October 17, 2003

APPLICANT: Omaha Public Power District

FACILITY: Fort Calhoun Station, Unit 1

SUBJECT: SUMMARY OF MEETING WITH OMAHA PUBLIC POWER DISTRICT (OPPD) TO DISCUSS DRAFT RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION (RAIs) FOR THE RENEWAL OF THE OPERATING LICENSE FOR FORT CALHOUN STATION, UNIT 1 (FCS)

On November 20 and 21, 2002, the NRC staff (the staff) and representatives from OPPD held meetings to discuss draft responses to RAIs resulting from the staff's review of license renewal application (LRA) Sections 2.1 (Scoping and Screening Methodology), 2.1.4 (Plant Level Scoping of Systems and Structures), 2.3.1.2 (Reactor Coolant), 2.3.2.1 (Safety Injection and Containment Spray), 2.3.3.20 (Radiation Monitoring - Mechanical), 2.3.3.5 (Auxiliary Boiler Fuel Oil and Fire Protection Fuel Oil), 2.3.3.3 (Emergency Diesel Generators), 2.3.3.15 (Raw Water), 2.3.3.16 (Component Cooling), 2.3.4 (Steam and Power Conversion Systems) 2.4.1 (Containment), 2.4.2 (Other Structures), 2.4.2.5 (Fuel Handling and Heavy Load Handling Equipment), 2.4.2.6 (Component Supports), 2.5 (Scoping and Screening Results: Electrical), 3.1 (Aging Management of Reactor Coolant Systems), 3.2 (Engineered Safety Features), 3.6 (Aging Management of Electrical and Instrumentation and Control), 4.2 (Reactor Vessel Neutron Embrittlement), 4.7.1 (Reactor Coolant Pump Flywheels), B.1.7 (Reactor Vessel Integrity Program), B.2.9 (Steam Generator Program), B.3.1 (Alloy 600 Program), B.3.2 (Buried Surfaces External Corrosion Program), B.3.3 (General Corrosion of External Surfaces Program, Auxiliary Systems). Meeting participants are enclosed.

2.1 <u>Scoping and Screening Methodology</u>

RAI 2.1-1 <u>10 CFR 54.4(a)(2) Scoping Criteria for Non-Safety-Related SSCs</u>

By letters dated December 3, 2001, and March 15, 2002, respectively, the Nuclear Regulatory Commission (NRC) issued a staff position to the Nuclear Energy Institute (NEI) which described areas to be considered, and options the NRC expects licensees to use, to determine what systems, structures, or components (SSCs) meet the 10 CFR 54.4(a)(2) criterion (i.e., all non-safety-related SSCs whose failure could prevent satisfactory accomplishment of any safety-related functions identified in paragraphs (a)(1)(i),(ii),(iii) of this section).

The December 3, 2001, letter provided specific examples of operating experience which identified pipe failure events (summarized in Information Notice (IN) 2001-09, "Main Feedwater System Degradation in Safety-Related ASME Code Class 2 Piping Inside the Containment of a Pressurized Water Reactor") and the approaches that the NRC considers acceptable to determine which piping systems should be included in scope based on the 54.4(a)(2) criterion.

The March 15, 2002, letter further described the staff's expectations for the evaluation of non-piping SSCs to determine which additional non-safety-related SSCs are within scope. The position states that applicants should not consider hypothetical failures, but rather should base their evaluation on the plant's current licensing basis (CLB), engineering judgement and analyses, and relevant operating experience. The paper further describes operating experience as all documented plant-specific and industry-wide experience which can be used to determine the plausibility of a failure. Documentation would include NRC generic communications and event reports, plant-specific condition reports, industry reports such as SOERs, and engineering evaluations.

Consistent with the staff position described in the aforementioned letters, please describe your scoping methodology implemented for the evaluation of the 10 CFR 54.4(a)(2) criterion? As part of your response please indicate the option(s) credited, list the SSCs included within scope as a result of your efforts, list those structures and components (SCs) for which aging management reviews were conducted, and, for each SC, describe the aging management programs, as applicable, to be credited for managing the identified aging effects?

Response:

OPPD is still in the process of addressing the 10 CFR 54.4(a)(2) scoping criterion issue; therefore, results are incomplete. OPPD plans to finish this effort before the completion of the license renewal inspections. The following methodology is being utilized to address the issue.

1. Identify non-safety related systems with the potential for adverse spatial interaction with safety-related SSCs.

   – Review LRA Boundary drawings
   – Review drawings continued beyond the scope of boundary
   – Review plant level scoping and screening
   – Review systems identified that are not within the scope of license renewal
   – Review piping plan and elevation drawings for structures within the scope of license renewal

2. In addition to Seismic II/I systems, all non-seismic systems containing steam or liquid will be included unless specific justification exists for their exclusion, such as location remote from susceptible safety related SSCs.

   – Identify systems containing steam or liquid

3. Based on a preliminary evaluation, applicable aging management programs (AMPs) for the identified systems are expected to be the FAC Program and the Structures Monitoring Program.

– Initiate an Engineering Analysis (EA) to document the results of the evaluation
– Complete the scoping and screening evaluation and incorporate the results into the EA

4. Perform a review of site operating experience for at least the last five years. The intent of the review is to confirm that FAC-related failures are the only credible failures of non-safety related SSCs that could impact the successful completion of required safety functions. If it is determined that system or component failures as a result of aging effects other than FAC could impede safety-related systems from performing their safety functions, then additional screening and aging management review (AMR) of the system is required.

   – Review FCS operating experience for FAC failures
   – Complete AMR and incorporate the results in the EA

Meeting Discussion:

OPPD said they would include walkdown results, include air systems (and document operating experience), provide clarification for including the 2-over-1 criteria for 10 CFR 54.4(a)(2) for plant and industry experience, and justify the use of desktop reviews when walkdowns are not performed.

2.1.4 Plant Level Scoping of Systems and Structures

RAI 2.1.4-1 Pressurized Thermal Shock (PTS) and Anticipated Transient Without Scram (ATWS)

On page 2-9 of the LRA, it is stated that no additional equipment was included within the scope of license renewal due to the PTS Rule and all systems credited for ATWS mitigation are within the scope of license renewal for reasons other than ATWS mitigation. It is not clear what SSCs need to be within the scope of license renewal to meet the requirements of 10 CFR 54.4a(3) for PTS and ATWS. Identify which SSCs are credited for meeting the requirements of 10 CFR 54.4a(3) for PTS and ATWS. This information is necessary in order for the staff to have reasonable assurance that all the SSCs have been correctly identified as being within scope and subject to an AMR in accordance with 10 CFR 54.

Response:

As a general comment, 10 CFR 54.21, Contents of Application, does not require the application to identify in the LRA the criterion by which a component ultimately ends up being in scope for LR and subject to aging management review. It focuses only on those SCs subject to aging management review. The component-by-component identification of the criteria by which SSCs are within the scope of license renewal is contained in the individual system LR

Engineering Analyses (EAs) that are available for inspection at the Fort Calhoun site.

Relative to PTS, the reactor vessel belt line plates and welds are the only SSC included within the scope of LR for PTS. Relative to ATWS, as described on page 2-9 of the application, the Diverse Scram System and its structures and components are within the scope of LR for ATWS. Please see Section 2.5.17 on page 2-130 of the application.

Meeting Discussion:

OPPD agreed to verify that the requirements of 10 CFR 50.62(c)(1) are addressed. The RAI response will be revised to mirror the (c)(1) requirements.

2.3.1.2 Reactor Coolant

RAI 2.3.1.2-1 The FCS CLB for FP complies with certain sections of Appendix R, particularly Section III.G, which provides the requirements for the fire protection safe shutdown capability. Discuss if the pressurizer spray heads are credited and relied upon in the fire protection safe shutdown analysis to bring the plant to cold shutdown conditions within a given time for compliance with Appendix R. If it is credited in the fire protection safe shutdown analysis, the component would satisfy 10 CFR 50.48, Appendix R requirements and therefore should be included within the scope of license renewal.

Response:

The pressurizer spray heads are credited for safe shutdown in the Fire Protection Safe Shutdown Analysis. They are addressed in Table 2.3.1.2-1 under the component types Nozzles and Pipes and Fittings. The applicable components are linked to AMR Results Items 3.1.1.01, 3.1.1.10, 3.1.1.11, 3.1.1.21, 3.1.1.25, and 3.1.2.02.

As a general comment for the reviewer, the applicable components of all systems that are credited for fire protection have been indicated as in scope for fire protection on a component-by-component basis within each individual system LR Engineering Analysis (EA). These EAs will be available for review during the Scoping and Screening Inspection.

Meeting Discussion:

OPPD agreed to conduct more research and verify if the pressurizer spray system is needed for safe-cold shutdown conditions and will be able to reduce pressure within containment. Also, determine if more alternatives are available.

RAI 2.3.1.2-3 Steam generators (SG) are generally equipped with flow restrictors, one of whose intended functions is to limit steam line flow during a steam line rupture. Over the extended life of the plant, it is essential to maintain the flow area of the

flow restrictors used in the CLB to calculate the amount of steam released. The staff also believes that such components are susceptible to aging effects such as loss of material and cracking. Accordingly, the staff requests the applicant to provide the following information:

a) Are the SGs at FCS equipped with such components?

b) If so, include the components within the scope of license renewal and subject to an AMR, so that the intended function mentioned above can be maintained over the extended period of operation, or provide a justification for their exclusion.

Response:

The FCS SGs have no flow restrictors. The FCS flow limiters are of the venturi type and are built into the piping downstream of the first elbow in the horizontal Main Steam piping runs leaving the steam generators. For aging management, they are treated as part of the piping in which they are contained. This piping, including the limiters, is included in Table 2.3.4.3-1 under Pipes & Fittings. They are linked to AMR Results Items 3.4.1.01, 3.4.1.05, 3.4.1.06, 3.4.1.07, and 3.4.1.13.

Meeting Discussion:

OPPD will revise its response to clarify that the intended functions that must be maintained during the period of extended operation are fission product retention and containment pressure reduction capability; that this is accomplished through maintenance of adequate spray and spray distribution; and that adequate spray and spray distribution is assured by maintaining the structural integrity of the containment spray headers and nozzles, which are identified in LRA Table 2.3.1.2-3, under Pipes & Fittings. RAI will be renumbered to 2.3.1.2-3. Also, it was determined that no AMP is needed, due to the flow restrictors being constructed of stainless steel versus carbon steel.

2.3.2.1 <u>Safety Injection and Containment Spray</u>

RAI 2.3.2.1-1 LRA Section 2.3.2.1 states that the function of the Containment Spray (CS) system is to limit the containment structure pressure rise by providing a means for cooling the containment atmosphere after the occurrence of a LOCA. Pressure reduction is accomplished by spraying cool, borated water into the containment atmosphere. The CS System also reduces the leakage of airborne radioactivity by effectively removing radioactive particulates from the containment atmosphere. Removal of radioactive particulates is accomplished by spraying water into the containment atmosphere. The particulates become attached to the water droplets, which fall to the floor and are washed into the containment sump. During recirculation, the CS pumps discharge the borated water through two heat exchangers to a dual set of spray headers and spray nozzles in the containment. These spray headers are supported from the

containment roof and are arranged to give essentially complete spray coverage of the containment horizontal cross sectional area. The staff believes that the above mentioned statements in the LRA justify the need to include the spray headers and spray nozzles within the scope of license renewal, and that an aging management review be submitted in order to preserve the spraying function from degradation due to cracking, corrosion, loss of material and/or blockage. However, it appears that the subject components and the intended functions were not identified in either LRA Table 2.3.2.1-1 or drawing E-23866-210-130 as being within scope and requiring aging management. Please include these components within scope and subject to an ARM or justify their exclusion.

Response:

The containment spray ring and nozzles are WSLR as shown on Dwg. E-23866-210-130, Sheet 2 in zones B, C, D – 1, 2, 3 and are included in Table 2.3.2.1-1 under the component type Pipes & Fittings. The applicable components are linked to AMR Results Items 3.2.1.01, 3.2.1.10, and 3.2.2.04.

