AGING MANAGEMENT PROGRAMS FOR CONTAINMENT, STRUCTURES AND COMPONENTS THAT ARE NOT ADDRESSED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Removable slab lifting devices	Bronze	Plant Indoor Air	None	Not Applicable
Intake Structure sluice gate operator gland, pump gland and gland bolting	Bronze, brass	Raw Water	Loss of Material • Crevice and pitting corrosion and MIC due to stagnant conditions • Galvanic corrosion due to the conductivity of the process fluid and the presence of dissimilar metals in contact	Structures Monitoring Program (B.2.10)
Class A pipe piles are partially filled with soil during placement and then are filled with sand to the point four feet below the top of the pile. The remaining four feet are then filled with concrete.	Carbon Steel	Below Grade	None	Not Applicable
Class B pipe piles	Carbon Steel	Below Grade	None	Not Applicable
Diesel engine fuel oil storage tank H-piles	Carbon Steel	Below Grade	None	Not Applicable
Class B pipe piles	Carbon Steel	Concrete	None	Not Applicable

TABLE 3.5-2 (CONTINUED)
AGING MANAGEMENT PROGRAMS FOR CONTAINMENT, STRUCTURES AND COMPONENTS THAT ARE NOT ADDRESSED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Class A pipe piles are partially filled with soil during placement and then are filled with sand to the point four feet below the top of the pile. The remaining four feet are then filled with concrete.	Carbon Steel	Concrete/Sand/Soil	None	Not Applicable
Subcomponent-Manhole flange, Structural steel	Carbon Steel	Outside Air	Loss of Material General corrosion due to the exposure of external surfaces to varying levels of humidity	Periodic Surveillance and Preventive Maintenance Program (B.2.8)
Intake Structure carbon steel pipe, pipe sleeve, flange and pipe casing floor penetration	Carbon Steel	Raw Water	Loss of Material • Crevice and general corrosion due to oxygenated raw water environment • Pitting corrosion due to oxygenated raw water environment and stagnant or low flow conditions • Galvanic corrosion due to the conductivity of the process fluid and dissimilar metals in contact • MIC due to exposure to microbiological activity	Periodic Surveillance and Preventive Maintenance Program (B.2.8)

TABLE 3.5-2 (CONTINUED)
AGING MANAGEMENT PROGRAMS FOR CONTAINMENT, STRUCTURES AND COMPONENTS THAT ARE NOT ADDRESSED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Intake Structure cast iron stuffing box floor penetration	Cast Iron	Raw Water	Loss of Material • Crevice and general corrosion due to oxygenated raw water environment • Pitting corrosion due to oxygenated raw water environment and stagnant or low flow conditions • Galvanic corrosion due to the conductivity of the process fluid and dissimilar metals in contact • MIC due to exposure to microbiological activity	Structures Monitoring Program (B.2.10)
Concrete caissons	Concrete	Below Grade	None	Not Applicable

TABLE 3.5-2 (CONTINUED)
AGING MANAGEMENT PROGRAMS FOR CONTAINMENT, STRUCTURES AND COMPONENTS THAT ARE NOT ADDRESSED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Concrete encased in Class B pipe piles is protected from aggressive environments. The concrete has a compressive strength of 4000 psi, a maximum water-to-cement ratio of 6 gallons/sack or 0.53, and the minimum cement content is 6.50 sacks/cubic yard. The aggregate used has been specified to be non-reactive when mixed with portland cement and water.	Concrete	Carbon Steel	None	Not Applicable
Pneumatic flood panel seals	EPDM Rubber	Plant Indoor Air	None	Not Applicable
Intake Structure EDPM rubber Link-Seal	EPDM Rubber	Raw Water	Change in Material Properties due to chemical exposure	Structures Monitoring Program (B.2.10)
Intake Structure raw water pump rubber foundation seal	EPDM Rubber	Raw Water	Change in Material Properties due to chemical exposure	General Corrosion of External Surfaces Program (B.3.3)
Intake Structure sand and gravel surrounding the diesel fire pump fuel storage tank	Gravel	Ambient Air Protected from Weather	None	Not Applicable

TABLE 3.5-2 (CONTINUED)
AGING MANAGEMENT PROGRAMS FOR CONTAINMENT, STRUCTURES AND COMPONENTS THAT ARE NOT ADDRESSED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Manhole covers and flange	Gray Cast Iron	Ambient Air	Loss of Material General corrosion and selective leaching due to the exposure of external surfaces to varying levels of humidity	Periodic Surveillance and Preventive Maintenance Program (B.2.8)
Flood panel seals	Neoprene	Plant Indoor Air	Change in Material Properties Due to the prolonged exposure of neoprene to temperatures above 95 deg F	Periodic Surveillance and Preventive Maintenance Program (B.2.8)
Flood panel seals	Neoprene	Plant Indoor Air	Cracking Due to the prolonged exposure of neoprene to temperatures above 95 deg F	Periodic Surveillance and Preventive Maintenance Program (B.2.8)
Manhole foam blocks	Polyurethane foam	Ambient Air Protected from Weather	Cracking Due to vibration, movement, and shrinkage	Periodic Surveillance and Preventive Maintenance Program (B.2.8)
Manhole foam blocks	Polyurethane foam	Ambient Air Protected from Weather	Separation Due to vibration, movement, and shrinkage	Periodic Surveillance and Preventive Maintenance Program (B.2.8)

TABLE 3.5-2 (CONTINUED)
AGING MANAGEMENT PROGRAMS FOR CONTAINMENT, STRUCTURES AND COMPONENTS THAT ARE NOT ADDRESSED IN NUREG-1801

Component Types	Material	Environment	AERMs	Program/Activity
Auxiliary building pressure relief panels	PVC	Outside Air	Change in Material Properties Due to ultraviolet (UV) radiation exposure	Structures Monitoring Program (B.2.10)
Auxiliary building pressure relief panels	PVC	Outside Air	Cracking Due to ultraviolet (UV) radiation exposure	Structures Monitoring Program (B.2.10)
Intake Structure stainless steel raw water pump gland bolting	Stainless Steel	Raw Water	Loss of Material • Crevice corrosion due to the presence of dissolved oxygen and impurities • MIC due to exposure to microbiological activity • Pitting corrosion due to (1) stagnant or low-flow conditions, and (2) halide ions, chlorides, bromides or hypochlorites	General Corrosion of External Surfaces Program (B.3.3)
Stainless steel threaded fasteners	Stainless Steel	Ambient Air	None	Not Applicable
Trisodium phosphate baskets	Stainless Steel	Ambient Air	None	Not Applicable
Boot clamps for auxiliary building boot sealed fire penetration barrier	Stainless Steel	Ambient Air	None	Not Applicable

3.6 AGING MANAGEMENT OF ELECTRICAL AND INSTRUMENTATION AND CONTROLS

The components for FCS evaluated in this section of the application consist of the electrical cables, connectors, splices, fuse blocks, terminal blocks, electrical penetrations, and electrical bus bars subject to aging management review.

Cables and their associated connectors perform the function of providing electrical energy (either continuously or intermittently) to power various equipment and components throughout the plant. Cables and connectors associated with the 10 CFR 50.49 program (Environmental Qualification) are addressed either as short lived, periodically replaced, or long-lived Time Limited Aging Analysis (TLAA) candidates; as such, those candidates are not included in the set of cables and connectors requiring additional aging management review.

Electrical penetrations electrically connect specified sections of an electrical circuit through the containment boundary to deliver voltage, current or signal while maintaining the integrity of containment. The pigtail at each end of the penetration is connected to the field cable by industry accepted methods such as connectors, terminal blocks or splice connections.

Bus bars electrically connect specified sections of an electrical circuit to deliver voltage, current or signal. The standoffs support the electrical bus bars. This assessment includes the bus bars located in the 480-volt motor control centers.

Operating Experience:

Site: A review of plant specific operating experience was conducted, including the review of Condition Reports and discussions with appropriate site personnel to identify AERM. These reviews concluded that the AERM identified by the FCS specific operating experience were consistent with those identified in NUREG-1801.

Industry: A review of industry-wide operating experience was conducted to identify aging effects requiring management. This included a review of operating experience issued during 2001. This review concluded that the AERM identified by industry operating experience were consistent with those identified in NUREG-1801.

On-Going: The on-going review of plant specific and industry-wide operating experience is conducted in accordance with the FCS Operating Experience Program.

3.6.1 AGING MANAGEMENT PROGRAMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

Table 3.6-1 shows the aging management groups (combinations of components, materials and environments) and the aging management programs evaluated in NUREG-1801 that are relied on for license renewal of the Electrical and Instrumentation and Controls Systems at FCS. Information on FCS specific components, materials and aging effects, not listed in NUREG-1801 but included in the component group of this application, is included in the discussion column.

TABLE 3.6-1
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR ELECTRICAL AND INSTRUMENTATION AND CONTROLS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Electrical equipment subject to 10 CFR 50.49 environmental qualification (EQ) requirements	Degradation due to various aging mechanisms	Environmental qualification of electric components	Yes, TLAA	The environmental qualification time limited aging analyses are discussed in Section 4.4 of this application.
Electrical cables and connections not subject to 10 CFR 50.49 EQ requirements	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure caused by thermal/ thermoxidative degradation of organics; radiolysis and photolysis (ultraviolet [UV] sensitive materials only) of organics; radiation-induced oxidation; moisture intrusion	Aging management program for electrical cables and connections not subject to 10 CFR 50.49 EQ requirements	No	Addressed in FCS Plant Specific Non-EQ Cable Aging management Program (B.3.4), which is described in Appendix B of this application.
Electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor insulation resistance	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR; electrical failure caused by thermal/ thermoxidative degradation of organics; radiation-induced oxidation; moisture intrusion	Aging management program for electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements	No	Addressed in FCS Plant Specific Non-EQ Cable Aging management Program (B.3.4), which is described in Appendix B of this application.

TABLE 3.6-1 (CONTINUED)
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR ELECTRICAL AND INSTRUMENTATION AND CONTROLS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR FCS LICENSE RENEWAL

Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Inaccessible medium-voltage (2kV to 15kV) cables (e.g., installed in conduit or direct buried) not subject to 10 CFR 50.49 EQ requirements	Formation of water trees; localized damage leading to electrical failure (breakdown of insulation) caused by moisture intrusion and water trees	Aging management program for inaccessible medium-voltage cables not subject to 10 CFR 50.49 EQ requirements	No	Modifications were made to the Duct Banks to preclude moisture intrusion; therefore, there is no aging effect requiring management.
Electrical connectors not subject to 10 CFR 50.49 EQ requirements that are exposed to borated water leakage	Corrosion of connector contact surfaces caused by intrusion of borated water	Boric acid corrosion	No	1. The aging management results are consistent with the results documented in NUREG-1801. 2. Consistent with NUREG-1801, this group includes connectors exposed to borated water leakage at FCS.

