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September 9, 1999

MEMORANDUM TO: Oconee Docket Files (50-269, 50-270, and 50-287)

FROM: Joseph Sebrosky, Project Manager
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Joseph M. Sebrosky

SUBJECT: PRELIMINARY RESPONSES TO OPEN ITEMS AND CONFIRMATORY ITEMS FOR SAFETY EVALUATION REPORT ON THE DUKE ENERGY CORPORATION (DUKE) LICENSE RENEWAL APPLICATION FOR OCONEE NUCLEAR STATION UNITS 1, 2, AND 3.

Enclosure 1 to this memorandum contains preliminary responses from Duke to open and confirmatory items associated with the safety evaluation report for the Oconee license renewal application. The responses were sent to the staff in order to facilitate discussions and resolution of the open and confirmatory items. Duke will submit all its responses to the open and confirmatory items formally by the Oconee milestone date of October 15, 1999.

Docket Nos.50–269, 50-270, and 50-287

Enclosure: As stated

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SER Open Item 2.2.3-1
Since the RCW system is relied upon to supply cooling water to the SFP cooling system coolers to maintain the bulk SFP coolant temperature below the SFP design limits and below assumptions for the fuel handling accident analysis described in Section 15.11.2.1 of the UFSAR, the staff believes that this system should be included within the scope of license renewal based on the criteria of 10 CFR 54.4 (a)(1)(iii) and its components subject to an AMR in accordance with the requirements of 10 CFR 54.21.

Duke Response to SER Open Item 2.2.3-1
The fuel handling accident analysis assumes that spent fuel cooling, and thus the Recirculated Cooling Water System, is not functional during or following the event. The safety analysis acceptance criteria of the fuel handling accident in Section 15.11 of the UFSAR demonstrates that resultant consequences of the accident remain within 10 CFR 100 guidelines. The initial spent fuel pool temperature established by normal operating procedures ensures that the safety analysis does not require cooling to mitigate the accident. License renewal scoping criteria of §54.4(a)(1)(iii) requires that systems and components required to remain functional during and following design basis events to prevent or mitigate the consequences that could result in potential offsite exposure comparable to 10 CFR Part 100 guidelines be within scope. Since spent fuel cooling is not required to remain functional during or following the fuel handling accident to prevent or mitigate the consequences that could result in potential offsite exposure comparable to 10 CFR Part 100 guidelines, the systems and components required to fulfill the function of spent fuel cooling, including the Recirculated Cooling Water System, are not within the scope of license renewal.

SER Open Item 2.2.3.4.3.2.1-1
Also in the May 10,1999, letter, the applicant provides reasons why the Chilled Water System (CWS) (which supports the cooling function for the CRPFS) is not included within the scope of license renewal. The applicant states that for certain design-basis events, the CRPFS maintains a positive pressure in the control room and that air conditioning is not required. The applicant states that failure of the CWS does not prevent the CRFPS from maintaining a positive pressure in the control room for accident conditions and is not classified Oconee Piping Class D for seismic II/I concerns. Further, the applicant stated that the CRFPS is credited with maintaining a suitable environment in the control room during a fire event and providing for smoke removal from the control room, neither of which require air conditioning supported by the CWS system. The applicant also noted that the CRPFS and the supporting CWS do not perform an intended function in support of any other regulated event listed in 10 CFR 54.4(a)(3). The applicant concludes from this evaluation that the CWS is not within the scope of license renewal. The staff does not agree with this conclusion. It appears to the staff that the CWS is needed at ONS in order to assure the capability to shutdown the reactor and maintain it in a shutdown condition. The applicant should identify where in the current licensing basis the loss of the CWS has been addressed, and clarify why the CWS is not within the scope of license renewal and subject to an aging management review.

Duke Response to SER Open Item 2.2.3.4.3.2.1-1
The requirement to have an installed Chilled Water System that can withstand a single active failure is addressed in UFSAR 3.11.4. The Chilled Water system has also recently been added to the current licensing basis of Oconee via Improved Technical Specification (ITS)

implementation. Improved Technical Specifications, which were implemented March 27, 1999, includes a new requirement in section 3.7.16 to have two trains of Chilled Water operable. The Chilled Water System is used to maintain control room and control area temperatures within ITS prescribed limits. While the Chilled Water System is addressed in the Oconee current licensing basis, a Loss of Chilled Water System event is not addressed in the Oconee current licensing basis.