Meeting Discussion:

OPPD will revise its response to clarify that the intended functions that must be maintained during the period of extended operation are fission product retention and containment pressure reduction capability; that this is accomplished through maintenance of adequate spray and spray distribution; and that adequate spray and spray distribution is assured by maintaining the structural integrity of the containment spray headers and nozzles, which are identified in LRA Table 2.3.2.1-1, under Pipes & Fittings. RAI will be renumbered to 2.3.2.1-1.

2.3.3.3 Emergency Diesel Generators

RAI 2.3.3.3-2 The components (expansion joints and mufflers) are identified in drawing E-4183, Rev.1, “Diesel Generator Intake Air & Exhaust Diagram,” as being within the scope of license renewal. However, these components are not contained in Table 2.3.3.3-1, which lists components subject to an AMR. The staff believes that these components are passive and long-lived, and therefore should be subject to an AMR. Please clarify whether these components are subject to an AMR, or justify their exclusion.

* Expansion joints (C-1, E~F-1, C-8 and E~F-8)
* Mufflers (C-4 and F-4)

Response:

The expansion joints and mufflers are included in Table 2.3.3.3-1 under the component type Pipes and Fittings. The expansion joints and the mufflers are managed for aging per the AMR Results Items listed for the component type.

Meeting Discussion:

OPPD provided corrected tables for the draft response for review.

2.3.3.5 Auxiliary Boiler and Fire Protection Fuel Oil

RAI 2.3.3.5-2 LRA Table 2.3.3.5-1 states that hoses and hose couplings will be replaced based on performance or condition in accordance with the periodic surveillance and preventive maintenance program. In accordance with the guidance provided in Table 2.1-3 of the SRP-LR, hoses and hose couplings are consumable components and, as such, are typically replaced based on performance or condition monitoring that identifies whether these components are at the end of their qualified lives and may be excluded, on a plant-specific basis, from an AMR. The guidance further states that the applicant should identify the standards that are relied on for the replacement as part of the methodology description. The periodic surveillance and preventive maintenance program, as described in the LRA, does not provide such a methodology description. On this basis, the staff requests the applicant to identify the standards that are relied on for replacement.

Response:

Hoses and hose couplings identified in LRA Table 2.3.3.5-1 are inspected for fraying, cracking, splitting, embrittlement, corrosion damage, or degradation which could prevent them from performing their intended function. This inspection is performed per approved plant procedures in accordance with the Periodic Surveillance and Preventive Maintenance Program (B.2-7 in Appendix B of the application). Condition determination is made by craft and engineering judgement and, if necessary, the hose and/or couplings are replaced based on condition in accordance with the Corrective Action Program.

Meeting Discussion:

OPPD will provide verification that the Periodic Surveillance AMR provides guidance to replace and inspect hoses and hose couplings.

2.3.3.15 Raw Water

RAI 2.3.3.15-1 Drawing 11405-M-100 depicts several license renewal boundary flags at locations E-8, D-8, and D-7 that are at design class boundaries not associated with an isolation valve. Please justify the location of these boundaries with regard to protection of essential systems from internal flooding or relocate the license renewal boundary to an appropriately located isolation valve.

Response:

Boundary flags located at class boundaries on drawing 11405-M-100 reflect results of an engineering analysis and a calculation that determined that the class boundaries are acceptable at a non-valve location. Therefore, the license renewal boundary locations are acceptable as identified on drawing 11405-M-100. The engineering analysis and calculation are available for staff review at Fort Calhoun Station.

Meeting Discussion:

OPPD will revise the RAI response to include a brief discussion of the scope of the engineering analysis and associated calculation, including the possible specific negative outcomes of a pipe break at the identified locations.

2.3.3.16 Component Cooling

RAI 2.3.3.16-2 Drawing 11405-M-119, for the component cooling water system depicts the control element assembly seal coolers as within license renewal scope as part of the reactor vessel internals, and the associated component cooling water supply and return piping as within scope for the component cooling water system. However, LRA Table 2.3.1.1-1, which lists components comprising the reactor vessel internals, does not include the control element assembly seal coolers nor their intended function of maintaining the component cooling water system pressure boundary. Also, LRA Section 2.3.1.1, does not reference drawing 11405-M-119. Please clarify whether the control element assembly seal coolers are included within the scope of license renewal and subject to an AMR or justify their exclusion. In addition, please provide drawings and other design information for the control element assembly seal area that provides sufficient detail to identify other potential intended functions of the seal cooler, such as reactor coolant system pressure boundary and heat transfer (i.e., the seal must be cooled to maintain reactor coolant system pressure boundary integrity)."

Response:

Drawing 11405-M-119 incorrectly identified the CEDM Seal Housing Assemblies as being included within the Reactor Vessel Internals system. They are actually included with the Reactor Vessel (RV). The drawing has been corrected and is included electronically with the response to this RAI.

The CEDM Seal Housing Assembly Coolers are within license renewal scope, but were inadvertently omitted during the IPA because they were not part of the CEOG Generic Aging Management Review Report for the Reactor Vessel. They have been added to the FCS IPA. Drawing CND-E-2935, Seal Housing Assembly Details, has been included electronically with this RAI response to show the configuration of this cooler. It consists of a machined depression in the housing over which a nippled sleeve is fitted and welded into place such that a

cooling water channel is created. The drawing is not a LR Boundary P&ID. It does not, therefore, have LR boundary flags.

The CEDM Seal Housing Assembly is a subcomponent within the Component Type ‘Pipes and Fittings, CEDM Housings’ in Table 2.3.1.3-1 of the LRA. It is fabricated of austenitic stainless steel, has an internal environment of borated, treated water >482°F, and an external environment of containment air. A further subcomponent to this assembly has been added to the FCS IPA. The new subcomponent is the CEDM Seal Housing Assembly Cooling Channel. It has only an internal environment of nitrite-corrosion-inhibited, treated water (component cooling water). Its external environment is the external environment of the housing assembly itself. The AMPs credited for the aging management of the cooling channel are the Chemistry Program and the Cooling Water Corrosion Program. Links to LRA AMR Items 3.3.3.13 and 3.3.3.15 will be added to the "Pipes and Fittings, CEDM Housings" Component Type in Table 2.3.1.3-1 of the LRA.

Meeting Discussion:

OPPD agreed to revise the RAI to clarify that the purpose of the control element assembly seal is only for protection of the non-safety-related control element assembly motor because the motor is within the reactor coolant system pressure boundary. Therefore, the intended function of the seal cooler assembly is limited to providing a pressure boundary for the safety-related component cooling water system.

2.3.3.20 Radiation Monitoring-Mechanical

RAI 2.3.3.20-1 Drawing 11405-M-1, Sheet 2 is the only drawing listed as showing the license renewal boundaries for this system. The drawing appears to show only three equipment cabinets as being within the scope of license renewal. Table 2.3.3.20-1 lists five component types subject to aging management review. Clarify where the components within the scope of license renewal for the Radiation Monitoring-Mechanical system are shown and/or listed? Provide an inclusive drawing or drawings showing the Radiation Monitoring-Mechanical system license renewal boundaries. This information is necessary in order for the staff to have reasonable assurance that all the SSCs have been correctly identified as being within scope and subject to an AMR in accordance with 10 CFR 54.

Response:

There were three (3) LR Boundary Drawings for the Radiation Monitoring – Mechanical System that were not sent with the application because they are proprietary to the vendor that supplied the radiation monitors. The vendor does not allow reproduction of these drawings for distribution to parties outside of the organization of the purchaser of the equipment. For this reason, these drawings were not docketed with the LRA. There is a hard copy of each drawing included

in EA-FC-00-115, Radiation Monitoring – Mechanical Scoping, Screening, and Aging Management Review for License Renewal, that will be available to the NRC Inspection Team or these drawings can be transmitted to the NRC as proprietary information.

Meeting Discussion:

OPPD will provide a more formal procedure or method for NRC reviewers to obtain proprietary information.

2.3.4 Steam and Power Conversion Systems

RAI 2.3.4-1 The Steam Generator Blowdown System is identified in LRA Section 3.4 as being included in the Steam and Power Conversion Systems group. The steam generator blowdown system is not part of the Steam and Power Conversion Systems listed in LRA Section 2.3.4. Additionally, LRA Table 2.2-1, "Plant Level Scoping Results," lists the steam generator feedwater blowdown system as being within the scope of license renewal. Given these discrepancies, in order for the staff to understand whether the steam generator feedwater blowdown system is within scope and subject to an AMR, please identify where in the application the steam generator feedwater blowdown system is addressed.

Response:

The Steam Generator Blowdown System is WSLR as noted in Table 2.2-1; however, a link was not provided to a system component type screening table as was done for the other WSLR systems because the system has been evaluated within other WSLR systems. The Steam Generator Blowdown System component type screening results have been included with the applicable component types listed in Table 2.3.1.2-1, Reactor Coolant System (includes SGs); Table 2.3.2.2-1, Containment Penetration and System Interface Components for Non-CQE Systems; Table 2.3.3.19-1, Primary Sampling; and Table 2.3.4.1, Feedwater.

Meeting Discussion:

OPPD agreed to provide the NRC with more details on system components and describe which components are in each of the above discussed systems.

2.3.4.1 Feedwater

RAI 2.3.4.1-1 There are numerous pressure and level transmitters highlighted on Drawing 11405-M-253, Sheet 1. What is the intended function of the pressure and level transmitters? From the drawing, it appears the instrument housings form part of a pressure boundary with their associated piping. Therefore, the instrument housings should be listed in Table 2.3.4.1-1 as being subject to an AMR in accordance with 10CFR54.21. Justify not making the instrument housings subject to an AMR.

Response:

OPPD has followed the guidance provided in Appendix B of NEI 95-10, Revision 3, (endorsed by NRC) for pressure and level transmitters (Item 105) and scoped these as active components that do not require AMR. Appendix B of NEI 95-10 is explicit regarding electrical and I&C components that need to be subject to AMR due to a pressure boundary function.

Meeting Discussion:

OPPD agreed to clarify whether the instrument housings are either a pressure boundary and subject to an AMR or not a pressure boundary and not subject to an AMR.

2.4.1 Containment

RAI 2.4.1-1 Section 2.4.1 of the LRA states that the tendon anchors are accessible for inspection, testing, and re-tensioning via the tendon access gallery located beneath the containment cylindrical wall and at the dome roof. Table 2.4.1-1 of the LRA lists all the components for the containment that are subject to an AMR. However, the tendon access gallery is not listed in the table. The staff believes that these components are long-lived components with a passive function, and therefore are subject to an AMR in accordance with 10 CFR 54.21. Explain whether the concrete structure of the tendon access gallery is in scope and subject to an AMR for license renewal. Provide justification if it should not be in scope.

Response:

The function of the tendon gallery is to provide access to the tendon anchorage for inspection and testing. The concrete structure of the tendon gallery does not provide support for Containment, and does not make up part of the pressure boundary of Containment (i.e., it is not required to prevent or mitigate the consequences of an accident that could result in potential offsite exposure). Therefore, it is not in scope for license renewal (i.e., it has no intended function per 10 CFR 54.4). However, the concrete where the tendons are anchored in the tendon gallery is within scope for license renewal.

Meeting Discussion:

OPPD will provide architectural layout drawings for verification for this RAI system.

RAI 2.4.1-7 Section 5.11 of the Updated Safety Analysis Report (USAR) states that special steel structures were used around the steam generators for the purpose of limiting the motion of the steam generator in case a rupture occurs in the reactor coolant piping or main steam piping, or in the feedwater pipe. These special steel structures are not addressed in Section 2.4.1 of the LRA. The staff

believes that these passive long-lived structures are needed to ensure the functionality of the steam generators and are therefore within the scope of license renewal and subject to an AMR. Clarify whether these components are within scope and subject to an AMR, or justify their exclusion.