3.6.2 COMPONENTS OR AGING EFFECTS THAT ARE NOT ADDRESSED IN NUREG-1801

All components subject to aging management review and aging effects for FCS Electrical and Instrumentation and Controls systems are addressed in Section 3.6.1above.

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).

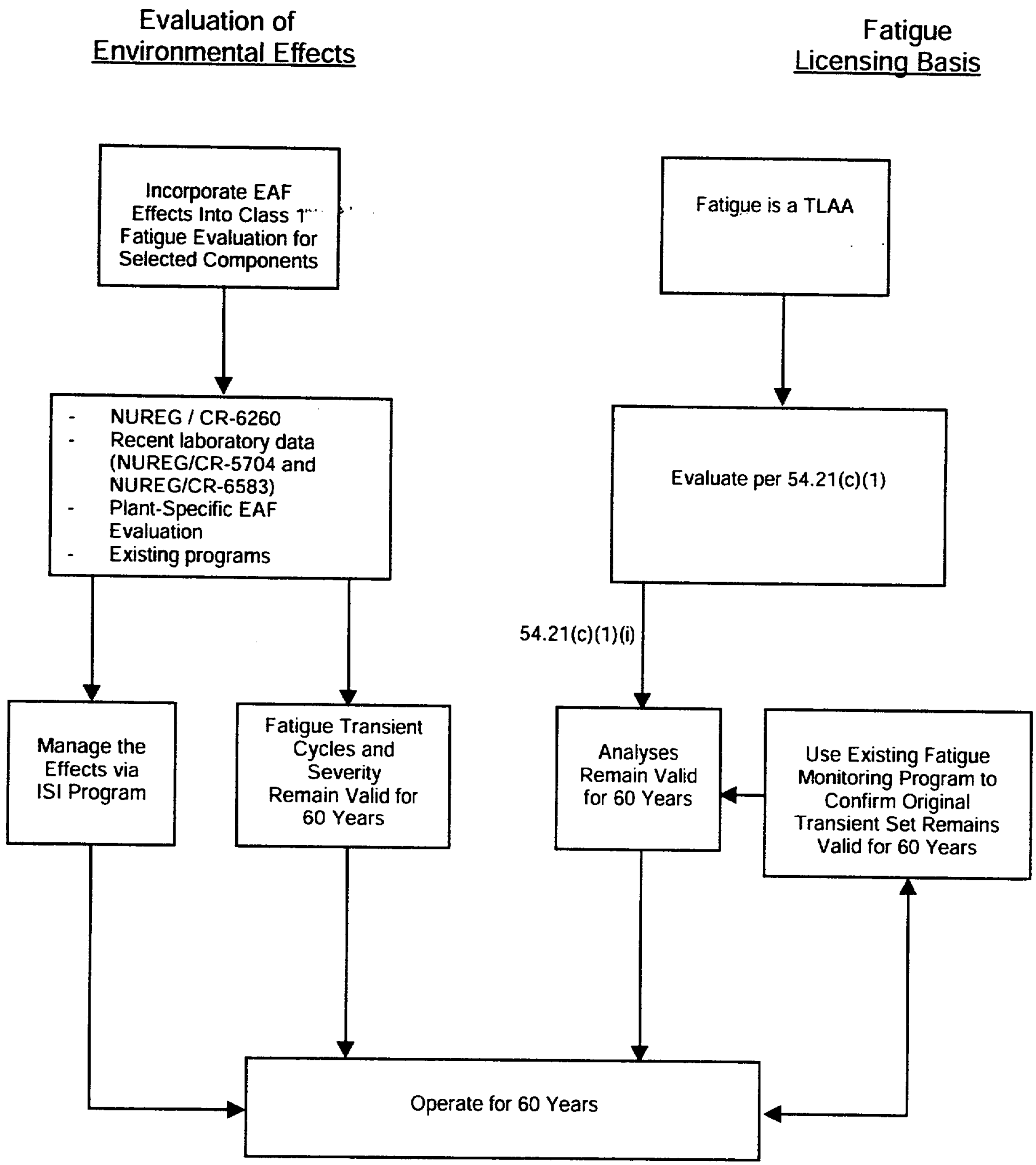
As noted above, several pressurizer surge line welds have been ultrasonically examined. No reportable indications have been identified. In addition, OPPD plans to inspect the limiting surge line welds during the third and fourth inservice inspection interval, and prior to entering the extended period 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

4.2-1 Letter from OPPD (WG Gates) to NRC (Document Control Desk) dated 8/3/2000 (LIC-00-0064)

4.2-2 Letter from OPPD (SK Gambhir) to NRC (Document Control Desk) dated 11/17/2000 (LIC-00-0096)

4.2-3 Letter from OPPD (SK Gambhir) to NRC (Document Control Desk) dated 2/14/2001 (LIC-01-0018)

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

4.3-2 Generic Safety Issue 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life," U. S. Nuclear Regulatory Commission.

4.3-3 Memorandum, Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, to William D. Travers, Executive Director of Operations - "Closeout of Generic Safety Issue 190, Fatigue Evaluation of Metal Components for 60 Year Plant Life," U. S. Nuclear Regulatory Commission, December 26, 1999.

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.

4.3-5 EPRI Report No. TR-107515, "Evaluation of Thermal Fatigue Effects on Systems Requiring Aging Management Review for License Renewal for the Calvert Cliffs Nuclear Power Plant," Electric Power Research Institute, January 1998.

4.3-6 EPRI Report No. TR-110043, "Evaluation of Environmental Fatigue Effects for a Westinghouse Nuclear Power Plant," Electric Power Research Institute, April 1998.

4.3-7 EPRI Report No. TR-110356, "Evaluation of Environmental Thermal Fatigue Effects on Selected Components in a Boiling Water Reactor Plant," Electric Power Research Institute, April 1998.

4.3-8 EPRI Report No. TR-107943, "Environmental Fatigue Evaluations of Representative BWR Components," Electric Power Research Institute, May 1998.

4.3-9 NUREG/CR-6260 (INEL-95/0045), "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," U. S. Nuclear Regulatory Commission, March 1995.

4.3-10 NUREG/CR-5999 (ANL-93/3), "Interim Fatigue Design Curves for Carbon, Low-Alloy, and Austenitic Stainless Steels in LWR Environments," U. S. Nuclear Regulatory Commission, August 1993.

4.3-11 NUREG/CR-6583 (ANL-97/18), "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," U. S. Nuclear Regulatory Commission, March 1998.

4.3-12 NUREG/CR-5704 (ANL-98/31), "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," U. S. Nuclear Regulatory Commission, April 1999.

APPENDIX A – UPDATED SAFETY ANALYSIS REPORT (USAR) SUPPLEMENT

A.1 INTRODUCTION

The application for a renewed operating license is required by 10 CFR 54.21(d) to include "an FSAR Supplement." This appendix provides that supplement for the FCS USAR. Section 2 of this appendix contains a summarized description of the programs and activities for managing the effects of aging. Section 3 of this appendix contains a summary of the evaluation of time-limited aging analyses (TLAAs) for the period of extended operation.

A.2 PROGRAMS AND ACTIVITIES FOR MANAGING THE EFFECTS OF AGING

This section provides summaries of the programs and activities credited for managing the effects of aging, in alphabetical order. The FCS Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Section A.2 of NUREG 1800, *Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants*, published July 2001. The Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls, and is applicable to the safety-related and non-safety-related structures, systems, and components that are within the scope of license renewal.

A.2.1 ALLOY 600 PROGRAM

The Alloy 600 Program includes a primary water stress corrosion cracking (PWSCC) susceptibility assessment to identify susceptible components and inservice inspection (ISI) of Reactor Coolant System penetrations to monitor PWSCC and its effect on the intended function of the component. For susceptible penetrations and locations, the program includes an industry-wide, integrated, long-term inspection program based on the industry response to NRC Generic Letter (GL) 97-01, *Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations.*

A.2.2 BOLTING INTEGRITY PROGRAM

The Bolting Integrity Program includes periodic inspection of closure and structural bolting for indications of potential problems, including loss of material, crack initiation, and loss of preload. The program implements guidelines on materials selection, strength and hardness properties, installation procedures, lubricants and sealants, corrosion considerations in the selection and installation of pressure-retaining bolting, and enhanced inspection techniques. The program is based on (1) the bolting integrity program delineated in NUREG-1339, *Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants; (2)* industry's recommendations

delineated in ERPRI NP-5769, *Degradation and Failure of Bolting in Nuclear Power Plants*, with the exceptions noted in NUREG-1339 for safety-related bolting; (3) EPRI TR-104213, *Bolted Joint Maintenance and Application Guide*, for pressure retaining bolting and structural bolting; and (4) routine examinations and inspections performed in accordance with the requirements of ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*.

A.2.3 BORIC ACID CORROSION PREVENTION PROGRAM

The Boric Acid Corrosion (BAC) Prevention Program implements administrative controls to (1) perform visual inspections of external surfaces that are potentially exposed to borated water leakage, (2) ensure timely discovery of leak path and removal of the boric acid residues, (3) perform assessments of degradation, and (4) perform follow-up inspections for adequacy of corrective actions. The program is implemented in response to NRC GL 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants*.

A.2.4 BURIED SURFACES EXTERNAL CORROSION PROGRAM

The Buried Surfaces External Corrosion Program provides for inspection of buried piping, tanks, and valves whenever they are uncovered due to excavation for maintenance or modifications. Piping and component coatings and wrappings will be inspected for degraded conditions that could be indicative of possible surface corrosion of the protected metal beneath. The scope and periodicity of inspections will be established and/or adjusted based on the inspection results.

A.2.5 CHEMISTRY PROGRAM

The FCS Chemistry Program controls water chemistry to minimize contaminant concentration and provide chemical additions, such as corrosion inhibitors and biocides, to mitigate aging effects due to corrosion. The program includes specifications for chemical species, limits, representative sampling and analysis frequencies, and corrective actions for control of water chemistry. The program is based on EPRI Guidelines TR-105714, *PWR Primary Water Chemistry Guidelines*, for primary water chemistry, TR-102134, *PWR Secondary Water Chemistry Guideline*, for secondary water chemistry, and TR-107396, *Closed Cooling Water Chemistry Guideline*, for closed-cycle cooling water corrosion inhibitor concentration.