License renewal scoping criteria requires that systems, structures, and components (SSCs) relied upon to remain functional during and following design basis events, SSCs whose failure could prevent satisfactory function of the aforementioned, and SSCs whose function is required to demonstrate compliance with certain regulations, be included within scope. The Chilled Water System is a non-safety related system and engineering analysis shows that its loss of function during a design basis event can be withstood for 24 hours, upon which time several options of compensatory actions exist. As stated in Oconee Nuclear Station Licensee Event Report 287/1998-06, "The compensatory actions are simple in that air circulation can be established by opening doors and positioning dampers. The actions do not require complicated engineering analyses or detailed logistics to complete." Based on the engineering analysis, the Chilled Water System has historically not been included in other regulated programs as a system relied upon to support design basis event mitigation.

Notwithstanding the above, Duke now commits to including the Chilled Water System within the scope of license renewal.

SER Open Item 2.2.3.6.1.2.1-1
In Section 2.7.2, "Structural Components," of Exhibit A to the LRA, the applicant does not identify water stops, expansion joints, and structural sealants or caulking as structural components requiring an AMR. Section.2.7.3 of Exhibit A of the LRA states that all below grade construction joints in exterior walls are protected by cast-in-place water stops. The applicant stated (in response to RAI 2.7-3) that the water stops do not support any component intended functions and therefore are not subject to an AMR. The staff does not agree with the applicant's response because ground water in-leakage into the auxiliary building could occur as a result of degradation to the water stops. This leakage may cause flooding of equipment within the scope of license renewal and should be subject to an AMR (UFSAR Section 3.4.1, "Flood Protection," discusses the effects of flooding).

As discussed in Subsection 3.8.3.1 of this report, expansion joints are nonmetallic components that play important roles in maintaining the integrity of the components to which they are connected. Expansion joints perform their intended functions without moving parts or a change in configuration or properties, are not typically replaced based on a qualified life or specified time period, and therefore, should be subject to aging.

In addition, structural sealants or caulking are not addressed in Table 2.7-1or any other subsection under Section 2.7 of Exhibit A of the LRA. As discussed in Subsection 3.8.3.1 of this report, caulking is a nonmetallic component that plays important roles in maintaining the integrity of the components to which it is connected. These structural sealants perform their intended functions without moving parts or a change in configuration or properties, are not typically replaced based on a qualified life or specified time period. In addition, as stated in the staff's position regarding consumables (see License Renewal Issue No. 98-0012, "Consumables," dated April 20, 1999), structural sealants that are within the scope of license renewal typically meet the requirements under 10 CFR 54.21(a)(1)(i) and (a)(1)(ii). Structural sealants are often required for containment and structural integrity of safety-related structures, and perform these functions without moving parts or change in configuration or properties. These sealants are typically not replaced based on qualified life or specified time period, are often relied upon for decades of service, and are subject to aging. Therefore, structural sealants should be subject to an aging management review.

On the basis of the above evaluation, water stops, expansion joints, and structural sealants or caulking that are within the scope of license renewal, should be subject to an AMR.

Note: In addition to the above, the following discussion can be found in Section 3.8.3.1.8 of the SER.

In staff RAI 3.7.6-4, Duke was asked to discuss the basis for not including waterproofing membranes in Table 3.7-4 of the ONS LRA if they were used in the Keowee structures' exterior walls and base slabs to protect the concrete foundations or inhibit infiltration/seepage of ground water. The applicant was also asked to discuss ONS's approach to managing the effects of aging on the waterproofing membranes. Duke's response to the RAI stated that waterproofing membranes were not used in the Keowee structures to protect the concrete foundations or inhibit infiltration/seepage of groundwater. Duke's response, however, did not indicate whether the Keowee structure or other inscope structures experienced any kind of seepage of groundwater or whether the groundwater leaching that might be anticipated at the construction joints was observed at the SSF during a recently performed scoping inspection at the ONS. Duke is requested to provide a list of the ONS inscope structures that had or are experiencing observable seepage or leaching by groundwater from aging degradation of sealants and caulking in concrete components, and is requested to discuss its approach for managing the aging effects. This information should be provided as part of Open Item 2.2.3.6.1.2.1-1.