Response:

These special structures are included in the component type "Component Support Weathering Carbon Steel in Ambient Air" in LRA Table 2.4.2.6-1. They are in scope for license renewal and are subject to AMR.

Meeting Discussion:

OPPD agreed to provide a list of the specific components and strucutres of this special system.

2.4.2 Other Structures

RAI 2.4.2-1 Please provide general plan drawings for the containment, auxiliary building, turbine building, and service building that show the structural arrangement and internals, and highlight the boundaries that are within the scope of license renewal. This information is necessary in order for the staff to have reasonable assurance that all the SSCs have been correctly identified as being within scope and subject to an AMR in accordance with 10 CFR Part 54.

Response:

The structural drawings for the above referenced buildings were not marked up with LR boundaries and forwarded with the other LR Boundary drawings because Sections 2.4.1, 2.4.2.1, and 2.4.2.2 of the LRA indicate that the above structures are within the scope of license renewal. This means that every portion of these buildings identified on their respective structural drawings is within the scope of license renewal.

Meeting Discussion:

OPPD agreed to provide a architectural drawings for verification which demonstrates a clear connection between turbine building and service building.

2.4.2.1 Auxiliary Building

RAI 2.4.2.1-1 Section 2.4.2.1 of the LRA states that the spent fuel pool, which consists of a stainless steel lined concrete structure, is contained within the auxiliary building. However, Table 2.4.2.1-1 of the LRA only lists the spent fuel pool liner as the component subject to an AMR. The staff believes that other components of the spent fuel pool structure meet the criteria in 10 CFR Part 54 and should be included within the scope of license renewal and be subject to an AMR. Please clarify what other component types listed in LRA Table 2.4.2.1-1 (or in another table) are applicable to the spent fuel pool structure.

Response:

The spent fuel pool concrete is included in the component type “Auxiliary Building Concrete in Ambient Air” in LRA Table 2.4.2.1-1.

Meeting Discussion:

OPPD will revise the RAI to describe the locations of racks, mechanical, and structural components in the Spent Fuel Pool System.

2.4.2.2 Turbine Building/Service Building

RAI 2.4.2.2-1 Section 2.4.2.2 of the LRA describes the turbine building and service building. Table 2.4.2.2-1 of the LRA lists the component groups that have the intended functions to act as structural support to non-CQE, pipe restraints, and high-energy-line-break shielding. It is not clear from the information provided which portions of these buildings are in scope and what are the components that perform these intended functions. Specify the structural components of the turbine building and service building that are in scope for license renewal and subject to an AMR.

Response:

The intended function “provide pipe whip restraint” is fulfilled by the main steam and feedwater pipe whip restraints for High Energy Line Break (HELB) Analysis. The intended function “provide shielding against HELB” is fulfilled by the plates attached to or adjacent to the Turbine Building side of the Auxiliary Building wall. The intended function “provide structural support to non-safety related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions” is fulfilled by the concrete and structural steel of the Turbine and Service Buildings. The Turbine Building concrete and structural steel support the pipe restraints and HELB shielding. Service Building concrete and the structural steel support a CQE component for the Raw Water System (valve HCV-2861 is located in the Service Building basement).

Meeting Discussion:

OPPD will clarify in a revised RAI response that the entire turbine building is included within the scope for License Renewal.

2.4.2.5 Fuel Handling and Heavy Load Handling

RAI 2.4.2.5-1 Section 2.4.3.2 , "Structural Components Subject to an Aging Management Review," of NUREG-1800 states that, in general, structural components are "passive" and "long-lived." Thus, they are subject to an AMR if they are within the scope of license renewal. For each of the plant-level structures within the scope of license renewal, an applicant should identify those structural components that have intended functions. LRA Table 2.4.2.5-1 lists the following structures:

concrete slab removal cranes	containment crane
containment equipment hatch crane and jib	deborating demineralizing area crane
fuel transfer conveyor	new and spent fuel handling tools
refueling area crane	refueling machine
tilting machine	upper guide lift rig
waste evaporator equipment handling crane	reactor vessel closure head lift rig

For these SSCs the applicant should have identified structural components of beams, supporting columns, base plates, rails, rail clips, crane girders, crane bridge, structural members, monorail flanges, monorail, rail bolts, anchorages, trolley rails, trolley, baseplates and anchors for attachment to structures, and retaining clips. LRA Table 2.4.2.5-1 does not include the above structural components which should be included within the scope of license renewal and subject to AMR. If these structural components are not subject to AMR the applicant should provide a justification for their exclusion from LRA Table 2.4.2.5-1.

Response:

This RAI lists all of the components in Table 2.4.2.5-1 and then focuses on subcomponents that are found in cranes and other lifting devices. It is therefore assumed that the RAI is meant to address the aging management of crane subcomponents.

For the cranes that are within scope of license renewal, Table 2.4.2.5-1 identifies the link to AMR Item 3.3.1.15. This AMR Item is in Table 3.3.1 indicating that the aging management of these items is consistent with the GALL Report. In this case, AMR Item 3.3.1.15 is consistent with GALL Report Items VII.B.1-b and

VII.B.2-a. It identifies the Overhead Heavy Load and Light Load Handling Systems Program (GALL Report Section XI.M23) and indicates in the Discussion column of the AMR Item that OPPD aging management results are consistent with the GALL Report.

In the FCS LRA, this means that the equipment, the material, the environment, the AERMs, and the AMP are the same as those included in the applicable GALL Report line items. No exceptions have been taken to XI.M23 and, in fact, enhancements have been made to this program to better address the aging of concrete anchors for the equipment included in the program. These enhancements are discussed in Section B.2.6 of Appendix B of the LRA.

Since the aging management of the cranes is consistent with the GALL Report, which does not provide a detailed listing of crane/lifting device subcomponents, OPPD did not deem it necessary to list subcomponents in Table 2.4.2.5-1.

Meeting Discussion:

OPPD will revise the RAI response to reflect the engineering analysis which includes system subcomponents for scoping.

RAI 2.4.2.5-2 LRA Table 2.4.2.5-1 identifies the spent fuel storage racks as having an intended function of providing structural support to CQE reactivity control. However, the staff, after review of USAR Section 9.5.1.2, “Prevention of Criticality During Transfer and Storage,” found that boral panels protected with stainless steel were attached to the racks to support the prevention of criticality in the spent fuel pool. The staff finds that the passive, long-lived boral panels and their stainless steel covering should be included within the scope of license renewal and subject to AMR, or the applicant should provide a justification for their exclusion. Additionally, the staff finds that the boral panels and the spent fuel storage rack arrangement support the prevention of criticality within the spent fuel pool. As a result, they perform an intended function of preventing criticality. The intended function of preventing criticality is not included within LRA Table 2.4.2.5-1. If it should not be included, the applicant should provide its justification for excluding the intended function of preventing criticality from LRA Table 2.4.2.5-1.

Response:

Per telecon with the NRC, no response is required for this RAI. A discussion will be provided in the telecon summary to address this issue.

Meeting Discussion:

OPPD will revise the RAI response to provide a link for the location of where passive, long-lived boral panels and stainless steel coverings are located in the License Renewal Application. Table 2.4.2.5-1 will be revised for clarification.

RAI 2.4.2.5-3 USAR Section 14.24, “Heavy Load Incident,” identifies heavy load cranes that were evaluated following the guidelines of NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants.” The evaluations were performed to determine compliance with the following criteria of NUREG-0612, Section 5.1:

1. Any release of radioactive material that may result from damage to spent fuel based on calculations involving accidental dropping of postulated heavy load will produce doses that are well within 10 CFR Part 100 limits of 300 rem thyroid, 25 rem whole body (analyses show that doses are equal to or less than one-fourth of Part 100 limits);

2. Damage to fuel and fuel storage racks based on calculations involving accidental dropping of a postulated heavy load does not result in a configuration of fuel such that k_{eff} is larger than 0.95;

3. Damage to the reactor vessel or the spent fuel pool based on calculations of damage following accidental dropping of a postulated load is limited so as not to result in water leakage that could uncover the fuel, (makeup water provided to overcome leakage should be from a borated water source of adequate concentration if the water being lost is borated); and

4. Damage to equipment in redundant or dual safe shutdown paths, based on calculations assuming the accidental dropping of a postulated heavy load, will be limited so as not to result in a loss of required safe shutdown functions.

The staff found that the containment polar crane, auxiliary building crane, and intake structure overhead crane met one or more of the above criteria and as such should be included within the scope of license renewal, and its passive long-lived structural components should be subject to an AMR. Otherwise, the applicant should provide a justification for excluding the above cranes and their passive long-lived structural components from the scope of license renewal.

Response:

The Containment Crane is within the scope of license renewal and is included in Table 2.4.2.5-1.

The Auxiliary Building Crane at FCS is called the Refueling Area Crane. It is within the scope of license renewal and is included in Table 2.4.2.5-1.

Administrative operating restrictions and the presence of rail guides (travel limiters) provide the basis for the exclusion of the Intake Structure Traveling Crane from the scope of license renewal.

Meeting Discussion:

OPPD will revise the RAI response to clarified that the intake structure traveling crane is not in scope.

2.4.2.6 Component Supports

RAI 2.4.2.6-1 10 CFR 54.21(a)(1) requires the applicant to identify and list structures and components subject to an AMR. The staff found that the applicant, in LRA Table 2.4.2.6-1, had not uniquely identified and listed component supports. Instead, LRA Table 2.4.2.6-1 generically refers to component support and provides the material and environment in the first column of the table. The staff believes that components such as battery racks, cable tray and conduit, cable tray and conduit supports, Class 1 (NSSS) supports, control boards, control room ceiling, HVAC duct supports, instrument racks and frames, instrument line supports, lead shielding supports, pipe supports, electrical and instrument panels and enclosures, equipment component supports, wireway gutters, and stair, platform and grating supports should be included within the scope of license renewal and subject to an AMR. Otherwise, the applicant should provide a justification for their exclusion from LRA Table 2.4.2.6-1.

Response:

Component Supports have been treated as a commodity group. For all of the components included within the scope of license renewal, the applicable supports are also within the scope of license renewal and are contained within the commodity groupings of Component Supports. OPPD determined that there was no need to list specific support types in the AMR results. They have been grouped by material, environment, general support type, and aging management program. With the exception of stainless steel components in air and structural stainless steel in borated water, which are not addressed in the GALL Report, the aging management of all component supports within scope of license renewal is consistent with the provisions of the GALL Report as indicated in Table 2.4.2.6-1 by the links to AMR Items from the 3.5-1 and 3.5-3 AMR Tables.

Meeting Discussion:

OPPD will revise the RAI response to reflect the engineering analysis which includes system components for scoping.

2.5 Scoping and Screening Results: Electrical

RAI 2.5-1 The screening results in LRA Section 2.5 do not include any offsite power system structures or components. 10 CFR 54.4(a)(3), requires that, "all systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission’s regulations for.....station blackout (10 CFR 50.63)" be included within the scope of license renewal. 10 CFR 50.63(a)(1), requires that each light-water-cooled power plant licensed to operate be able to withstand and recover from a station blackout of a specified duration (the coping duration) that is based upon factors that include" (i) The redundancy of the onsite emergency power sources; (ii) The reliability of the onsite emergency power sources; (iii) The expected frequency of loss of offsite power; and (iv) The probable time needed to restore offsite power." Licensees’ plant evaluations followed the

guidance specified in NRC Regulatory Guide (RG) 1.155, “Station Blackout,” and NUMARC 87-00, ”Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors" (1991), to determine their required plant specific coping duration. The criteria specified in RG 1.155 to calculate a plant specific coping duration were based upon the expected frequency of loss of offsite power and the probable time needed to restore offsite power, as well as the other two factors (onsite emergency ac power source redundancy and reliability) specified in 10 CFR 50.63(a)(1). In requiring that a plant’s coping duration be based on the probable time needed to restore offsite power, 10 CFR 50.63(a)(1) is specifying that the offsite power system be an assumed method of recovering from an SBO. Disregarding the offsite power system as a means of recovering from an SBO would not meet the requirements of the rule and would result in a longer required coping duration. The function of the offsite power system within the SBO rule is, therefore, to provide a means of recovering from the SBO. This meets the criteria within license renewal 10 CFR 54.4(a)(3) as a system that performs a function that demonstrates compliance with the Commission’s regulations on SBO. Based on this information, the staff requires that applicable offsite power system structures and components be included within the scope of license renewal and subject to an aging management review, or additional justification for their exclusion be provided.