A.2.6 CONTAINMENT INSERVICE INSPECTION PROGRAM

The Containment Inservice Inspection Program implements the examination requirements of ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWE, *Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants,* and Subsection IWL, *Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants,* for the containment structure and support components. The ASME Section XI,

Subsection IWL program consists of periodic visual inspection of concrete surfaces and periodic visual inspection and sample tendon testing for signs of degradation, assessment of damage, and corrective actions. Measured tendon lift-off forces are compared to predicted tendon forces calculated in accordance with NRC Regulatory Guide (RG) 1.35, *Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments*. The ASME Section XI, Subsection IWE program consists of periodic visual, surface, and volumetric inspection of pressure retaining components for signs of degradation, assessment of damage and corrective actions. This program is in accordance with the requirements of 10 CFR 50.55a and ASME Section XI, Subsections IWE and IWL, 1992 edition including 1992 addenda.

A.2.7 CONTAINMENT LEAK RATE PROGRAM

The Containment Leak Rate Program implements the requirements of 10 CFR Part 50, Appendix J, as well as those examination requirements needed to comply with ASME Section XI, Subsection IWE, RG 1.163, *Performance-Based Containment Leak-Test Program,* and NEI 94-01, *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J,* Rev. 0 for the containment structure and pressure retaining components. The program consists of monitoring of leakage rates through containment liner/welds, penetrations, fittings, and other access openings for detecting degradation of the containment pressure boundary. Corrective actions are taken if leakage rates exceed acceptance criteria.

A.2.8 COOLING WATER CORROSION PROGRAM

The Cooling Water Corrosion Program monitors and detects aging effects through inspection and nondestructive evaluations. The program also involves some mitigation activities of periodic flushing and draining. The program's aging management activities are based on EPRI TR-107396, *Closed Cooling Water Chemistry Guideline,* for closed-cycle cooling water systems and NRC GL 89-13, *Service Water System Problems Affecting Safety-Related Equipment,* for open-cycle cooling water systems.

A.2.9 DIESEL FUEL MONITORING AND STORAGE PROGRAM

The FCS Diesel Fuel Monitoring and Storage Program monitors and controls diesel fuel quality regarding water and other contaminants in accordance with the guidelines of ASTM Standards D2709, *Standard Test Method for Water and Sediment in Middle Distillate Fuels by Centrifuge,* and D4057, *Standard Practice for Manual Sampling of Petroleum and Petroleum Products.* Exposure to fuel oil contaminants such as water and microbiological organisms is minimized by periodic cleaning and draining of tanks and by verifying the quality of new fuel oil before its introduction into the storage tanks.

A.2.10 FATIGUE MONITORING PROGRAM

The Fatigue Monitoring Program provides for the monitoring of reactor coolant and associated systems thermal fatigue, pressurizer surge line thermal stratification, and thermal fatigue of selected Class II and III components over the life of the plant to ensure that their operation does not result in exceeding the number of design basis transients included in the design basis of their respective design codes. It will be centered on the industry's automated cycle counting software, FatiguePro. Plant locations that cannot be counted automatically will continue to be counted manually. An FCS site specific evaluation is being performed to address environmental fatigue. Appropriate program enhancements will be made prior to the period of extended operation based on the evaluation results.

A.2.11 FIRE PROTECTION PROGRAM

The FCS Fire Protection Program provides administrative requirements for ensuring the operability of fire protection equipment required to ensure plant safe shutdown. The program includes visual inspections, system flushing, and performance tests of fire barriers (penetration seals, fire doors, walls, ceilings, and floors), fire suppression system components (piping, valves, nozzles, yard hydrants and hose stations, sprinkler heads, and halon systems and cylinders), and the diesel fire pump. The FCS Fire Protection Program includes the requirements identified in Appendix A to NRC Branch Technical Position APCSB 9.5-1 and 10 CFR 50 Appendix R, Section III.G, J, and O and is further described in Section 9.11 of the USAR.

A.2.12 FLOW ACCELERATED CORROSION PROGRAM

The FCS Flow Accelerated Corrosion (FAC) Program implements administrative controls to conduct appropriate analysis and baseline inspections, determine extent of thinning, replace/repair components, and perform follow-up inspections to confirm or quantify and take longer-term corrective actions. The program relies on implementation of EPRI guidelines of NSAC-202L-R2, *Recommendations for an Effective Flow-Accelerated Corrosion Program.*

A.2.13 GENERAL CORROSION OF EXTERNAL SURFACES PROGRAM

The General Corrosion of External Surfaces Program implements systematic inspections and observations to detect corrosion of external surfaces and conditions that can result in corrosion such as damaged coatings and fluid leaks. Inspections and observations include (1) rounds by operators, (2) system engineer walkdowns, and (3) refueling interval inspections inside containment in accordance with RG 1.54, *Quality Assurance Requirements for Protective Coatings Applied to Water - Controlled Nuclear Power Plants.*

A.2.14 INSERVICE INSPECTION PROGRAM

The Fort Calhoun Station Inservice Inspection Program implements the examination requirements of the ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, Subsections IWB, IWC, IWD, IWF. The program consists of periodic volumetric, surface and/or visual examination of components and their supports for assessment, signs of degradation, and corrective actions. This program is in accordance with ASME Section XI, 1995 edition through the 1996 addenda.

A.2.15 NON-EQ CABLE AGING MANAGEMENT PROGRAM

The FCS Non-EQ Cable Aging Management Program establishes a service life value for the Non-EQ cable in a similar fashion as the FCS EQ Program establishes a Qualified Life for the safety related equipment, components, and cable. Corrective actions for Non-EQ Cable, determined not to meet the operational (Service Life) requirements established for the full period of extended operation, will consider using: (1) state of the art analytical techniques to determine if the service life can be further extended; (2) industry accepted and regulatory approved cable inspection techniques that provide aging related data; and/or (3) state of the art, in-situ, non-destructive testing of cable performance, and/or laboratory testing of cable to extend life. Cable replacement will be considered should the aforementioned methodologies not succeed in extending the required service life.

A.2.16 ONE-TIME INSPECTION PROGRAM

The FCS One-Time Inspection Program is a new program that will implement a one-time inspection of internal surfaces of selected components to verify the effectiveness of mitigating programs such as the chemistry and diesel fuel oil programs. Inspections will be performed using suitable techniques at the most susceptible locations to verify that aging effects are not occurring or that the aging effect is progressing at such a slow rate it will not impact the intended function during the period of extended operation.

A.2.17 OVERHEAD LOAD HANDLING SYSTEMS INSPECTION PROGRAM

The Overhead Load Handling Systems Inspection Program implements FCS commitments made in response to NRC GL 81-07, *Control of Heavy Loads at Nuclear Power Plants,* and the maintenance monitoring requirements of 10 CFR 50.65. The program includes assessment of crane lift capabilities, periodic inspections of structural components, and functional tests to assure their integrity.

A.2.18 PERIODIC SURVEILLANCE AND PREVENTIVE MAINTENANCE PROGRAM

The Periodic Surveillance and Preventive Maintenance Program provides for periodic inspections and examinations of specific system and structural components using established NDE techniques. The inspection and examination techniques used and the periodicity of their performance provide reasonable assurance that age related degradation will not compromise the structure or component intended function(s) before the next scheduled inspection.

A.2.19 REACTOR VESSEL INTEGRITY PROGRAM

The Reactor Vessel Integrity Program monitors the extent of changes in material properties and loss of fracture toughness of irradiated reactor pressure vessel materials by periodic removal and testing of surveillance capsules located within the reactor vessel in accordance with RG 1.99, *Radiation Embrittlement of Reactor Vessel Materials,* Rev. 2. The surveillance capsule removal and evaluation is an NRC-approved program that meets the requirements of 10 CFR 50, Appendix H. The program includes revising the FCS surveillance capsule removal schedule in order to optimize the program through the end of the period of extended operation. In addition, the program verifies 10 CFR 50, Appendix G and 10 CFR 50.61 requirements.

A.2.20 REACTOR VESSEL INTERNALS INSPECTION PROGRAM

The Reactor Vessel (RV) Internals Inspection Program includes the following elements for cast austenitic stainless steel (CASS) and other reactor vessel internal components: (a) determination of the susceptibility of CASS components to thermal aging and neutron irradiation embrittlement, (b) identification of the most susceptible or limiting items, (c) development of appropriate inspection techniques to permit detection and characterizing of the feature (cracks) of interest and demonstrate the effectiveness of the proposed technique, and (d) implementation of required inspections prior to the period of extended operation.

A.2.21 SELECTIVE LEACHING PROGRAM

The FCS Selective Leaching Program implements inspection requirements for susceptible components for indication of selective leaching through dezincification or graphitization.

A.2.22 STEAM GENERATOR PROGRAM

The FCS Steam Generator Program consists of inspection scope, frequency, and acceptance criteria for various steam generator components, including the plugging and repair of flawed tubes in accordance with the plant Technical Specifications and the guidance of NEI 97-06, *Steam Generator Program Guidelines.*

A.2.23 STRUCTURES MONITORING PROGRAM

The Structures Monitoring Program provides for periodic visual inspection of designated FCS structures and component supports to ensure that aging degradation will be detected, evaluated, and repaired prior to any loss of intended functions. The inspection requirements are based on the following industry documents: NRC Bulletin 80-11, *Masonry Wall Design;* NRC IN 87-67, *Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11;* NUMARC 93-01, *Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (Line-In/Line-Out Version)*, Rev. 2; and NRC RG 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Rev. 2.

A.2.24 THERMAL AGING EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL PROGRAM

The Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program includes evaluation of the reactor coolant piping as bounded by the Leak-Before-Break (LBB) analysis, assessment of other CASS components for susceptibility to thermal embrittlement, and performance of volumetric inspection of piping or component-specific flaw tolerance evaluation for susceptible components.

A.3 EVALUATION OF TIME-LIMITED AGING ANALYSES

A.3.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT

There are four analyses affected by neutron embrittlement that have been identified as TLAAs:

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

A.3.1.1 PRESSURE/TEMPERATURE (P/T) CURVES

Appendix G to 10 CFR 50 requires that PT limits be established during all phases of reactor operation and that thermal stresses be limited by determining maximum heatup and cooldown rates. 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.