The staff guidance on the scoping of equipment relied on to meet the requirements of the SBO rule are documented in the staff letter from NRC (Matthews) to NEI (Nelson) and UCS (Lochbaum) dated April 1, 2002 (ML020920464).

Response:

The OPPD license renewal documentation has been revised to comply with the NRC Interim Staff Guidance (ISG) related to SBO. OPPD has completed the FCS SBO analysis; that analysis will be reflected in the LRA annual update as well as the update to the USAR Supplement, as appropriate. A copy of the SBO Boundary drawing is included for your review. There is no medium voltage cable within the scope of the SBO analysis. Low voltage <2Kv (typically 120 v) cable is located in Troughs and Duct Banks. High Voltage > 5Kv cable is located overhead on towers (Single Phase). The arrangement of the Duct Banks is such that they are pitched/sloped at no less than 1/8“ per foot to maintain cable out of long term water immersion by precluding standing water.

The substation (SBO restoration) includes transformers, circuit breakers, disconnect switches (manual and motor operated) High voltage buswork and transmission cables, transmission towers, supports, actuating relays, blocking relays, indicating lights, alarm logic, and miscellaneous electronic components and switches to allow isolation, transformation, and distribution, of 345-Kv, 161-Kv, and 22-Kv power to supply the plant 4.16-Kv system.

Two offsite startup power sources are available. The dedicated offsite 161-Kv system is brought in via two161-Kv/4.16Kv transformers. The 345-Kv system can be converted to an offsite power source by opening the motor operated main

generator/transformer disconnect switch DS-T1 and back feeding the plant using the main transformer as a step-down transformer to 22-Kv power in order to feed the 22-Kv/4.16-Kv transformers.

Either offsite power source can operate the four 4.16-Kv buses, 1A1, 1A2, 1A3 and 1A4. During normal plant startup the house service buses (1A3 and 1A4) and their associated 480-Volt systems are normally connected to the 161-Kv system and buses 1A1 and 1A2 are normally supplied from the 345-Kv/22-Kv back feed. During generator synchronization the four 4.16-Kv buses are powered from the 161-Kv supply.

The SBO Equipment includes Transformers, Circuit Breakers, Disconnect Switches (manual and motor operated), high voltage Bus work, Aluminum Conductor Steel Reinforced (ACSR) transmission cables, Insulators associated with the transmission conductors, Transmission Towers and supports, actuating relays, blocking relays, indicating lights, alarm logic, medium and low voltage cable, connectors, terminal blocks, fuse blocks, and miscellaneous electronic components and switches to allow isolation, transformation, and distribution, of 345-Kv, 161-Kv, and 22Kv power to supply the plant 4.16-Kv system.

All electrical components within the Substation SBO Restoration System have been considered and have been evaluated as in the license renewal boundary with the exception of Enclosures, Panels, Terminal Blocks, Fuse Blocks, Connectors and Medium and Low Voltage Cables. Enclosures, Panels, and Power Supplies were identified as Commodity Groups and are reviewed separately. Medium and Low Voltage Cables, Terminal Blocks, Fuse Blocks & Connectors are evaluated as a Commodity Group for the entire plant.

The SBO License Renewal Intended functions are:

– The Substation SBO Restoration system, as configured, provides, transforms, distributes, and isolates 345 Kv and 161 Kv power, respectively to support recovery from station blackout.

– The 22Kv system is the receiver for the 345Kv back feed as an alternate off site power source.

– The SBO Equipment provides an additional source of offsite power by back feeding from the 345-Kv system via the main and unit auxiliary transformers, and the isolated phase bus-duct

– The 161Kv substation, 1251, is the normal supply source for safety related busses 1A3 and 1A4 during all modes of plant operation.

– The following are the AMR results tables from Section 9.20 of the EA; these show the programs that have been credited for aging management.

The AMR results tables from Section 9.20 of the EA will be provided upon request.

Meeting Discussion:

OPPD agreed to clarify the statements regarding medium voltage cables and add a note to Table 1 describing the location of the 22kv cable. OPPD will also clarify that all medium voltage cables associated with Station blackout are included in the general commodity cable program.

3.1 Reactor Coolant Systems

RAI 3.1.1-1 Several line items in LRA Tables 3.1-1 (3.1.1.02, .3.1.1.16, and 3.1.1.17), and 3.1-2 (3.1.2.06 and 3.1.2.14) indicate that the Steam Generator Program includes methods to detect general, crevice and pitting corrosion of the steam generator shell assembly, loss of section thickness due to FAC for components identified in these items. However, the steam generator program described in the GALL report only discusses corrosion of steam generator tubes; it does not discuss corrosion, pitting, ligament cracking or FAC for components identified in these items. Identify the methods of detecting general corrosion and pitting of the steam generator shell assembly that are discussed in Information Notice 90-14 (sic) "Cracking of the Upper Shell-to-Transition Cone Girth Welds in Steam Generators," January 26, 1990, and loss of section thickness due to FAC for reactor coolant system components identified in line items 3.1.1.02, 3.1.1.16, 3.1.1.17, 3.1.2.06, and 3.1.2.14. In addition, confirm that the Steam Generator Program identified in Item 3.1.1.15 is program B.2.9.

Response:

(Note RAI typo: Information Notice 90-14 referenced instead of 90-04)

Information Notice (IN) 90-04, "Cracking of the Upper Shell-to-Transition Cone Girth Welds in Steam Generators," mainly addressed cracking; however, corrosion and pitting were mentioned:

> *A common factor was the general corrosion pitting on the inside surface of the SGs. Metallography information indicates that the degradation probably results from corrosion-assisted thermal fatigue.*

IN 90-04 noted that these flaws have only been observed in Westinghouse Model 44 and Model 51 vertical, recirculating, U-tube SGs with the feedwater ring design. FCS has Combustion Engineering (CE) steam generators. Based on an evaluation from CE, OPPD concluded that shell-to-cone girth welds at FCS will not be susceptible to cracking.

As Appendix B.2.9 indicates in the clarification bullets, the scope of the Steam Generator Program is expanded to include components in Tables 3.1.1 and Table 3.1.2 for which the Steam Generator Program is identified as an aging management program.

To address the other aging effects not associated with steam generator tubes, the Appendix B.2.9 scope clarifications have been amended as indicated by the italicized text and included with the response to this RAI.

> 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. *These activities involve secondary side steam generator visual inspections for loss of material.*
>
> The scope of the FCS Steam Generator Program includes those plant-specific components identified in Tables 3.1.2 of this application for which the Steam Generator Program is identified as an aging management program. *Aging management of these components involves secondary side steam generator visual inspections for loss of material.*

General corrosion and pitting of the steam generator shell assembly is managed by visual inspection of the secondary side of the steam generator.

The loss of material mechanisms in LRA AMR Items 3.1.1.02, 3.1.1.16, 3.1.1.17, 3.1.2.06 and 3.1.2.14 are managed by a visual inspection of the secondary side of the steam generator.

Discussion Item 2 in LRA AMR Item 3.1.1.15 has a typo in the Steam Generator Program reference. It will be corrected from "B1.7" to "B.2.9."

Meeting Discussion:

OPPD agreed to provide the CE evaluation which concludes that shell-to-cone girth welds are not susceptible to cracking or to provide a summary of the CE evaluation.

RAI 3.1.1-2 The GALL report indicates that the growth of intergranular separation (underclad cracks) in low alloy or carbon steel heat affected zones under austenitic stainless steel cladding is a TLAA to be evaluated for the period of extended operation for all the SA 508-CL2 forgings where the cladding was deposited with high-heat-input welding process. The applicant indicates underclad crack growth due to cyclic loading was not identified as a TLAA for FCS.

Underclad cracks were observed in SA 508 Class 3 nozzles clad with multiple-layer, strip electrode, submerged-arc welding processes where preheating and post-heating were applied to the first layer but not to the subsequent layers. In order for the staff to determine whether this issue is a TLAA for FCS, provide the following information:

a. Identify any reactor vessel components that were fabricated from SA 508 Class 2 or 3 forgings.

b. Indicate whether any of the SA 508 Class 2 or 3 forgings identified above are susceptible to underclad cracking.

c. Indicate whether any of the SA 508 Class 2 or 3 forgings are subject to neutron embrittlement (i.e., subject to a neutron fluence greater than or equal to 10^{17} n/cm^2 [E>1MeV]).

d. If any forgings are susceptible to underclad cracking, identify the basis for concluding that the cracks will not result in loss of reactor vessel integrity during the period of extended operation. The assessment should consider the impact of fatigue and neutron embrittlement on the underclad cracks.

Response:

The GALL Report identifies this scenario (Item IV.A2.5-d) only for the reactor vessel shell if it is made from SA 508, Class 2 and is exposed to a neutron fluence >10^{17} n/cm^2. The reactor vessel shell components at FCS are fabricated of SA 533, Grade B, Class 1. This scenario does not, therefore, apply to FCS.

Other reactor vessel components that are fabricated of SA 508, Class 2 steel and clad with a stainless steel or nickel-based alloy weld overlay are the reactor vessel flange, the closure head flange, and the primary coolant nozzles, nozzle extensions, and nozzle safe ends. Per a recent Westinghouse analysis performed for FCS, the flanges and nozzles will not experience a fluence of >10^{17} n/cm^2 by the end of the period of extended operation.

Meeting Discussion:

OPPD will obtain information from CE regarding underclad cracking.

RAI 3.1.1-3 LRA Table 3.1-1, row 3.1.1.09, indicates that crack initiation and growth due to SCC and PWSCC in PWR core support pads, instrument tubes (bottom head penetrations), pressurizer spray heads, and nozzles for the steam generator instruments and drains are managed by the Alloy 600 Program (B.3.1). The application indicates that the 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. XI.M11 indicates the scope of the program is to include those components currently identified as susceptible to PWSCC and those that will be susceptible during the period of license renewal. On this basis, using the latest model for susceptibility of Alloy 600 components to PWSCC, identify all Alloy 600 components that are susceptible to PWSCC during the current license term and the period of license renewal and identify the inspection methods to be used to detect PWSCC.

Response:

With the exception of the discussion column in LRA AMR Item 3.1.1.09, this comes directly from Volume 1 of the GALL Report. It is line 13 (first item on

page 8) from Table 1, Summary of Aging Management Programs for the Reactor Coolant System Evaluated in Chapter IV of the GALL Report.

The Alloy 600 components at FCS that are subject to PWSCC and are managed for aging per LRA AMR Item 3.1.1.09 are identified in Table 2.3.1.3-1 as the Core Stabilizing Lugs, the Core Support Lugs, the Surveillance Capsule Holders, and the associated RV Closure Head, Lower Shell, Middle Shell, Bottom Head, and RV Flange weld cladding.

The Alloy 600 Program is a new program at FCS. With this being the case, inspection methodologies for all of the components in the program have not yet been determined. Some of the components that are in the program are currently part of other programs like the Reactor Vessel Internals Inspection Program. The activities that occur under the interfacing programs relative to these components will be utilized to help analyze and determine the methodologies to be incorporated within the Alloy 600 Program for inspection of its included components. These analyses and determinations will be completed prior to entry into the period of extended operation.

Meeting Discussion:

See RAI 3.1.1-3.

RAI 3.1.1-4 Programs identified in NUREG-1801 are generic programs. When components experience unusual aging effects, the programs identified in NUREG-1801 may not be applicable. CRD Housings (LRA Table 3.1-1, row 3.1.1.25) are identified as being susceptible to SCC and PWSCC with aging management provided by the Inservice Inspection Program and Water Chemistry Program. Cracking has been reported on CRD Housings at FCS (January 25, 2002 letter from OPPD) and Palisades (Nuclear Management Company letters to the NRC dated August 20, 2001 and March 14, 2002). The Palisades and FCS CRD housing have a similar design.