A.3.1.2 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 neutron embrittlement must be evaluated during the period of extended operation. Therefore, the LTOP analyses will be projected to the end of the period of extended operation.

A.3.1.3 PRESSURIZED THERMAL SHOCK (PTS)

10 CFR 50.61 addresses another issue related to embrittlement and thermal stress called Pressurized Thermal Shock (PTS). 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 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 proposed license renewal period. Therefore, the analyses have been projected to the end of the period of extended operation.

A.3.1.4 REACTOR VESSEL UPPER SHELF ENERGY

Upper shelf energy is a measure of fracture toughness at temperatures above RT_{PTS} when the vessel is exposed to neutron 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.

A.3.2 METAL FATIGUE

There are three distinct issues considered separately under the TLAA for Metal Fatigue:

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

Each of these issues is managed by the Fatigue Monitoring Program which is addressed in Section A.2.10 of this Appendix.

A.3.3 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

10 CFR 50.49 requires that certain safety related and non-safety related electrical equipment remains functional during and after identified Design Basis Events (DBEs). For the period of extended operation, Environmental Qualification (EQ) is a TLAA affecting all equipment in the scope of the EQ program, with a qualified life longer than the original license period but shorter than the combined original license period plus the period of extended operation, whether active or passive.

The FCS Electrical Equipment Qualification (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. 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.

A.3.4 CONCRETE CONTAINMENT TENDON PRE-STRESS

The containment wall and dome were pre-stressed by means of unbonded post-tensioned tendons. 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. Pre-stressing tendon integrity is monitored and confirmed by a regular program of tendon surveillance. 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 calculated pre-stress at end of plant life exceeds by a reasonable margin the intensity required to meet the design criteria.

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.

A.3.5 CONTAINMENT LINER PLATE AND PENETRATION SLEEVE FATIGUE

The containment liner and penetration sleeves were designed to be 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 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. 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. This condition has been evaluated and found adequate for the current term. FCS 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.

A.3.6 PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

A.3.6.1 REACTOR COOLANT PUMP FLYWHEEL FATIGUE

A.3.6.1.1 GENERAL ELECTRIC RCP FLYWHEELS

The resistance to rupture of the reactor coolant pump flywheels has been examined at 120 percent overspeed. The conclusion was that over 185,000 complete cycles from zero to 120 percent overspeed would be required to cause a crack 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.

A.3.6.1.2 ASEA BROWN BOVERI (ABB) MOTOR FLYWHEEL

During the 1996 refueling outage, reactor coolant pump RC-3B motor was replaced with a motor manufactured by ABB Industries. A crack growth analysis was performed 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.

A.3.6.2 LEAK BEFORE BREAK (LBB) ANALYSIS FOR RESOLUTION OF USI A-2

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 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.

A.3.6.3 HIGH ENERGY LINE BREAK (HELB)

The High Energy Line Break (HELB) analysis 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 the Containment Structure. 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 the Containment Structure to identify locations with cumulative usage factors greater than 0.1 as one of the criteria for selecting postulated break locations. The Class I portions encompass the piping from the Containment Structure penetrations to the first isolation valves outside the Containment Structure.

For the Class I MFW piping, projection of the CUFs for the period of extended operation does not require either any additional pipe break analysis to be performed or hardware to be installed on the Class I portions of MFW outside the Containment Structure. Also, for the Class I MS piping, 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 the Containment Structure.

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.

APPENDIX B – AGING MANAGEMENT ACTIVITIES

INTRODUCTION

The aging management activity descriptions are provided in this appendix for each activity credited for managing aging effects based upon the aging management review results provided in Sections 3.1 through 3.6.

The FCS Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Appendix A.2 of NUREG-1800. The Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls, and is applicable to the safety-related and non-safety-related structures, systems, and components that are subject to aging management review.

In many cases, existing activities were found adequate for managing aging effects during the period of extended operation. In some cases, aging management reviews revealed that existing activities should be enhanced to adequately manage aging. In a few cases, new activities were developed to provide reasonable assurance that aging effects are adequately managed.

Each aging management activity presented in this appendix is characterized as one of the following:

Existing Activity: A current activity that will continue to be implemented during the period of extended operation.

Enhanced Activity: A current activity that will be modified to manage aging during the period of extended operation.

New Activity: An activity that does not currently exist, which will manage aging during the period of extended operation.

The following aging management activities are described in the sections listed in this appendix. Site specific programs are indicated. All other programs correlate to some degree with programs in NUREG-1801.

Existing Aging Management Activities

B.1.1	Chemistry Program
B.1.2	Containment Inservice Inspection Program
B.1.3	Containment Leak Rate Program
B.1.4	Flow Accelerated Corrosion Program
B.1.5	Inservice Inspection Program
B.1.6	Reactor Vessel Integrity Program
B.1.7	Steam Generator Program

Enhanced Aging Management Activities

B.2.1	Bolting Integrity Program
B.2.2	Boric Acid Corrosion Prevention Program
B.2.3	Cooling Water Corrosion Program
B.2.4	Diesel Fuel Monitoring and Storage Program
B.2.5	Fatigue Monitoring Program
B.2.6	Fire Protection Program
B.2.7	Overhead Load Handling Systems Inspection Program
B.2.8	Periodic Surveillance and Preventive Maintenance Program **(site specific program)**
B.2.9	Reactor Vessel Internals Inspection Program
B.2.10	Structures Monitoring Program
B.2.11	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel

New Aging Management Activities

B.3.1	Alloy 600 Program
B.3.2	Buried Surfaces External Corrosion Program
B.3.3	General Corrosion of External Surfaces Program **(site specific program)**
B.3.4	Non-EQ Cable Aging Management Program **(site specific program)**
B.3.5	One-Time Inspection Program
B.3.6	Selective Leaching Program

Correlation between NUREG-1801 (Generic Aging Lessons Learned (GALL)) programs and FCS programs are shown below. For the FCS Programs, appropriate references to sections of this application are provided.

NUREG-1801 ID Number	NUREG-1801 Program	FCS Program
XI.M1	ASME Section XI Inservice Inspection, Subsection IWB, IWC, IWD	Inservice Inspection Program (B.1.5)
XI.M2	Water Chemistry	Chemistry Program (B.1.1)
XI.M3	Reactor Head Closure Studs	Bolting Integrity Program (B.2.1)
XI.M4	BWR Vessel ID Attachment Welds	Not applicable, FCS is a PWR.
XI.M5	BWR Feedwater Nozzle	Not applicable, FCS is a PWR.
XI.M6	BWR Control Rod Drive Return Line Nozzle	Not applicable, FCS is a PWR.
XI.M7	BWR Stress Corrosion Cracking	Not applicable, FCS is a PWR.
XI.M8	BWR Penetrations	Not applicable, FCS is a PWR.
XI.M9	BWR Vessel Internals	Not applicable, FCS is a PWR.
XI.M10	Boric Acid Corrosion	Boric Acid Corrosion Prevention Program (B.2.2)
XI.M11	Nickel-Alloy Nozzles and Penetrations	Alloy 600 Program (B.3.1)
XI.M12	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) (B.2.11)
XI.M13	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)	Reactor Vessel Internals Inspection Program (B.2.9)
XI.M14	Loose Part Monitoring	Not credited for aging management. Reactor vessel internals inspections were determined to be adequate to manage identified aging effects.

NUREG-1801 ID Number	NUREG-1801 Program	FCS Program
XI.M15	Neutron Noise Monitoring	Reactor vessel internals vibration monitoring is a current FCS licensing commitment. The implementing task is incorporated in the Reactor Vessel Internals Inspection Program (B.2.9).
XI.M16	PWR Vessel Internals	Reactor Vessel Internals Inspection Program (B.2.9)
XI.M17	Flow-Accelerated Corrosion	Flow-Accelerated Corrosion Program (B.1.4)
XI.M18	Bolting Integrity	Bolting Integrity Program (B.2.1)
XI.M19	Steam Generator Tube Integrity	Steam Generator Program (B.1.7)
XI.M20	Open-Cycle Cooling Water System	Cooling Water Corrosion Program (B.2.3)
XI.M21	Closed-Cycle Cooling Water System	Cooling Water Corrosion Program (B.2.3)
XI.M22	Boraflex Monitoring	Not applicable, FCS does not have Boraflex.
XI.M23	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Overhead Loading Handling Systems Inspection Program (B.2.7)
XI.M24	Compressed Air Monitoring	Not credited for aging management. No aging effects requiring management were identified for the Compressed Air System.
XI.M25	BWR Reactor Water Cleanup System	Not applicable, FCS is a PWR.
XI.M26	Fire Protection	Fire Protection Program (B.2.6)
XI.M27	Fire Water System	Fire Protection Program (B.2.6)

NUREG-1801 ID Number	NUREG-1801 Program	FCS Program
XI.M28	Buried Piping and Tanks Surveillance	Not credited for aging management. FCS cathodic protection was not credited for managing aging effects. The FCS aging management program was based on the requirements of NUREG-1801 XI.M34.
XI.M29	Aboveground Carbon Steel Tanks	Not credited for aging management. Steel tanks were not treated as separate components from their respective systems. Applicable aging management activities have been incorporated into programs credited for similar component, material, and environments in the system.
XI.M30	Fuel Oil Chemistry	Diesel Fuel Monitoring and Storage Program (B.2.4)
XI.M31	Reactor Vessel Surveillance	Reactor Vessel Integrity Program (B.1.6)
XI.M32	One-Time Inspection	One-Time Inspection Program (B.3.5)
XI.M33	Selective Leaching of Materials	Selective Leaching Program (B.3.6)
XI.M34	Buried Piping and Tanks Inspection	Buried Surfaces External Corrosion Program (B.3.2)
XI.E1	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Plant Specific Program - Non-EQ Cable Aging Management Program (B.3.4)
XI.E2	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Plant Specific Program - Non-EQ Cable Aging Management Program (B.3.4)

NUREG-1801 ID Number	NUREG-1801 Program	FCS Program
XI.E3	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Plant Specific Program - Non-EQ Cable Aging Management Program (B.3.4)
XI.S1	ASME Section XI, Subsection IWE	Containment Inservice Inspection Program (B.1.2)
XI.S2	ASME Section XI, Subsection IWL	Containment Inservice Inspection Program (B.1.2)
XI.S3	ASME Section XI, Subsection IWF	Inservice Inspection Program (B.1.5)
XI.S4	10 CFR 50, Appendix J	Containment Leak Rate Program (B.1.3)
XI.S5	Masonry Wall Program	Structures Monitoring Program (B.2.10)
XI.S6	Structures Monitoring Program	Structures Monitoring Program (B.2.10)
XI.S7	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants	Structures Monitoring Program (B.2.10)
XI.S8	Protective Coating Monitoring and Maintenance Program	Not credited for aging management.
Chapter 10		
X.M1	Metal Fatigue of Reactor Coolant Pressure Boundary	Fatigue Monitoring Program (B.2.5)
X.E1	Environmental Qualification (EQ) of Electric Components	See Section 4.4 of this application.
X.S1	Concrete Containment Tendon Prestress	Containment Inservice Inspection Program (B.1.2)

B.1 EXISTING AGING MANAGEMENT ACTIVITIES

B.1.1 CHEMISTRY PROGRAM

The FCS Chemistry Program will be consistent with XI.M2, *Water Chemistry,* and the chemistry-related portions of XI.M21, *Closed-Cycle Cooling Water System,* as identified in NUREG-1801 prior to the period of extended operation.