Because this operating experience was not considered in the development of the LRA, the staff requests the following information in order to understand how this experience impacts license renewal:

a. Identify the CRD locations, the materials and aging mechanisms that are responsible for the cracking in the CRD Housings at FCS and Palisades.

b. Identify any design, materials and environmental factors that would preclude cracking of the type identified in Item (a) above.

c. Identify how the cracks in Item (a) were detected. Identify the current program and the frequency of examination required to ensure that the cracks in Item (a) do not result in loss of CRD Housing integrity. Were the cracks detected using NDE methods identified in the Inservice Inspection Program? Were alternative examination methods (methods not identified in the ASME Code) used to detect these cracks ?

d. As a result of the discussion above, will the Inservice Inspection Program and Water Chemistry Program be adequate for managing the aging effects discussed in Item (a)

Provide the basis for this conclusion.

Response:

The cracks that were discovered at FCS in 1990 occurred in two unvented, spare CEDM upper housings. Details are documented in FCS Licensee Event Report 90-28, dated January 14, 1991 (LIC-91-0003L). The cracks were discovered based on the visual observation of boric acid in the vicinity of the cracks while on line that resulted in a forced outage. OPPD determined that the unvented, stagnant conditions in these spare CEDMs were responsible for the cracking and as a result, these conditions were eliminated. Since that time, there has been no further cracking of the CEDMs at FCS.

Because of industry experience, OPPD in 1999 began a proactive approach to dealing with the CEDM cracking phenomenon with the establishment of a CEDM Material Reliability Management Program Plan to monitor the CEDMs on an outage-by-outage basis through the performance of eddy current testing of the CEDMs. Details of the OPPD approach are contained in a letter from OPPD (R. L. Phelps) to NRC (Document Control Desk), dated January 25, 2002, "Fort Calhoun Station (FCS) Discussion of Control Element Drive Mechanism (CEDM) Housing Reliability" (LIC-02-0007), and in a letter from OPPD (R. L. Phelps) to NRC (Document Control Desk), dated October 15, 2001, "Fort Calhoun Station (FCS) Control Element Drive Mechanism (CEDM) Housing Reliability Management" (LIC-01-0095).

The program activities described in the referenced letters are not regulatory commitments, but will be continued unless other events or analyses dictate otherwise. OPPD considers this to be a current licensing basis issue, with the resolution carried over if necessary into the period of extended operation.

Meeting Discussion:

OPPD will clarify whether the ISI and Water Chemistry Programs are adequate for managing the aging effects for CRD Housing cracking.

RAI 3.1.2-1 LRA Table 3.1-2, rows 3.1.2.04 and 3.1.2.05, indicates that the steam generator lower head, manway cladding, primary side tube sheet and reactor coolant pump thermal barrier are subject to cracking and the aging management is the Chemistry Program. The Chemistry Program will, to some extent, mitigate cracking; but will not monitor cracking. Provide your basis for concluding monitoring of crack initiation and growth is not necessary for these components. If adequate justification is not provided, provide a program to monitor crack initiation and growth.

Response:

For the steam generator primary head cladding, the Steam Generator Program will also be credited for aging management; therefore, in Table 2.3.1.2-1 (page 2-34 of the application), the Aging Management Review Results link of 3.1.2.04 will be changed to 3.1.1.15 for "Steam Generator Primary Head (Cladding)."

"Steam Generator Primary Manways (Cladding)," is the next line item in Table 2.3.1.2-1. It references AMR Results Item 3.1.1.33, which credits both Chemistry and the In-Service Inspection Program for the aging management of cracking. Although this activity is not identified in the GALL, it is a credited activity in the FCS Steam Generator Program.

For the primary side tubesheet cladding, as for the steam generator primary head cladding above, the Steam Generator Program will also be credited for aging management. Therefore, in Table 2.3.1.2-1 (page 2-35 of the application), the Aging Management Review Results link of 3.1.2.04 will be changed to 3.1.1.15 for "Steam Generator Tubesheet (Primary Side)." Although this activity is not identified in the GALL, it is a credited activity in the FCS Steam Generator Program.

The reactor coolant pump (RCP) thermal barriers are not accessible for routine maintenance or inspection. During the 2001 refueling outage, the "A" RCP rotating assembly was replaced with a new rotating assembly and the existing assembly was sent to a vendor for refurbishment. As part of the refurbishment the thermal barrier on the "A" RCP was visually inspected and a dye-penetrant exam performed. No indications of cracks were identified. A visual inspection was performed on the "C" RCP after it was removed for refurbishment during the 2002 refueling outage. No indication of degradation was identified. OPPD will continue to visually inspect and perform a dye-penetrant exam on the two remaining RCP thermal barriers when the rotating assemblies are refurbished.

Meeting Discussion:

OPPD agreed to include the fracture mechanics analysis in the One-Time Inspection program (GALL XI.M32 One-Time Inspection). OPPD will include these activities in Fort Calhoun’s one-time inspection program B.3.5.

RAI 3.1.2-3 LRA Table 3.1.2, rows 3.1.2.08 and 3.1.2.11, indicates that void swelling, and reduction in fracture toughness of the reactor vessel internals flow skirt are managed by the Reactor Vessel Internals Inspection Program and Row 3.1.2.09 indicates that cracking of the reactor vessel internals flow skirt is managed by the Alloy 600 Program. The Alloy 600 Program is for piping and head penetrations and is dependent on leakage detection for detection of cracking. Identify the inspections and frequency of inspection to be performed as part of the Alloy 600 program to detect cracks in the reactor vessel internals flow skirt. Since the Reactor Vessel Internals Inspection Program indicates that a fluence, stress, and fracture mechanics analysis will be performed to determine the critical location, acceptance criteria and appropriate inspection technique,

confirm that the applicant is planning to perform these analyses for the reactor vessel internals flow skirt to manage the aging effects of void swelling and reduction in fracture toughness.

Response:

Refer to the response to RAI 3.1.1-D3 above. The flow skirt is one of those components currently included under the scopes of the Reactor Vessel Internals Inspection Program and the Alloy 600 Program. Exactly how the flow skirt is to be managed under the Alloy 600 Program is yet to be determined; however, that determination will be made before entry into the period of extended operation.

The fracture mechanics analysis committed to in Section B.2.8 of the LRA will be performed prior to the period of extended operation.

Meeting Discussion:

See RAI 3.1.1-3.

RAI 3.1.3-1 LRA Table 3.1-3, Row 03, "Bolt-Thermal Shield," credits the Inservice Inspection program for managing loss of preload in the thermal shield bolts. As stated in the justification column of 3.1.3.03, the basis for crediting ISI is that the material, environment, and aging effects are the same as for components evaluated in Volume 2, IV.B3.4-h, of the GALL report. This section of the GALL report states that GALL programs XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," XI.M14, "Loose Part Monitoring," are credited with managing aging in the components similar to the thermal shield bolts. On page B-3 of the LRA, the applicant states that a loose parts monitoring program is not credited for license renewal at FCS. Instead, the reactor vessel internals program (RVII, LRA Section B.2.8) is credited with managing aging. The RVII program states that it is consistent with GALL Program XI.M16, "PWR Vessel Internals," with an exception that no augmented inspection of bolting is scheduled. This exception refers to bolting for the reactor vessel. In addition, the staff's review of the operating experience discussed in LRA Section B.2.8 does not specifically discuss bolting for the thermal shield. In order to have reasonable assurance that the thermal shield bolting will be adequately managed during the period of extended operation, the staff requests the following information:

1. Identify plant-specific and industry operating experience with respect to cracking and loss of preload of thermal shield bolts. Identify how the proposed program for thermal shield bolts will ensure bolting integrity.

2. Chapter XI.M16, "PWR Vessel Internals" states, under the subsection discussing the detection of aging effects, "For bolted components, augmented ISI is to include other demonstrated acceptable inspection methods to detect cracks between the bolt head and the shank. Alternatively, the applicant may perform a component-specific evaluation, including a mechanical loading assessment to determine the maximum

tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criterion requires continued use of the augmented inspection methods." Will the thermal shield bolt program satisfy this inspection/analysis?"

Response:

OPPD has incorporated an augmented inspection of the thermal shield bolting or pins within the Reactor Vessel Internals Inspection Program. In 1984, a commitment was made to the NRC to perform an inspection of the FCS 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 (Letter from OPPD (R. L. Andrews) to NRC (A. C. Thadini), dated August 28, 1986, Fort Calhoun Thermal Shield Support System Inspection Deferral (LIC-86-421)). 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. The preload of 11 of the 16 lower positioning pins was also performed. Based on the measurements and an analytical evaluation of preload, 7 lower and 4 upper pins were replaced. This action reduced vibrations back to normal levels. No unacceptable vibration has been detected since 1992 and OPPD continues to monitor thermal shield vibrations using the Internals Vibration Monitoring Program. Data are acquired on a monthly basis and evaluated on a quarterly basis as described in a letter from OPPD (R. L. Andrews) to NRC (Document Control Desk), dated February 25, 1988, "Threshold Levels for the Fort Calhoun Internals Vibration Monitoring System" (LIC-88-091). This letter provided acceptance criteria and corrective action. Any unacceptable vibration will be corrected.

Meeting Discussion:

OPPD will clarify that the task of obtaining data to monitor internal vibrations is included in B.2.7, Periodic Surveillance and Preventive Maintenance program.

3.2 <u>Engineered Safety Features</u>

RAI 3.2.1-1 Table 3.2-1, Row 3.2.1.04, states in the "Discussion" column, that no FCS containment isolation valves (CIVs) and associated piping, in systems that are not addressed in this or other sections of this application were determined to be subject to the aging effect of loss of material due to microbiologically influenced corrosion (MIC). This statement is not clear. To determine whether these components are applicable to FCS and to assess the adequacy of the management of the aging effects associated with these components, please clarify what the statement means. Specifically, does the GALL report address the CIVs and associated piping at FCS? If so, provide the evaluation recommended by the GALL report and identify the associated aging

management program(s) credited for managing loss of material due to MIC in these components.

Response:

The “Discussion” column of LRA AMR Item 3.2.1.04 has a typographical error. The word “not” at the end of the second line of that entry should not be there. The discussion item should read, “No FCS containment isolation valves and associated piping in systems that are addressed in this or other sections of this application were determined to be subject to the aging effect of loss of material due to MIC.”

Meeting Discussion:

OPPD will revise the response to provide background information to support their conclusion that no FCS containment isolation valves and associated piping were determined to be subject to loss of material due MIC.

RAI 3.2.3-1 In LRA Table 3.2-3, if the terms, “safety injection tank” and “accumulator”, are used interchangeably for FCS, explain why FCS safety injection tanks (cf. Row Number 3.2.3.01) are associated with the material of stainless steel, whereas accumulators (cf. Row Number 3.2.3.02) are associated with carbon steel with stainless steel cladding, for the same kind of environment.

Response:

These component terms are not used interchangeably since they are two separate and distinct components. They have been identified separately in Table 2.3.2.1-1 of the LRA specifically to prevent such confusion. Referring to this table, the SI leakage cooler accumulators are addressed under the Component Type “Leakage Accumulators.” The safety injection tanks are addressed under the Component Type “Injection Tanks.”

The SI leakage cooler accumulators, SI-7A, -7B, -7C, and -7D, are carbon steel vessels internally clad with stainless steel. SI-7A and -7B are located in zones A5 and A8, respectively, on drawing E-23866-210-130, Sheet 2. SI-7C and -7D are located in zones A8 and A5, respectively, on drawing E-23866-210-130, Sheet 2B.

The safety injection tanks, SI-6A, -6B, -6C, and -6D, are wholly stainless steel vessels. SI-6A and -6B are located in zones C/D3 and C/D6, respectively, on Drawing E-23866-210-130, Sheet 2. SI-6C and -6D are located in zones C/D6 and C/D3, respectively, on drawing E-23866-210-130, Sheet 2B.