Operating Experience:

Over the FCS operating history, chemistry related situations have occasionally occurred. These include a steam generator tube leak, condenser tube leaks, and some resin intrusion into the primary water storage tank. These situations were properly corrected and long-term corrective actions were implemented to prevent recurrence. Chemistry management of aging effects has evolved over the years based on FCS and industry experience. OPPD has adopted industry practices throughout the years, and continues to do so in order to enhance chemistry control. The low percentage of plugged steam generator tubes based on the number of years the generators have been in service is indicative of the effective chemistry control. The overall experience illustrates that the Chemistry Program is effective in managing aging.

Conclusion:

The Chemistry Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.1.2 CONTAINMENT INSERVICE INSPECTION PROGRAM

The FCS Containment Inservice Inspection Program is consistent with X.S1, *Concrete Containment Tendon Prestress*, XI.S1, *ASME Section XI, Subsection IWE*, and XI.S2, *ASME Section XI, Subsection IWL*, as identified in NUREG-1801.

The 10 Year Containment (IWE & IWL) Inservice Inspection Program Plan for FCS, incorporating Subsection IWE and Subsection IWL examination requirements, has been developed and implemented.

Operating Experience:

Inspections of the Containment Liner have been conducted in accordance with the Containment Leak Rate Testing Program and the Maintenance Rule Implementation Program. Inspections of the tendons and tendon anchorages have been conducted in accordance with Technical Specifications, the USAR, and plant procedures. The ASME Section XI, Subsection IWL Inservice Inspection Program incorporates all of the inspection criteria and guidelines of the previous tendon inspection program and is implemented using existing plant procedures. No significant age related degradation has been identified in the inspections performed.

Conclusion:

The Containment Inservice Inspection Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.1.3 CONTAINMENT LEAK RATE PROGRAM

The FCS Containment Leak Rate Program is consistent with XI.S4, *10 CFR Part 50, Appendix J,* and applicable sections of XI.S1, *ASME XI, Subsection IWE* related to Appendix J testing as identified in NUREG-1801.

Operating Experience:

Containment leak-tight verification and visual examination of the steel components that are part of the leak-tight barrier have been conducted at FCS since initial unit startup. Prior to the development of the ASME Section XI, Subsection IWE Inservice Inspection Program, examinations were performed in accordance with 10 CFR 50, Appendix J. No significant age related degradation has been identified in the inspections performed.

Conclusion:

The Containment Leak Rate Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.1.4 FLOW ACCELERATED CORROSION PROGRAM

The FCS Flow Accelerated Corrosion (FAC) Program is consistent with XI.M17, *Flow-Accelerated Corrosion*, as identified in NUREG-1801.

Operating Experience:

FAC inspections have been performed periodically on both in-scope and out-of-scope piping. These inspections have gone on for many years and the FAC program has been adjusted based on inspection and other results. On occasion, pipe wall has been found below established screening criteria and visual observations have identified through-wall erosion. These deficiencies were documented in accordance with the FCS corrective action program and resulted in repair or replacement of the affected areas. A rupture occurred on a non-CQE extraction steam line in 1997 which resulted in significant upgrades to the FAC program. Internal audits and NRC inspection of the program since 1997 have found the program to be maintained in accordance with NSAC-202L-R2, *Recommendations for an Effective Flow-Accelerated Corrosion Program.*

Conclusion:

The Flow Accelerated Corrosion Program provides reasonable assurance that flow accelerated corrosion will be managed such that components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.1.5 INSERVICE INSPECTION PROGRAM

The Inservice Inspection Program is consistent with XI.M1, *ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,* and XI.S3, *ASME Section XI, Subsections IWF,* as identified in NUREG-1801.

Operating Experience:

Review of the plant specific operating experience indicates that the FCS Inservice Inspection Program has been effective in managing the aging effects of components. No significant age related deterioration has been identified in the inspections performed.

Conclusion:

The FCS Inservice Inspection Program provides reasonable assurance that the aging effects will be managed such that the ASME Class 1, 2, and 3 components and their integral supports subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.1.6 REACTOR VESSEL INTEGRITY PROGRAM

The FCS Reactor Vessel Integrity Program will be consistent with XI.31, *Reactor Vessel Surveillance*, as identified in NUREG-1801 prior to the period of extended operation.

Operating Experience:

At FCS, three surveillance capsules have been removed and the materials tested. The FCS operating experience is being supplemented by surveillance capsule test results from other operating reactors whose surveillance capsules include materials that exactly match the materials of the various FCS reactor vessel beltline welds, including the limiting or critical weld.

The results of testing of the early surveillance capsules, use of the chemistry factors for the limiting weld, and the early results of the updated fluence analysis indicated that the FCS reactor vessel could exceed the PTS screening criteria of 10 CFR 50.61 before the end of the current 40-year license period in 2013. As a result, FCS implemented core design limitations aimed at restricting the fluence of the reactor vessel beltline region. Analysis has been completed which demonstrates that FCS will be able to operate to the end of the extended period of operation without exceeding the PTS screening criteria. These analysis results have been reviewed and NRC approved by Amendment 199 to the FCS Operating License.

Conclusion:

The Reactor Vessel Integrity Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.1.7 STEAM GENERATOR PROGRAM

The FCS Steam Generator Program will be consistent with XI.M19, *Steam Generator Tube Integrity,* as identified in NUREG-1801 prior to the period of extended operation with the following clarification:

- In addition to the requirements of XI.M19, the FCS Steam Generator Program also includes aging management activities to address plant specific AMP requirements identified in Table 3.1.1.

Operating Experience:

Steam generator management of aging effects has evolved and improved over the years based on industry experience. FCS has adopted industry practices throughout the years, and continues to do so. Past NRC inspections on this program cited sample plans and inspection evaluation as a strength. Only one noteworthy situation has occurred. In 1984, a misplug and misdiagnosed tube problem led to a tube rupture. This situation was corrected and long-term corrective actions were implemented to prevent recurrence. Current FCS practices are state-of-the-art. The overall experience illustrates that the Steam Generator Program is effective in managing aging.

Conclusion:

The FCS Steam Generator Program provides reasonable assurance that the aging effects will be managed such that components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2 ENHANCED AGING MANAGEMENT ACTIVITIES

B.2.1 BOLTING INTEGRITY PROGRAM

The Bolting Integrity Program will be consistent with XI.M3, *Reactor Head Closure Studs* and XI.M18, *Bolting Integrity*, as identified in NUREG-1801 prior to the period of extended operation.

Operating Experience:

Inspections of bolted components have been conducted under the FCS Inservice Inspection Program (based on ASME Section XI code requirements), the FCS Boric Acid Corrosion (BAC) Prevention Program, and the Structures Monitoring Program. Visual inspections conducted under the Boric Acid Corrosion Prevention Program include inspection of bolted components in borated systems. Any indication of BAC residue or damage is reported and evaluated to determine if a component can remain in service per established procedures. Documentation of operating experience is included in the BAC Inspection Program. On occasion, visual observations have identified BAC damage. These deficiencies were documented in accordance with the FCS corrective action program and resulted in repair or replacement if required.

Review of the plant specific operating experience indicates that the inspections have been effective in managing the aging effects of bolted components.

Conclusion:

The Bolting Integrity Program provides reasonable assurance that the aging effects will be managed such that the bolted components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.2 BORIC ACID CORROSION PREVENTION PROGRAM

The FCS Boric Acid Corrosion (BAC) Prevention Program will be consistent with XI.M10, *Boric Acid Corrosion*, as identified in NUREG-1801 prior to the period of extended operation.

Operating Experience:

FCS experienced severe boric acid corrosion on reactor coolant pump studs as documented in NRC Generic Letter (GL) 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants*. Significant program improvements were implemented in response to that GL. A review of the post GL 88-05 operating history indicates that the BAC Prevention Program at FCS routinely identifies and corrects borated water leakage and BAC in the Reactor Coolant System and other borated water systems, including any adjacent structures or components that could be adversely affected.

Conclusion:

The FCS Boric Acid Corrosion Prevention Program provides reasonable assurance that the aging effects will be managed such that the susceptible components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.3 COOLING WATER CORROSION PROGRAM

The FCS Cooling Water Corrosion Program will be consistent with XI.M20, *Open-Cycle Cooling Water System,* and XI.21, *Closed-Cycle Cooling Water System,* as identified in NUREG-1801 prior to the period of extended operation, with the following clarifications:

- XI.M20 - Program Description, 3. Parameters Monitored/Inspected, 4. Detection of Aging Effects, 5. Monitoring and Trending, and 6. Acceptance Criteria

  External coatings are addressed by the FCS General Corrosion of External Surfaces Program.

- XI.M21 - Program Description, 2. Preventative Actions, 5. Monitoring and Trending, 6. Acceptance Criteria, and 7. Corrective Action

  The Chemistry-related portions of the program are addressed in the FCS Chemistry Program.

The FCS Cooling Water Corrosion Program will also include the following exceptions to NUREG-1801:

- XI.M21 - 3. Parameters Monitored/Inspected, 4. Detection of Aging Effects, and 5. Monitoring and Trending

  The license renewal commitment for these programs relates only to the maintenance of the pressure boundary and not the maintenance of fluid flow. Fluid flow is considered an active function. Performance testing and other active system function testing is not performed on an 18 month or 5 year frequency in accordance with EPRI TR-107396, *Closed Cooling Water Chemistry Guideline,* because this EPRI document does not address this criteria or specify that testing frequency. Non-destructive testing and heat transfer performance to identify pressure boundary integrity are performed per EPRI TR-107396.