Meeting Discussion:

On the basis of the information provided, the staff found this response satisfactory and does not require a formal response to the RAI.

RAI 3.2.3-2 In LRA Table 3.2-3, Row 3.2.3.02, based on the review results of the GALL report, for leakage accumulators (or safety injection tanks) with leaking chemically treated borated water, the corresponding FCS AERMs should be loss of material/boric acid corrosion, instead of crack initiation and growth/stress corrosion cracking. Also, according to Volume 2, V.D1.7-a of the GALL report, the aging management program to be relied on for this aging effect should be Chapter XI.M10, "Boric Acid Corrosion", instead of Chapter XI.M2, "Water Chemistry", as required by V.D1.7-b. Explain the discrepancies.

Response:

LRA AMR Item 3.2.3.02 applies to the stainless steel liner of the SI leakage cooler accumulators, as indicated in the "FCS Material" column of that entry. Refer to Table 2.3.2.1-1 of the LRA. For the "Leakage Accumulator" Component Type in that table, the reader is provided with links to LRA AMR Item 3.2.3.02, for the accumulator internals, and to LRA AMR Item 3.2.1.11 for the accumulator externals. For the "Injection Tank" Component Type, the reader is provided links to LRA AMR Items 3.2.2.04 and 3.2.3.01.

Meeting Discussion:

On the basis of the information provided, the staff found this response satisfactory and does not require a formal response to the RAI.

3.6 Electrical and Instrumentation and Controls

RAI 3.6-1 For inaccessible medium-voltage (2 kV to 15 kV) cables (e.g., installed in conduit or direct buried) not subject to 10 CFR 50.49 EQ requirements, LRA Table 3.6-1, Row 3.6.1.04, states that modifications were made to the duct banks to preclude moisture intrusion; therefore there is no aging effect requiring management. However, it is not clear to the staff what actions will be taken to assure that the modifications made to prevent inaccessible non-EQ medium-voltage cables from being exposed to significant moisture will be maintained intact during the period of extended operation. Therefore, for these non-EQ cables that are within the scope of license renewal, provide a description of the program that will assure that the modifications are maintained intact to prevent intrusion of water into the duct banks. In addition, provide a description of the AMP that will be relied upon for accessible and inaccessible medium-voltage cables installed in conduits, cable trenches, cable troughs, underground vaults or direct buried installations.

Response:

See Response to RAI 2.5-D-1 and B.3.4-D-1

Meeting Discussion:

OPPD agreed to remove the trend analysis discussion from the draft RAI response.

4.2 Reactor Vessel Neutron Embrittlement

RAI 4.2-2 Section 4.2.4 of the application indicates the preliminary calculations have shown that the vessel beltline Charpy upper-shelf energy (USE) for the limiting weld will be approximately 54.5 ft-lb based on position 1.2 of RG 1.99. The applicant indicates that this analysis will be finalized and formally revised to reflect that it bounds the minimum approved fluence value at the end of plant life. In order for the staff to complete its review of this time-limited aging analysis (TLAA) issue, the applicant must submit the results of its analysis based on the projected neutron fluence at the end of the period of extended operation. Therefore, the applicant is to provide the following information:

a) The projected peak neutron fluence at a depth of 1/4 T (thickness) for each beltline material at the end of the period of extended operation

b) The method (either position 1.2 or position 2.2 of RG 1.99, Revision 2) of determining the decrease in Charpy USE for each beltline material

c) The unirradiated Charpy USE for each beltline material

d) The amount of copper for each beltline material and references for all surveillance data

e) Based on the information in Items (a) through (d), the projected Charpy USE for each beltline material at the end of the period of extended operation

f) The impact of surveillance data on the projected Charpy USE

Response:

a)

Plate/Weld Number	Plate/Weld Heat No.	Fluence at 1/4T ($n/cm^2 \times 10^{19}$
D 4802-1	C 2585-3	2.28
D 4802-2	A 1768-1	2.28
D4802-3	A 1768-2	2.28
D4812-1	C3213-2	2.28
D4812-2	C3143-2	2.28
D4812-3	03143-3	2.28
2-410	51989	1.62
3-410	13253	1.62
3-410	27204	1.62
3-410	13253/12008	1.62
3-410	12008/27204	1.62
9-410	20291	2.28

b) Position 2.2 of RG 1.99, Revision 2, was used in determining the decrease in Charpy USE for each beltline material.

c)

PLATE #	HEAT #	Cu CONTENT (%)	INITIAL USE (FT-LB)
D4802-1	C 2585-3	0.12	75.4
D4802-2	A 1768-1	0.10	120
D4802-3	A 1768-2	0.11	77.4
D4812-1	C 3213-2	0.12	88.5
D4812-2	C 3143-2	0.10	87
D4812-3	C 3143-3	0.10	89.7

Weld Number	Wire Heat Number	Initial USE (ft~lb)*	Flux Type	Flux Lot No.	Cu Content (%)
2-410	51989	84	Linde 124	3687	0.170
3-410	12008 and 13253 (Tandem)	97	Linde 1092	3774	0.210
3-410	27204 (Tandem)	94	Linde 1092	3774	0.203
3-410	13253 (Tandem)	110	Linde 1092	3774	0.221
3-410	12008 and 27204 (Tandem)	97.8	Linde 1092	3774	0.219
9~410	20291	105	Linde 1092	3833	0.216

d) See response to Item c). References for surveillance data are included in (1) Letter from OPPD (W. G. Gates) to USNRC (Document Control Desk), Dated July 6,1992. (LIC 92-203R) “Response to NRC Generic Letter (GL) 92-01, Revision 1: Reactor Vessel Structural Integrity,” and (2) CEOG Report CEN-636, Revision 2, Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials, Westinghouse Electric CE Nuclear Power, Final Report dated July 19, 2000.

e)

Plate/Weld Number	Plate/Weld Heat No.	Fluence at1/4 t (n/cm^2)	Cu (%)	Initial USE (ft-lb)	Position 2.2 Capsule Modified % USE Decrease	Predicted Irradiated USE from Position 2.2(ft-lb)
D 4802-1	C 2585-3	$2.28x10^{19}$	0.120	75.4	33.5	50.1
D4802-2	A 1768-1	$2.28x10^{19}$	0.100	120	30.8	83.0
D4802-3	A 1768-2	$2.28x10^{19}$	0.110	77.4	32.2	52.5
D4812-1	C 3213-2	$2.28x10^{19}$	0.120	86.5	33.5	57.5
D4812-2	C 3143-2	$2.28x10^{19}$	0.100	87	30.8	60.2
D4812-3	C 3143-3	$2.28x10^{19}$	0.100	89.7	30.8	62.1
2-410	51989	$1.62x10^{19}$	0.170	84	37.0	52.9
3-410	13253	$1.62x10^{19}$	0.221	110	42.2	63.6
3-410	27204	$1.62x10^{19}$	0.203	94	40.1	56.3
3-410	13253/12008	$1.62x10^{19}$	0.210	97	41.2	57.0
3-410	12008/27204	$1.62x10^{19}$	0.219	97.8	42.2	56.5
9-410	20291	$2.28x10^{19}$	0.216	105	45.4	57.3

f)

CAPSULE NUMBER AND MATERIAL	FLUENCE	INITIAL USE (FT-LB)	IRRADIATED USE (FT~LB)	CU CONTENT (%)	PREDICTED DECREASE IN USE (FT-LB)	PREDICTED IRRADIATED USE (FT-LB)	MEASURED DECREASE IN USE (%)
W-265 (TL) Plate	0.77×10^{19}	120	93	0.10	18.0	98.4	22.5
W-225 (LT) Plate	0.55×10^{19}	141	122	0.10	16.5	117.7	13.5
W-265 (TL) Plate	0.77×10^{19}	141	109	0.10	18.0	115.6	22.7
W-275 (TL) Plate	1.28×10^{19}	120	88	0.10	20.0	96.0	26.7
W-275 (TL) Plate	1.28×10^{19}	141	107	0.10	20.0	112.8	24.1
W-225 Weld	0.55×10^{19}	104	65	0.35	39.0	63.4	37.5
W-275 Weld	1.28×10^{19}	103.5	60	0.35	41.0	61.4	43.3
W-265 Weld	0.77×10^{19}	104	59	0.35	44.0	57.96	42.0
Mihama 1 (1st monitoring)	0.6×10^{19}	97.8	68.4	0.19	29.5	68.95	30.1
Mihama 1 (2nd monitoring)	1.2×10^{19}	97.8	61.0	0.19	35.0.	63.6	37.6
Mihama 1 (3rd monitoring)	2.1×10^{19}	97.8	61.0	0.19	39.5	59.2	37.6

Meeting Discussion:

OPPD pointed out a confusing sentence in the GALL report. OPPD will revise response to address same questions as outlined in RAI B.1.7-D1 Reactor Vessel Integrity Program.

4.7.1 Reactor Coolant Pump Flywheel Fatigue

RAI 4.7.1-1 Two crack growth analyses are referenced in Sections 4.7.1.1 and 4.7.1.2. One is described as Reference 4.7-1 and the other is described as an analysis performed by ABB. 10 CFR 54.21(c)(i) and (ii) discuss analyses required as part of the TLAA. In order to confirm that the applicant has satisfied the regulatory requirements, the staff needs to review these analyses. Please provide the analyses and provide any references that indicate that they have been previously reviewed by the NRC.

Response:

The TLAA discussion of the General Electric RCP motor flywheels in Section 4.7.1.1 consists primarily of excerpts from Section 4.3.5 of the FCS Updated Safety Analysis Report (USAR). The "Reference 4.7-1" notation is apparently a transcription error, and is noted as "Reference 4-4" in the source USAR paragraph. USAR Reference 4-4 is a fracture mechanics methodology used in the postulated failure analysis described in USAR Section 4.3.5 (see Note 6 to Table 4.3-4 for the USAR wording).

To satisfy 10 CFR 54.21(c)(i) and (ii) requirements, OPPD in the application provided a brief description of each crack growth analysis and its conclusion, as well as determinations that the results would remain valid through the period of extended operation. The supporting or referenced analyses are in the USAR or are available on site for NRC inspection. The USAR analysis of the General Electric RCP motor flywheels appears to date back to the FSAR approved during intial plant licensing. The proprietary analysis referenced in the USAR for the ABB motor flywheel has not been reviewed by the NRC.

Meeting Discussion:

OPPD agreed to clarify response to include flywheel proprietary information.

B.1.7 Reactor Vessel Integrity Program

RAI B.1.7-1 In a License Amendment dated August 3, 2000 and letters dated November 17, 2000 and February 14, 2001, the licensee provided RT_{PTS} analyses for the materials in the FCS reactor vessel. The August 3, 2000 letter contains Report CEN-636, “Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials.” Table 10 in CEN-636, Revision 2, provides the chemistry factor and the predicted RT_{PTS} value through 2033 for each plate and weld in the FCS reactor vessel beltline. Many of the materials RT_{PTS} values are dependent upon surveillance data which could effect their

RT_{PTS} value. In addition, one weld is projected to be only 2 ^{o}F below the PTS screening limit and one weld is projected to be 15 ^{o}F below the PTS screening limit at the end of the period of extended operation. To determine whether the Reactor Vessel Integrity Program will adequately monitor neutron irradiation embrittlement, provide the following information:

a. Confirm that the RT_{PTS} value identified in Table 10 of CEN-636 is applicable to the end of the period of extended operation for the FCS.

Response:

The RT_{PTS} value identified in Table 10 of CEN-636 is applicable to the end of the period of extended operation for FCS.

b. For each material in Table 10 of CEN-636 identify the projected neutron fluence at the end of the period of extended operation and the neutron flux assumed for future core loadings.