Operating Experience:

Review of FCS operating experience has identified some component part replacements (and repairs) due to corrosion and cracking in the Component Cooling Water and Raw Water Systems. Appropriate long term corrective actions were implemented based on these experiences. These included material changes, additional preventive maintenance, and increased sample evaluation.

Conclusion:

The FCS Cooling Water Corrosion Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.4 DIESEL FUEL MONITORING AND STORAGE PROGRAM

The FCS Diesel Fuel Monitoring and Storage Program will be consistent with XI.M30, *Fuel Oil Chemistry,* as identified in NUREG-1801 prior to the period of extended operation, with the following clarifications:

- XI.M30-3. Parameters Monitored/Inspected

  Although OPPD does perform particulate analysis of fuel oil, OPPD does not credit this analysis for any aging management. Particulate analysis is performed on diesel fuel for fuel burn quality concerns (i.e., clogging of filters), and does not have any impact on pressure boundary integrity.

- XI.M30-4. Detection of Aging Effects

  Ultrasonic testing is not performed on the fire protection diesel fuel oil tank due to the inaccessibility of the tank. Leak detection is employed to monitor the condition of the tank and is adequate to maintain the system design requirements.

Operating Experience:

FCS operating experience indicates there have been no instances of fuel oil system component failures due to aging effects.

Conclusion:

The FCS Diesel Fuel Monitoring and Storage Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.5 FATIGUE MONITORING PROGRAM

The FCS Fatigue Monitoring Program will be consistent with X.M1, *Metal Fatigue of Reactor Coolant Pressure Boundary*, as identified in NUREG-1801 prior to the period of extended operation with the following clarification:

- Program Description

  The FCS Fatigue Monitoring Program will also include Class 2 and 3 components not included in the NUREG-1801 program which are subject to fatigue as an aging effect requiring management.

The FCS Fatigue Monitoring Program will also include the following exception to NUREG-1801:

- Program Description, 2. Preventative Actions, and 6. Acceptance Criteria

  A site specific analysis will be performed to address environmental fatigue concerns identified in NUREG/CR-6260. Corrective actions or program enhancements will be implemented if necessary based on the results of evaluation.

Operating Experience:

There have been no thermal fatigue related failures at FCS; however, there have been two occurrences (with associated corrective action documents) at FCS relative to thermal fatigue that have resulted in enhancements to the FCS Fatigue Monitoring Program.

The first of these documents summarizes concerns about the operation of the Chemical and Volume Control System (CVCS) and whether specific components within or related to the system were having their thermal cycles monitored and tracked consistently. This resulted in the performance of an Engineering Assessment to document a review of Design Basis Documents, the USAR, Technical Specifications, and other documents to determine cycle counting requirements. This review resulted in revision to some of these documents. An operating history review was performed to determine the number of cycles that the components of concern actually experienced. Part of this review was to ensure that the thermal cycles counted were, in fact, a result of design basis conditions that merited inclusion in the cycle counting.

The other document was written after a rash of industry small bore piping failures (generally detected as small cracks or leaks as opposed to major pressure boundary ruptures) in primary coolant systems. Two of the resulting action items created a small bore piping fatigue program and a systematic program for thermal fatigue. These actions have been ongoing and are being integrated with license renewal specific thermal fatigue action items to form the basis for this new program.

The sample frequency of the Primary Sampling System is such that its limit of 7000 cycles will be exceeded before the end of the period of extended operation. Prior to entering the period of extended operation, a stress analysis will be performed based on the sampling evolution parameters to determine whether or not the applicable sampling evolution piping will have to be replaced before the end of the period of extended operation.

Pressurizer surge line thermal stratification is an issue identified by NRC Bulletin 88-11. 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 0.937 usage factor for the surge line after the 40-year design life that would obviously be exceeded during the period of extended operation. This value is based on the use of the most limiting configuration of the surge line that exists in the CE-designed plants 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 FCS Fatigue Monitoring Program. As part of this program, realistic usage factors will be compiled for the critical areas based on actual plant operating data to include the effects of thermal stratification. These are expected to be lower than those predicted by the generic evaluation. This reevaluation will take place prior to the period of extended operation. Based on the results of this plant specific analysis, realistic fatigue usage for the surge line will be tracked, and actions will be taken to reevaluate, repair, or replace the surge line as necessary.

Conclusion:

The Fatigue Monitoring Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

B.2.6 FIRE PROTECTION PROGRAM

The Fire Protection Program will be consistent with XI.M26, *Fire Protection,* and XI.M27, *Fire Water System,* as identified in NUREG-1801 prior to the period of extended operation, with the following clarification:

- XI.M27-2. Preventative Action

  NUREG-1801 specifies in Section XI.M27, *Fire Water System,* that "portions of the fire protection sprinkler system, which are not routinely subjected to flow, are to be subjected to full flow tests at the maximum design flow and pressure." The FCS USAR, Table 9.11-3, directs flow testing to be performed using a clean water source. The demineralized water booster pumps or Blair City water are used for flow testing at pressures slightly lower than the normal system operating pressure. This is not consistent with NUREG-1801; however, both the pressure and resulting flow are sufficient to effectively entrain and adequately flow test/flush the sprinkler system piping. This ensures that aging effects are managed such that the intended function is maintained.

Operating Experience:

Routine visual inspections of fire barriers have proven effective in identifying material degradation and damage. A recent decline in the number of identified fire barrier penetration discrepancies is attributed to a recent fire barrier and penetration upgrade effort. Historical operating experience shows fire barrier walls, ceilings, doors and floors are adequately managed through inspections.

Through-wall leakage of seamed fire protection system piping has been identified at FCS. Routine walkdowns and piping inspections (internal inspections performed when the system is breached for repair) have been implemented to accurately detect and identify early stages of pressure boundary deterioration and leakage. Historical operating experience and discussions with fire protection personnel show this program effectively manages and corrects pressure boundary failures.

Operating history for yard fire hydrants, fire dampers, sprinklers and nozzles shows adequate management of the aging effects identified by chapters XI.M26 and XI.M27 of NUREG-1801. Halon system piping and tanks have shown few historical discrepancies and are adequately managed by the FCS program. No historical experience was identified concerning the diesel fire pump fuel oil supply line.

Conclusion:

The Fire Protection Program provides reasonable assurance that the aging effects will be managed such that the structures and components subject to aging management review will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

B.2.7 OVERHEAD LOAD HANDLING SYSTEMS INSPECTION PROGRAM

The FCS Overhead Load Handling Systems Inspection Program will be consistent with XI.M23, *Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems*, as identified in NUREG-1801 prior to the period of extended operation.

Operating Experience:

The subject load handling equipment is periodically inspected for degradation. No aging effects which impact the intended functions of the structures or components were identified in the inspections performed.

Conclusion:

The FCS Overhead Load Handling Systems Inspection Program provides reasonable assurance that aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.8 PERIODIC SURVEILLANCE AND PREVENTIVE MAINTENANCE (PM) PROGRAM

The stated purpose of the PM program is to prevent or minimize equipment breakdown and to maintain equipment in a condition that will enable it to perform its normal and emergency functions. The program and the site administrative control processes provide for a systematic approach in establishing the method, frequency, acceptance criteria, and documentation of results.

The FCS Periodic Surveillance and Preventive Maintenance Program consists of periodic inspections and tests that are relied on to manage aging for system and structural components and that are not evaluated as part of the other aging management programs addressed in this appendix. The preventive maintenance and surveillance testing activities are implemented through periodic work orders that provide for assurance of functionality of the components by confirmation of integrity of applicable parameters.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Program:

The FCS Periodic Surveillance and Preventive Maintenance Program provides for periodic inspection and testing of components in the following systems and structures.

- Auxiliary Building
- Auxiliary Building HVAC
- Auxiliary Feedwater
- Chemical and Volume Control
- Component Cooling
- Containment
- Containment HVAC
- Control Room HVAC and Toxic Gas Monitoring
- Diesel Generator Lube Oil
- Duct Banks
- Emergency Diesel Generators
- Fire Protection
- Fuel Handling Equipment/Heavy Load Cranes
- Intake Structure
- Liquid Waste Disposal
- Containment Penetration, and System Interface Components for Non-CQE Systems
- Reactor Coolant
- Safety Injection and Containment Spray
- Ventilating Air

(2) Preventive Actions:

The Periodic Surveillance and Preventive Maintenance Program includes periodic refurbishment or replacement of components, which could be considered to be preventive or mitigative actions. The inspections and testing to identify component aging degradation effects do not constitute preventive actions in the context of this element.

(3) Parameters Monitored or Inspected:

Inspection and testing activities monitor parameters including surface condition, loss of material, presence of corrosion products, signs of cracking and presence of water in oil samples.

(4) Detection of Aging Effects:

Preventive maintenance and surveillance testing activities provide for periodic component inspections and testing to detect the following aging effects and mechanisms:

- Change in Material Properties
- Cracking
- Fouling
- Loss of Material
- Loss of Material - Crevice Corrosion
- Loss of Material - Fretting
- Loss of Material - General Corrosion
- Loss of Material - Pitting Corrosion
- Loss of Material - Pitting/ Crevice/Gen. Corrosion
- Loss of Material - Wear
- Separation

The extent and schedule of the inspections and testing assures detection of component degradation prior to the loss of their intended functions. Established techniques such as visual inspections and dye penetrant testing are used.

(5) Monitoring and Trending:

Preventive maintenance and surveillance testing activities provide for monitoring and trending of aging degradation. Inspection intervals are established such that they provide for timely detection of component degradation. Inspection intervals are dependent on the component material and environment and take into consideration industry and plant-specific operating experience and manufacturers' recommendations.

The program includes provisions for monitoring and trending with the stated intent of identifying potential failures or degradation and making adjustments to ensure components

remain capable of performing their functions. PM review and update guidelines are provided that include adjustment of PM task and frequency based on the as-found results of previous performance of the PM. In particular, responsible system engineers are required to periodically review the results of preventive maintenance and recommend changes based on these reviews. The program includes guidance to assist the system engineers in achieving efficient and effective trending.

(6) Acceptance Criteria:

Periodic Surveillance and Preventive Maintenance Program acceptance criteria are defined in the specific inspection and testing procedures. They confirm component integrity by verifying the absence of the aging effect or by comparing applicable parameters to limits based on the applicable intended function(s) as established by the plant design basis.