Response:

For the 2-410 and 3-410 axial welds the fluence used was 2.43E+19 n/cm^2. This value conservatively corresponds to the end of the fuel Cycle 41 (March 2034) that FCS would be operating during 2033. In this projection a fast neutron flux equal to that of Cycle 15 was used for all future cycles, i.e., the fluence accumulation rate is 4.0E+17 n/cm^2/EFPY.

For all plate material and the 9-410 circumferential weld, the maximum circumferential fluence occurs at 45^{o}. A bounding value at 48 EFPY, which would not be reached until after the period of extended operation, is 3.50E+19 n/cm^2. For the materials below the bounding values of RT_{PTS} would be:

Plate/Weld	CF(^{o}F)	RT_{PTS}	Screening Limit	Margin
D4802-1	82.2	143	270	127
D4802-2	72.0	130	270	140
D4802-3	73.1	131	270	139
D4812-1	83	144	270	126
D4812-2	65	120	270	150
D4812-3	65	120	270	150
9-410	188.41	259	300	41

c. For each chemistry factor in Table 10 of CEN 636 that was calculated using surveillance material identify the source of the surveillance material.

Response:

All cases where surveillance data were used were for the numerous combinations of the 3-410 tandem arc weld (fabricated with weld wire heats

12008, 13253, and 27204) and for the FCS surveillance plate, D4802-2. The weld materials data used were:

<u>12008/13253:</u>
-Mihama 1 (3 capsules)

<u>27204:</u>
-Diablo Canyon 1 (2 capsules)
-Palisades Supplemental Capsule using FCS nozzle drop-out (1 capsule)

<u>13253:</u>
-DC Cook 1 (4 capsules)
-Salem 2 (3 capsules)

d. Explain how the Reactor Vessel Integrity Program will monitor future core loadings to ensure that no beltline materials will exceed the PTS screening limit in 10 CFR 50.61.

Response:

Compliance with 10 CFR 50.61 is monitored as part of the Program Basis Document for Reactor Vessel Integrity. This program is administered by the FCS Design Engineering-Nuclear Engineering Department. The Nuclear Engineering Department also performs core reload analyses in-house, including core design. During core loading development core patterns are quantitatively evaluated to ensure that neutron flux to the limiting 60°/300° 3-410 welds is maintained approximately the same as that of Cycle 15, which formed the basis of the fluence analysis. This is done by summing the peripheral fuel assembly relative power densities multiplied by weighting factors derived from the fluence analysis adjoint flux solution. Thus values from a new fuel cycle can be compared to that of Cycle 15 to determine if there has been a net increase or decrease, with a goal of having a time average value the same as Cycle 15. Periodic updates of the fluence analysis are planned. RT_{PTS} is also tracked on an ongoing basis.

e. Identify how the Reactor Vessel Integrity Program will monitor future surveillance capsule data from FCS and other facilities to ensure that no beltline materials will exceed the PTS screening limit in 10 CFR 50.61 or the Charpy upper-shelf energy screening criteria in Appendix G, 10 CFR Part 50.

Response:

Section 4.2 of the Program Basis Document for Reactor Vessel Integrity includes a schedule for obtaining applicable surveillance data from other plants to ensure that the chemistry factors of CEN-636 remain applicable. The FCS surveillance program was recently revised and approved by the NRC staff (Letter from NRC (S. Dembeck) to OPPD (R. T. Ridenoure) dated May 2, 2002, "Fort Calhoun

Station, Unit No. 1 - Reactor Vessel Surveillance Capsule Removal Schedule Change (TAC No. MB3422)"). The next FCS surveillance capsule to be removed is W-275S at 33.6 EFPY. This supplemental capsule contains the limiting weld material 12008/13253 (Maine Yankee nozzle drop-out) as well as 27204 (FCS nozzle drop-out). In addition, if any major changes in plant operation do affect PTS or USE projections, reanalysis will be performed to ensure compliance with 10 CFR 50.61 or 10 CFR 50, Appendix G, as applicable.

Meeting Discussion:

OPPD agreed to revise the RAI response to address (1) how the fluence was calculated (2) did the methodology adhere to the guidance of Regulatory Guide 1.190 (3) where is the CF and the FF for limiting welds 2-410 and 3-410 (4) 48 EFPY for the extended license corresponds to an 80% load factor; if PWRs are operating at a 90 to 95% load factor and Fort Calhoun uses the same calculation, then Fort Calhoun is likely to overshoot the margin and violate the screening criteria defined in 10 CFR 50.61. How will OPPD address this issues?

B.2.9 Steam Generator Program

RAI B.2.9-2 The applicant stated that the steam generator program is consistent with XI.M19, “Steam Generator Tube Integrity,“ in the GALL report, with the exception of two enhancements. The applicant stated that its steam generator program also includes aging management activities to address plant-specific AMP requirements identified in Tables 3.1-1 and 3.1-2. However, the GALL report states that the scope of XI.M19 is specific to steam generator tubes. Therefore, please respond to the following related questions:

1. Table 3.1-1, Row 3.1.1.02, “Steam Generator Shell Assembly”, states that the aging effect for this component (i.e., loss of material due to pitting and crevice corrosion) is managed, in part, by the steam generator program (B.2.9). It is not clear to the staff how the steam generator program manages this aging effect. In addition, because the GALL report states that the scope of XI.M19 is specific to steam generator tubes, provide details for the following attributes for this component: Preventive Actions; Parameters Monitored/Inspected; Detection of Aging Effects; Monitoring and Trending; and Acceptance Criteria. Ensure that the discussion identifies how the steam generator program manages this aging effect (e.g., identify the part of this component that is managed by the steam generator program and how it is managed by the steam generator program).

2. Table 3.1-1, Row 3.1.1.15, “(Alloy 600) Steam generator tubes, repair sleeves, and plugs,” states that the aging effect for these components are managed, in part, by the steam generator program (B.2.9). The GALL report states that the scope of XI.M19 is specific to steam generator tubes, therefore, provide details for the following attributes for the repair sleeves and plugs: Preventive Actions; Parameters Monitored/Inspected; Detection of Aging Effects; Monitoring and Trending; and Acceptance Criteria.

3. Table 3.1-1, Row 3.1.1.16, “Tube support lattice bars made of carbon steel,” states that the aging effect for this component is managed by the steam generator program (B.2.9). The GALL report states that the scope of XI.M19 is specific to steam generator tubes, therefore, provide details for the following attributes for this component: Preventive Actions; Parameters Monitored/Inspected; Detection of Aging Effects; Monitoring and Trending; and Acceptance Criteria.

4. Table 3.1-1, Row 3.1.1.17, “Carbon steel tube support plate,” states that the aging effect for this component is managed by the steam generator program (B.2.9). The GALL report states that the scope of XI.M19 is specific to steam generator tubes, therefore, provide details for the following attributes for this component: Preventive Actions; Parameters Monitored/Inspected; Detection of Aging Effects; Monitoring and Trending; and Acceptance Criteria.

5. Table 3.1-2, Row 3.1.2.06, “Secondary side of the tubesheet, steam generator feedwater, steam and instrument nozzles, and feedwater nozzle safe ends,” states that the aging effect for these components is managed by the steam generator program (B.2.9). The GALL report states that the scope of XI.M19 is specific to steam generator tubes, therefore, provide details for the following attributes for this component: Preventive Actions; Parameters Monitored/Inspected; Detection of Aging Effects; Monitoring and Trending; and Acceptance Criteria.

6. Table 3.1-2, Row 3.1.2.07, “Steam generator tube plugs,” states that the aging effect for this component is managed by the steam generator program (B.2.9). The GALL report states that the scope of XI.M19 is specific to steam generator tubes, therefore, provide details for the following attributes for this component: Preventive Actions; Parameters Monitored/Inspected; Detection of Aging Effects; Monitoring and Trending; and Acceptance Criteria.

7. Table 3.1.2, Row 3.1.2.14, “Steam generator steam nozzle safe end, steam generator feed ring,” states that the aging effect for these components is managed by the steam generator program (B.2.9). The GALL report states that the scope of XI.M19 is specific to steam generator tubes, therefore, provide details for the following attributes for this component: Preventive Actions; Parameters Monitored/Inspected; Detection of Aging Effects; Monitoring and Trending; and Acceptance Criteria.

Meeting Discussion:

During a telecon held on September 25, 2002, the staff asked that OPPD address each component identified in the RAI for each program element identified in the RAI (e.g., preventive actions, parameters monitored/inspected, etc.). OPPD subsequently revised the response and provided it to the staff for review. During this telecon, the staff suggested that OPPD follow the guidance provided in Branch Technical Position (BTP) RLSB-1, in NUREG-1800. OPPD

questioned the level of detail that the staff expected. The staff indicated that it would review previous applications to ascertain what level of detail the staff has accepted for AMP information. The staff and OPPD also discussed whether the requested information could be reviewed during the AMR inspection. The staff indicated that it would investigate this possibility (subsequently, the staff determined that following the guidance in the BTP would yield the appropriate level of detail, and that review of the information during the AMR inspection would not be appropriate).

B.3.1 Alloy 600 Program

RAI B.3.1-1 Background

In NRC Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” the staff summarized circumferential cracking that had occurred in control rod drive mechanism (CRDM) nozzle J-groove welds at the Oconee Unit 1 and Unit 3 nuclear stations, and emphasized the need for licensees who own pressurized water reactors (PWRs) to perform bare-surface visual examinations of their reactor vessel heads. In NRC Bulletin 2002-01, “Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,” the staff summarized excess boric acid wastage that had occurred in the Davis Besse reactor vessel head as a result of leaking CRDM nozzles and excessive boric acid buildup on the head. The Davis Besse event indicates that boric acid wastage inspection programs implemented in accordance with staff requests in NRC Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,” may not, by themselves, be capable of effectively monitoring for and controlling leakage past CRDM or other vessel head penetration nozzles, or boric acid-induced wastage of the low-alloy steel reactor vessel heads that the penetration nozzles are welded to. Based on the Davis Besse event, in NRC Bulletin 2002-02, “Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,” the staff emphasized the need to perform augmented inspections of CRDM and other vessel head penetrations beyond the bare surface visual examinations of reactor vessel heads that were recommended in NRC Bulletin 2002-01.

In addition, other prominent PWSCC cracking events have occurred since January 2000. On October 7, 2000, during a containment inspection of the V. C. Summer nuclear plant after entering a refueling outage, the licensee identified a large quantity of boron on the floor and protruding from the air boot around the "A" loop RCS hot leg pipe. On October 12, 2000, a liquid penetrant test (PT) performed by the licensee indicated the existence of a 4-inch long circumferential indication in the first Alloy 82/182 weld between the reactor vessel nozzle and the "A" loop hot leg piping, approximately 3 feet from the reactor vessel. Additional non-destructive testing of the "A" loop hot leg piping did not confirm a flaw at the location of the circumferential indication. These tests identified, at a different location, an axial crack-like indication, approximately 2.7 inches long, and located approximately 9 degrees counterclockwise from top dead center of the weld. This indication extends from approximately the centerline of the weld toward the reactor nozzle. Visual examination from the

outside diameter of the pipe identified a small "weephole" in the center of the weld at approximately the same circumferential location as the axial indication. On this basis, to ensure that the proposed AMP will adequately manage the aging effects associated with this industry experience, the staff requests the following information:

The program elements addressed by this portion of the RAI are [Detection of Aging Effects], [Monitoring and Trending], [Acceptance Criteria] and [Corrective Actions]. On the basis of the issues raised in Bulletins 2001-01, 2002-01, and 2002-02, the staff is currently determining, with the U.S. nuclear power industry, what the requirements should be for inspections of vessel head penetration (VHP) nozzles in U.S. PWRs. The scope of any actions and/or activities agreed upon between the NRC and the industry for resolution of this issue will need to address acceptable criteria for the monitoring, detection, evaluation, and correction of potential cracking that occurs in the VHP nozzles of U.S. PWRs. Since this issue might not be resolved prior to issuance of the renewed operating license for FCS, the staff requests that the applicant commit to implement, as part of the Alloy 600 Program, any actions that are agreed upon between the NRC, Nuclear Energy Intitute (NEI), Materials Research Program (MRP), and the nuclear power industry, for the inspection, detection, evaluation (including the establishment of acceptable acceptance criteria for the VHP nozzle inspection techniques that are agreed on between the staff and the industry), and correction of cracking that may occur in VHP nozzles of U.S. PWRs, and specifically as the actions relate to ensuring the integrity of VHP nozzles in the FCS upper RV head during the period of extended operation.