(7) Corrective Actions:

Identified deviations are evaluated within the FCS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation. The FCS corrective action process is in accordance with 10 CFR 50 Appendix B.

(8) Confirmation Process:

The FCS corrective action process is in accordance with 10 CFR 50 Appendix B and includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls:

All credited aging management activities are subject to the FCS administrative controls process, which is in accordance with 10 CFR 50 Appendix B and requires formal reviews and approvals.

(10) Operating Experience:

Periodic surveillance and preventive maintenance activities have been in place at FCS since the plant began operation. These activities have a demonstrated history of detecting damaged and degraded components and causing their repair or replacement in accordance with the site corrective action process. With few exceptions, age-related degradation adverse to component intended functions was discovered and corrective actions were taken prior to loss of intended function.

Conclusion:

The Periodic Surveillance and Preventive Maintenance Program assures that various aging effects are managed for a wide range of components at FCS. Based on the program structure and administrative processes and FCS operating experience, there is reasonable assurance that the credited inspection and testing activities of the Periodic Surveillance and Preventive Maintenance Program will continue to adequately manage the identified aging effects of the applicable components so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

B.2.9 REACTOR VESSEL INTERNALS INSPECTION PROGRAM

The FCS Reactor Vessel Internals Inspection Program will be consistent with XI.M13, *Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)*, and XI.M16, *PWR Vessel Internals*, as identified in NUREG-1801 prior to the period of extended operation with the following exceptions:

- XI.M16-4. Detection of Aging Effects

  No augmented inspection of bolting is scheduled. The tensile stresses on the FCS reactor vessel internals bolting are lower than the industry levels where cracking was observed as an aging effect. Refer to discussion in Operating Experience below.

- XI.M16-Program Description, 1. Scope of Program and 2. Preventative Actions

  The Chemistry-related portions of the program are addressed in the FCS Chemistry Program.

Operating Experience:

No cracking has been discovered in US pressurized water reactor (PWR) reactor vessel (RV) internals fabricated with austenitic stainless steel except for various bolting applications for Babcock & Wilcox and Westinghouse-designed NSSSs and thermal shield components at St. Lucie Unit 1 and Millstone Unit 2, which are CE designed Nuclear Steam Supply System (NSSS). The cracking at St. Lucie and Millstone was caused by flow-induced high cycle fatigue; the thermal shields at these plants were removed. Cracking of core barrel, baffle, and former bolts at Electricite de France (EdF), Westinghouse and Babcock & Wilcox-designed RV internals has been discovered. The cause of cracking of core barrel bolts at Babcock & Wilcox designed plants was stress corrosion cracking (SCC) and the cracking of baffle bolts at Westinghouse and EdF plants is believed to be irradiation assisted stress corrosion cracking (IASCC).

Reactor vessel internals inspections are performed under the FCS Inservice Inspection Program. No cracking caused by high cycle fatigue was discovered in the FCS thermal shield and therefore the FCS thermal shield was not removed, as is the case for St. Lucie Unit 1 and Millstone Unit 2. In 1984 a commitment was made to the NRC to perform an inspection of the thermal shield during the 1987 refueling outage. However, in 1986 an inspection deferral program was implemented that allowed a thermal shield monitoring program to replace the inspection commitment. This monitoring program generated data from 1988 through 1990 that indicated the early stages of loosening of the thermal sleeve positioning pins. During the 1992 refueling outage, visual inspection of the support lugs and the positioning pins was performed. No noticeable cracks, weld cracks, missing parts, misalignment, gaps, looseness, or wear were found. Eleven pins (7 lower pins and 4 upper pins) were removed and replaced to reinstate the specified amount of initial relative displacement between the thermal shield and the core support barrel. The amount of initial relative displacement was based on maintaining specified preload over twenty years in the

pins while accounting for radiation-induced relaxation and wear. This action reduced vibrations to below specified levels. No unacceptable vibration has been detected since 1992 and FCS continues to monitor thermal shield vibrations using the Internals Vibration Monitoring program. Any unacceptable vibration will be corrected when appropriate.

To date, no cracking has been discovered in bolting for Combustion Engineering (CE)-designed RV internals bolting. The Combustion Engineering Owners Group (CEOG) provided an assessment of the cracking of the baffle former bolts reported in foreign PWRs, including the potential impact of the cracking on domestic CE plants. The results are in CEOG Report CE NPSD-1098 for CEOG Task 1011, *Evaluation of the Applicability of Baffle Bolt Cracking to Ft. Calhoun and Palisades Internals Bolts,* Final Report, Revision 00, April 1998. The most likely mechanism for the cracking of cold-worked 316 stainless steel baffle former bolts in foreign plants is IASCC. There are only two CE-designed plants (FCS and Palisades) that use bolts to attach the core shroud panels (i.e., the baffle plates) to the former plates. The report indicates that these bolts in FCS are less susceptible to IASCC because: (1) the material used in these bolts is annealed 316 stainless steel, which is not cold worked; (2) the bolt stress from preload, as a percentage of yield strength, is much less than the EdF plants; (3) the differential pressure across the core shroud panels does not result in tensile loads on the panel (i.e., the baffle bolts) during normal operation; and (4) the core shroud panel design allows for some flexing of the former plate relative to the core barrel, thus reducing the load on the panel bolts. Since CE NPSD-1098 was issued, cracking has been discovered in Point Beach baffle bolts. However, as with the EdF experience, cracked bolts were highly stressed during preload, tensile stresses were applied during operation because of the Westinghouse design, and the bolts were fabricated with cold worked 316 stainless steel. Therefore, the findings of CE NPSD-1098 still apply.

Stress corrosion cracking was identified in B&W lower thermal shield and lower core barrel bolts that were fabricated with A-286. Most of the failed bolts were highly stressed to at or over the yield strength. Cracked bolts were replaced with bolts of improved design fabricated with Inconel X-750. No cracking of these bolts has recurred. Although there have been no failures of CEA Shroud Bolts in CE-designed RV internals, there is a concern that SCC may occur since these bolts are fabricated with Alloy A-286. CE provided an evaluation of the stress level for these bolts in 1984. According to CEN-282 *Investigation and Evaluation of A286 Bolt Applications in C-E's NSSS,* September 1984, operating stress levels are just below 32 Ksi. The stress concentration factor for the CEA Shroud Bolts is 2.06, leading to a local stress of approximately 66 Ksi. Yield strength for A-286 is about 115 Ksi, so the stress is approximately 60 percent of yield. Most of the failed B&W bolts had working stresses of approximately 65 Ksi and a local stress of 134 Ksi which is above the yield strength of the material. There were no failed B&W bolts with working stresses of 35 Ksi. The conclusion of the report indicates a low probability for cracking of the CEA Shroud Bolts.

Conclusion:

The Reactor Vessel Internals Inspection Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

B.2.10 STRUCTURES MONITORING PROGRAM

The Structures Monitoring Program will be consistent with XI.S5, *Masonry Wall Program*, XI.S6, *Structures Monitoring Program*, and XI.S7, RG 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants*, as identified in NUREG-1801 prior to the period of extended operation with the following clarification:

- FCS does not have lubrite supports as identified in NUREG-1801, Chapter III, item A4.2-b. FCS does have lubrite on some steam generator supports which are inspected under the FCS Inservice Inspection Program rather than the Structures Monitoring Program.

- XI.S7. Program Description

  FCS is not committed to RG 1.127. Applicable attributes from RG 1.127 have been incorporated into the Structures Monitoring Program as specified in the program description.

Operating Experience:

Inspections have been performed in the Auxiliary Building, Containment, Intake Structure, and Turbine Building in 1996/1997 and 1999/2000. No significant deterioration was identified. Some examples of corrosion of support anchors have been observed and documented under the FCS corrective action program.

Conclusion:

The Structures Monitoring Program provides reasonable assurance that the identified aging effects will be managed such that the structures and components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.11 THERMAL AGING EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL

The FCS Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program will be consistent with XI.M12, *Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)*, as identified in NUREG-1801 prior to the period of extended operation.

Operating Experience:

No age related degradation associated with thermal embrittlement of CASS was identified in the FCS operating experience.

Conclusion:

The FCS Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Program provides reasonable assurance that the aging effects will be managed such that components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3 NEW AGING MANAGEMENT ACTIVITIES

B.3.1 ALLOY 600 PROGRAM

The FCS Alloy 600 Program will be consistent with the requirements of XI.M11, *Nickel-Alloy Nozzles and Penetrations*, as identified in NUREG-1801 prior to the period of extended operation with the following exceptions:

- XI.M11-4. Detection of Aging Effects

  The FCS Alloy 600 Program will not rely on an enhanced leakage detection system for detection of leaks caused by primary water stress corrosion cracking (PWSCC) as suggested by XI.M11 in NUREG-1801. Bounding evaluations exist that demonstrate that PWSCC cracks can be detected via boric acid leakage prior to the structural integrity of the pressure boundary being compromised and prior to unacceptable material loss of carbon steel vessels due to boric acid corrosion.

- XI.M11-Program Description, 1. Scope of Program, and 2. Preventative Action

  The Chemistry-related portions of the program are addressed in the FCS Chemistry Program.

The program includes participation in industry programs to determine an appropriate aging management program for PWSCC of Inconel 182 welds.

Operating Experience:

OPPD has proactively responded to industry experience with PWSCC of Alloy 600. In response to NRC Information Notice 90-10, *Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600*, OPPD initiated an investigation of the applications of Alloy 600, Alloy 82 and Alloy 182 in the FCS reactor coolant system. OPPD participated in the industry integrated inspection program used to respond to Generic Letter 97-01, *Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations*, and is currently following industry developments related to circumferential cracking in control rod drive mechanisms (CRDMS) as described in NRC Bulletin 2001-01. Experience with weld PWSCC at V.C. Summer and a pressurizer instrument nozzle leak at FCS (both in October, 2000) prompted OPPD to review fabrication records of Alloy 82 and Alloy 182 welds and Alloy 600 components for evidence of fabrication rework, since this was identified as a causal factor in both incidents. In response to the V.C. Summer incident, FCS engineering staff briefed plant operators and inspection personnel to sensitize them to the potential for Alloy 82/182 butt weld cracks.

Conclusion:

The Alloy 600 Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.2 BURIED SURFACES EXTERNAL CORROSION PROGRAM

The Buried Surfaces External Corrosion Program will be consistent with XI.M34, *Buried Piping and Tanks Inspection*, as identified in NUREG-1801 prior to the period of extended operation.