The program elements addressed by this portion of the RAI are [Scope], [Detection of Aging Effects], [Monitoring and Trending], and [Operating Experience]. The staff requests the applicant to identify the Alloy 600 and Alloy 82/182 locations in the FCS pressurizer, steam generators, and RCS piping. With respect to these locations, the staff requests that the applicant identify those locations that are most likely to develop PWSCC and those locations in which the applicant has already detected and reported leakage and/or indications of PWSCC. If leakage and/or PWSCC has been detected and reported in any of the Alloy 600 or Alloy 82/182 locations in the FCS pressurizer, steam generators, or RCS hot-leg piping, indicate whether applicable Section XI Code repairs have been made to the flawed areas or whether relief has been granted to use alternative repair or replacement methods for repairing the flawed areas (NOTE: if relief has been granted pursuant to 10 CFR 50.55a, and alternative repair/replacements have been implemented at FCS for these nozzles, appropriate TLAAs must be submitted for the alternative repair or replacement methods if long-term installation is to be implemented over the period of extended operation without the granting of multiple temporary reliefs by the NRC under the requirements of 10 CFR 50.55a). Additionally, the staff requests the applicant to describe the actions it plans to take for maintaining the integrity of these Alloy 600 and Alloy 82/182 locations over the period of extended operation for FCS. Include in your response a discussion of specific actions taken, if any, to resolve the V.C. Summer RCS hot leg cracking issue as it pertains to maintaining the structural integrity of RCS hot leg piping at FCS.

Pursuant to 10 CFR 54.21(c)(3), the next revision to the FSAR Supplement description for the Alloy 600 Program must reflect the applicant’s response to GL 97-01, “Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations,” and NRC Bulletins 2001-01, 2002-02, and 2002-02. When submitted, the staff also requests that the applicant incorporate its responses to Parts 1 and 2 of this RAI into applicant’s next revision to the FSAR Supplement description for the Alloy 600 Program, since the responses to the RAIs will provide clarifying content as to how the Alloy 600 Program will be sufficient to manage cracking in ASME Code Class 1 components made from Alloy 600 or Alloy 82/182 materials (i.e., Inconel alloy materials).

Response:

The FCS Alloy 600 Program currently includes a requirement to monitor industry operating experience and implement program enhancements as necessary. This issue of cracking of Alloy 600 and Alloy 82/182 material is being addressed as a current licensing basis issue.

OPPD will implement any required actions in the Alloy 600 Program necessary to maintain the current license.

The following locations of Alloy 82/182 have been identified in the Alloy 600 Program Basis Document for the pressurizer, steam generators, and RCS piping:

Pressurizer - internal attachments to lower shell, bottom head

Pressurizer top head - temperature nozzle, relief valve assembly (nozzle to spool piece and spool piece to safe end), level nozzle assembly (nozzle to spool piece and spool piece to safe end), safety nozzle to flange, spray nozzle assembly (nozzle to spool piece and spool piece to safe end), temperature nozzle safe end, spray nozzle radius and flat clad, and surge nozzle and spray nozzle clad

Pressurizer lower shell - temperature nozzle, cladding (12" wide band)

Pressurizer bottom head - heater sleeves, heater to heater sleeves, surge nozzle to safe end, temperature nozzle to safe end, level nozzle assembly (nozzle to spool piece and spool piece to safe end), and inner diameter cladding

Steam generator - primary head divider plate lane cladding, primary head to tubesheet girth backclad, tubesheet cladding, drain tube bracket to clad overlay, divider bar to head and tubesheet, and clamp rings and pins at primary head nozzle openings

Steam generator primary head - primary nozzle safe ends to nozzles

The most susceptible locations for PWSCC are the welds on the pressurizer heater sleeves and pressurizer temperature nozzles.

In 2001, FCS experienced a leak on the pressurizer lower shell temperature nozzle, TE-108. A code weld repair was made to the TE-108 nozzle, and a mechanical device was placed on pressurizer nozzle TE-107 as a precautionary measure. The mechanical seal device was removed during the next refueling outage when eddy current measurements verified no cracking was occurring at the nozzle. Review of historical information on this weld identified that a weld repair had been made at this location during original fabrication as a result of the nozzle being bent after the initial hydro test.

The root cause analysis of the V. C. Summer event indicated extensive rework of the affected nozzle, which resulted in changes to the as-designed weld configuration, was an important contributor to the failure. As a result, FCS performed a detailed review of FCS Alloy 600 and Alloy 82/182 component fabrication records. FCS is less susceptible to the cracking that occurred at V. C. Summer based on the following :

All the nozzle to safe end welds at FCS were shop welds, in contrast to the affected weld at V. C. Summer, which was a field weld. A review of FCS fabrication records for nozzle to safe end welds did not reveal any evidence of significant weld repairs.

FCS nozzle to safe end welds were heat treated following repairs. This was not performed on the affected V. C. Summer weld.

The FCS nozzle welds with more rework were inlet nozzles whose lower temperature would result in less driving force for PWSCC.

The reactor coolant system (RCS) inspection procedure used by FCS following opening, repair, or modification of the RCS system has been enhanced to specifically identify Alloy 600 and Alloy 82/182 components, and increase inspector awareness for evidence of potential leakage. Additional details and analysis results are documented in the FCS Alloy 600 Program Basis Document.

An assessment of Alloy 600 and Alloy 82/182 components has been performed and incorporated into the Alloy 600 Program Basis Document. The assessment included evaluation for history, susceptibility, safety considerations, inspection, repair, and mitigation, and provided conclusions and recommendations to address the specified components. The recommendations include items such as increased inspections (UT), repair, replacement, and more detailed assessments. These recommendations will be evaluated as part of the Alloy 600 program and implemented as necessary to ensure the reliability of the Alloy 600 and Alloy 82/182 components.

Response:

The level of detail provided in response to Part 2 of this RAI is not consistent with the overall level of detail currently in the FCS USAR; thus, not all of the response will be included in the USAR supplements. However, OPPD will incorporate appropriate information reflecting the above OPPD responses, and the responses to GL 97-01 and NRC Bulletins 2001-01, 2002-01, and 2002-02.

Meeting Discussion:

NRC is currently reviewing OPPD's response. OPPD agreed to refer to USAR, letters, bulletin for responses versus long discussion response.

B.3.2 Buried Surfaces External Corrosion Program

RAI B.3.2-2 The detection of aging effects in buried components is plant-specific and depends on plant operating experience as well as industry operating experience. Therefore, the staff must further evaluate the applicant's operating experience and proposed inspection frequency. The staff requests the licensee to expand the discussion of this AMP to include the inspection frequency and the applicable industry operating experience.

Response:

Buried piping and tanks will be inspected whenever they are excavated for maintenance. PM tasks to defuel, clean, and inspect the emergency diesel and auxiliary boiler diesel fuel tanks are performed on a 9-year frequency. An inspection of the emergency diesel and auxiliary boiler fuel tanks was performed in 1995. An ultrasonic test (UT) of the emergency diesel fuel oil tank was performed with no indication of degradation identified. The next inspection is scheduled to be performed in 2004, to include a UT and excavation of portions of the tanks surfaces. Recent excavations of buried piping have shown the pipe coatings to be well maintained with no evidence of degradation.

As part of the development of this new program, a Program Basis Document will be developed which will collect inspection results of buried piping inspections and monitor for adverse trends.

Meeting Discussion:

OPPD will clarify how UT is used on diesel fuel tanks and add a detailed history and description of piping excavations.

B.3.3 General Corrosion of External Surfaces Program

RAI B.3.3-2 In its description of the monitored or inspected parameters, the applicant describes the methods that will be employed to detect signs of external corrosion, and conditions that could result in external corrosion. Although fluid leakage is identified as a monitored parameter, the staff believes that other parameters, such as tank wall thickness, cracked sealant, or degraded coatings, are important to detect degraded surface conditions. Therefore, the staff requests the applicant to describe parameters, besides fluid leakage, that detect degradation of surface conditions on components within the scope of this program, and provide justification why these parameters need not be included in this aging management program to manage aging of components within the program scope.

Response:

Section (3) "Parameters Monitored or Inspected" of B.3.3 states, "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."

Fluid leakage was identified only as an example of a condition which could lead to component degradation if not corrected and that would be identified from the program walkdowns. The program includes monitoring of components and their external coatings for evidence of cracking, checking, blisters, rusting, pinholes, abrasions, delamination, and significant substrate defects (such as corrosion pits).

Meeting Discussion:

OPPD will revise to emphasize that fluid leakage is an indicator of change in environment and triggers a corrective action.

RAI B.3.3-5 The applicant states that plant procedures provide criteria for determining the acceptability of inspected components. In order to determine whether the acceptance criteria is adequate to ensure that appropriate corrective actions are taken upon the discovery of aging, the staff needs to understand the basis for the acceptance criteria. Therefore, the staff requests the applicant to discuss the NRC or industry guidance and operating experience used to establish the acceptance criteria. Does the criteria incorporate 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," Information Notice 86-99, "Degradation of Steel Containments," or Regulatory Guide 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants"?

Response:

Guidance from Regulatory Guide 1.54 is incorporated in the containment coatings inspection per Section 5.2.5 of the FCS USAR.

Degraded protective coatings discussed in Generic Letter 98-06 are a precursor of corrosion, but do not have any direct effect on the ability of the coated components to perform their intended functions. For System Engineer and Operator walk downs, acceptance criteria for when to initiate a maintenance work order or corrective action document is based on engineering judgment and the Operator’s experience. Initiation of a corrective action document is based on procedural guidance to identify damage or degradation that adversely affects the functional capability of a structure, system, or component. Procedural guidance is being enhanced as part of the implementation of this new program. This guidance will include acceptance criteria that a visual indication of loss of material or cracking identified by the accountable Operator or Engineer will not necessarily lead to an unacceptable component. Unacceptable components are those which are damaged or degraded such that they are not capable of performing their intended function, or if degradation were allowed to continue

uncorrected until the next normally scheduled inspection the component would not meet its design basis.

Meeting Discussion:

OPPD will revise guidance procedures to include more details (i.e. visual indication) with emphasis on gross (materials) material degradation. Explain how ventilation system examines sealants and gaskets.

B.3.3-D6 The General Corrosion of External Surfaces Program as described in LRA Section B.3.3 of the LRA is credited for managing loss of material and cracking. The application states that these aging effects can be detected by visual observation and inspection of external surfaces, including evidence of leaking fluid, for certain components that are not routinely accessible. The staff believes that inspection for evidence of leaking fluids also provides indirect monitoring of certain components that are not routinely accessible. The presence of fluid leakage from a component, however, would indicate that the component may not perform its intended function as a pressure boundary. Therefore, in order to determine whether this program will adequately manage the aging effects of inaccessible components, the staff requests the applicant to clarify whether the scope of systems listed in LRA Section B.3.3 includes components that are not routinely accessible and which rely on the indirect monitoring of fluid leakage. In addition, the applicant is requested to discuss the operating history of these components to demonstrate that the applicable aging effects will be adequately managed prior to the loss of their intended functions.

Response:

The scope of auxiliary systems listed in LRA Section B.3.3 excludes components that are not routinely accessible and may rely on the indirect monitoring of fluid leakage to manage aging effects. Aging management activities on inaccessible components, such as ultrasonic testing of the buried emergency diesel fuel oil tank, are incorporated into other plant programs (i.e., Buried Surfaces External Corrosion Program, Diesel Fuel Monitoring and Storage Program).

Meeting Discussion:

OPPD will revamp RAI response to describe non-accessible components are included in AMPs.

/***RA***/

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Meeting Discussion:

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/***RA***/
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