Operating Experience:

Tank wall thickness measurements, conducted as part of the Diesel Fuel Oil Monitoring and Storage Program for the emergency diesel generator and auxiliary boiler fuel oil storage tanks, have determined that there is no indication of external corrosion for either vessel.

As opportunities have arisen, visual inspections have been performed on excavated piping. A recent excavation for the repair of buried valves in the Fire Protection System also exposed sections of Raw Water System piping. The applied coatings and wrappings of the excavated Fire Protection and Raw Water System piping and components were found to be in good condition with no indication of loss of material from the metal beneath.

Conclusion:

The Buried Surfaces External Corrosion Program provides reasonable assurance that the identified aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.3 GENERAL CORROSION OF EXTERNAL SURFACES PROGRAM

The General Corrosion of External Surfaces Program at FCS is credited for aging management of the effects of loss of material and cracking for applicable components, including piping, valves, supports, tanks, and bolting, which are made of cadmium plated steel, carbon steel, cast iron, copper alloy, galvanized steel, low alloy steel, and neoprene.

(1) Scope of Program

The General Corrosion of External Surfaces Program consists of several FCS activities that manage the aging effects of loss of material and cracking for components in the following systems:

- Auxiliary Boiler Fuel Oil
- Auxiliary Building HVAC
- Auxiliary Feedwater (AFW)
- Chemical and Volume Control
- Component Cooling Water (CCW)
- Containment HVAC
- Control Room HVAC
- Diesel Generator Lube Oil
- Diesel Jacket Water
- Starting Air
- Feedwater
- Fire Protection Fuel Oil
- Gaseous Waste Disposal
- Instrument Air
- Main Steam (MS) and Turbine Steam Extraction
- Containment Penetration, and System Interface Components for Non-CQE Systems
- Nitrogen Gas
- Primary Sampling
- Raw Water
- Ventilating Air

(2) Preventive Actions

This program does not prevent aging.

(3) Parameters Monitored or Inspected

Surface conditions of components are monitored through visual observation and inspection to detect signs of external corrosion and to detect conditions that can result in external corrosion, such as fluid leakage.

(4) Detection of Aging Effects

The aging effects of concern are loss of material and cracking. These effects can be detected by visual observation and inspection of external surfaces. Inspection for evidence of leaking fluids also provides indirect monitoring of certain components that are not routinely accessible.

(5) Monitoring and Trending

Various plant personnel including operators and system engineers perform periodic material condition inspections and observations outside containment. These inspections are performed in accordance with approved plant procedures. Evidence of fluid leaks, significant coating damage, or significant corrosion is documented.

Inspections and observations are performed at intervals based on previous inspections and industry experience. Operator rounds occur several times daily and system engineer walkdowns occur at least quarterly. Inspections inside containment are conducted each refueling outage by a team that includes knowledgeable subject matter experts from Engineering and Quality Control. The in-containment inspections for corrosion are part of the containment coatings inspections described in the OPPD response to NRC Generic Letter 98-04, *Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment.*

(6) Acceptance Criteria

Plant procedures provide criteria for determining the acceptability of as-found conditions and for initiating the appropriate corrective action. The acceptance criteria and guidance are related to avoiding unacceptable degradation of the component intended functions, and include existence of leakage, presence of corrosion products, coating defects, and elastomer cracking. Appropriate provisions of NRC and industry guidance are incorporated.

(7) Corrective Action

The FCS corrective action process provides measures to verify completion and effectiveness of corrective action.

(8) Confirmation Process

The FCS corrective action process is in accordance with 10 CFR 50 Appendix B and includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls

The procedures governing inspections and observations for external corrosion are subject to the site administrative controls process which implements the requirements of 10 CFR 50, Appendix B.

(10) Operating Experience

The activities relied on to detect loss of material, cracking, and fouling of accessible cadmium plated steel, carbon steel, cast iron, copper alloy, galvanized steel, low alloy steel, and neoprene component external surfaces and the precursors thereof are a subset of a larger number of inspection activities that result in redundant inspections. The activities credited for license renewal were selected based on their effectiveness as indicated by a review of site corrective action documents.

The activities are elements of established FCS programs that have been ongoing for years. They have been enhanced over the years based on site and industry experience. Review of plant records indicates they are effective in detecting loss of material due to corrosion and its precursors for accessible external surfaces. These findings are consistent with the findings of recent internal and external assessments of these activities, such as audits and NRC inspections.

Conclusion:

The General Corrosion of External Surfaces Program provides reasonable assurance that aging effects will be managed such that components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.4 NON-EQ CABLE AGING MANAGEMENT PROGRAM

The FCS Non-EQ Cable Aging Management Program establishes a service life value for the Non-EQ cable in a similar fashion as the FCS EQ Program establishes a Qualified Life for the safety related equipment, components, and cable. Non-EQ cable was purchased to the same requirements and specifications as that included in the EQ Program for the cable installed and qualified under the FCS 10 CFR 50.49 Environmental Qualification Program. Additional temperature and environmental data utilized to extend the qualified life of the EQ Program equipment and cables will be utilized to analyze and establish a service life for the Non-EQ cables. These analyses are relied upon to predict the life expectancy of the Non-EQ cable under the normal and abnormal plant operating conditions. Cables not capable of having a 60-year service life will be further analyzed using state of the art analytical techniques to determine if the service life can be further extended. Industry accepted and regulatory approved cable inspection techniques that provide aging related data, as well as state of the art in-situ, non-destructive testing of cable performance, and/or laboratory testing of cable to extend life, may also be considered should the aforementioned methodologies not succeed in extending the required service life.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Program

The FCS Non-EQ Cable Aging Management Program is credited for managing the aging of all Non-EQ cables and connectors in the FCS plant electrical system subject to aging management review.

(2) Preventive Actions

The program does not prevent aging from occurring.

(3) Parameters Monitored or Inspected

The FCS Non-EQ Cable Program does not credit the inspections delineated within NUREG-1801 Section XI.E1, since specific analyses are provided for the Non-EQ cable which demonstrate that the cable will perform as intended. Additionally, this analysis takes credit for the manner in which the cable was procured, i. e., same as that in the EQ Program, and the methodology used to establish the 60 year service life.

(4) Detection of Aging Effects

The EQ program, as well as the program established for the Non-EQ cable, does not detect aging effects, but rather establishes a rate of aging based on analysis of materials (i.e., the insulation system). The material analysis includes consideration of material mechanical and electrical properties and their performance in ambient environments under operational conditions as well as self-heating effects. Additional environmental conditions such as humidity and radiation are also considered in the establishment of the service life. These analyses are relied upon to predict the life expectancy of the Non-EQ cable under the normal and abnormal plant operating conditions.

(5) Monitoring and Trending

The FCS Non-EQ Cable Aging Management Program establishes a service life value for the Non-EQ cable in a similar fashion as the FCS EQ Program establishes a Qualified Life for the safety related equipment, components, and cable. Non-EQ cable was purchased to the same requirements and specifications as that included in the EQ Program for the cable installed and qualified under the FCS 10 CFR 50.49 Environmental Qualification Program. Additional temperature and environmental data utilized to extend the qualified life of the EQ Program equipment and cables will be utilized to analyze and establish a service life for the Non-EQ cables.

(6) Acceptance Criteria

Acceptance criteria are based on the cable insulation service life (i.e., the predicted life expectancy). The service life evaluation of the cable insulation material includes consideration of material mechanical and electrical properties and their performance in ambient environments under operational conditions as well as self-heating effects. Additional environmental conditions such as humidity and radiation are also considered in the establishment of the service life. These analyses are relied upon to predict the life expectancy of the Non-EQ cable under the normal and abnormal plant operating conditions.

(7) Corrective Actions:

Cables for which a 60-year service life has not been or can not be demonstrated by state of the art analysis, inspection, or test, will be replaced prior to expiration of the established service life. This action is in accordance with the FCS 10 CFR 50 Appendix B corrective action process.

(8) Confirmation Process:

N/A. Cable replacement in accordance with the current licensing basis negates the need to confirm that the corrective action was effective in assuring the cable intended function(s). As noted above, the FCS corrective action process is in accordance with 10 CFR 50 Appendix B.

(9) Administrative Controls:

Non-EQ Cable Aging Management Program activities will be subject to the FCS administrative controls process, which is in accordance with 10 CFR 50 Appendix B and requires formal reviews and approvals.

(10) Operating Experience

This program is based on the EQ program, which has been shown through operating experience to be effective in managing cable aging. There is extensive industry and FCS experience in establishing and monitoring the service life of cables and other EQ equipment. The program will be improved, as appropriate, as additional industry experience becomes available.

Conclusion:

The FCS Non-EQ Cable Aging Management Program provides reasonable assurance that aging effects will be managed such that non-EQ cables subject to aging management review will continue to perform their intended functions consistent with the current licensing basis through the period on extended operation.

B.3.5 ONE-TIME INSPECTION PROGRAM

The FCS One-Time Inspection Program will be consistent with XI.M.32, *One-Time Inspections*, as identified in NUREG-1801 prior to the period of extended operation.

Operating Experience:

This is a new FCS program implemented to meet license renewal requirements specified in NUREG-1801. Results obtained from the required program inspections will be reviewed and documented in accordance with plant procedures. Corrective actions will be taken if necessary in accordance with the plant corrective action program.

Conclusion:

The One-Time Inspection Program will provide reasonable assurance that applicable aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.6 SELECTIVE LEACHING PROGRAM

The FCS Selective Leaching Program will be consistent with XI.M.33, *Selective Leaching of Materials*, as identified in NUREG-1801 prior to the period of extended operation, with the following clarification:

- XI.M33-Program Description, 3. Parameters Monitored/Inspected, and 4. Detection of Aging Effects

  OPPD does not perform hardness measurement, because brasses, bronzes, and other copper-alloys do not have hardness acceptance criteria. For cast irons, graphitization is easily visually identified and the ASTM and ASME standards do not prescribe hardness acceptance criteria.

Operating Experience:

FCS operating experience has revealed no problems related to selective leaching.

Conclusion:

The Selective Leaching Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

Appendix C is not being used in this application.

APPENDIX D - TECHNICAL SPECIFICATION CHANGES

10 CFR 54.22 requires that an application for license renewal include any technical specification changes or additions necessary to manage the effects of aging during the period of extended operation. No changes to the Fort Calhoun Station Unit 1 Technical Specifications are necessary in that regard.