Duke Response to SER Open Item 2.2.3.6.1.2.1-1
Duke does not define materials such as caulking, sealants, and waterstops to be structures or components. However, Duke recognizes that limited situations may exist where these materials are important in maintaining the integrity of the components to which they are connected.

The license renewal structure or component intended functions supported by these materials are limited to three functions. These functions are:

1) Maintaining pressure boundary. This function is limited to the Control Room. Control Room pressure boundary which is addressed by surveillance testing to demonstrate compliance with Oconee Improved Technical Specification 3.7.9.
2) Providing a rated fire barrier. Sealants and caulking that support this function are addressed as part of the fire barrier penetration seals in the Application. Fire barrier penetration seals are discussed in Sections 2.7.2.4 and 3.7.2.4 of Exhibit A of the Application. Aging of the fire barrier penetration seals is managed by the Oconee Fire Protection Program that is discussed in Section 4.16 of Exhibit A of the Application.
3) Providing a flood barrier. Caulking, sealants, and waterstops that support this function are limited to those contained in structures whose function is to provide a flood barrier.

This function is identified as function #8 in the tables in Section 2.7 of Exhibit A of the Application. Specifically, the Auxiliary Buildings and the Standby Shutdown Facility are identified as providing internal/external flood barriers.

Caulking, sealants, and waterstops that support the flood barrier function are addressed below. Sealants associated with the Control Room pressure boundary are addressed in response to SER Open Item 2.2.3.4.3.2.1-2.

Reinforced concrete walls located below grade and flood curbs in the Auxiliary Buildings and the Standby Shutdown Facility provide a protective barrier for internal/external flood events. Caulking, sealants, and waterstops are important in maintaining the integrity of these walls and flood curbs. Caulking, sealants, and waterstops are used to seal joints in the walls and flood curbs. Degradation of the caulking, sealants, and waterstops may result in loss of the ability of the walls and flood curbs to provide a flood barrier. Gross degradation of these materials would be required to allow enough water to seep through the joint to produce flooding.

Degradation of the caulking, sealants, and waterstops in the Auxiliary Buildings and the Standby Shutdown Facility is managed by the *Inspection Program for Civil Engineering Structures and Components*. The program is discussed in Section 4.19 of Exhibit A of the Application. The program visually inspects concrete for evidence of degradation of the caulking, sealants, and waterstops. Evidence of degradation may include but is not limited to water in-leakage, leaching, peeling paint, or discoloration of the concrete. Inspection findings are evaluated to determine the appropriate corrective action that may include monitoring or repair/replacement of the caulking or sealant.

Oconee operating experience has identified instances where degradation of these materials has resulted in discoloration of the concrete and leaching in the Keowee Powerhouse, the Auxiliary Buildings, and Standby Shutdown Facility. While Keowee concrete structures do not provide a flood barrier and water seepage is a normal occurrence in dam facilities, evidence of leaching and discoloration of concrete has been detected and corrective actions have been taken. For the Auxiliary Buildings and Standby Shutdown Facility, caulking and sealants have been repaired/replaced. Where discoloration and leaching have been identified along joints with waterstops, the joint has been sealed on the inside surface of the concrete. Continued implementation of the *Inspection Program for Civil Engineering Structures and Components* provides reasonable assurance that caulking, sealants, and waterstops will be maintained to support the intended functions of the Auxiliary Buildings and Standby Shutdown Facility in accordance with the CLB for the period of extended operation.

SER Open Item 2.2.3.6.4.2.1-1

The applicant stated that the Keowee structures use both reinforced concrete roof slabs and built-up roofing systems. The Keowee breaker vault that is located within the powerhouse has a reinforced concrete roof slab. The main structures, such as the Keowee powerhouse and the service bay structure have built-up roofing systems. The built-up roof system is comprised of a metal roof deck, covered with rigid insulation and rubberized material. The applicant stated that this roof system is a short-lived component and is subject to periodic replacement based on its service condition. Therefore , the applicant did not include the built-up roof system in Table 2.7-4 and did not consider it subject to an AMR. However, neither the rule nor the Commission guidance provided in the Statements of Consideration (SOC), allows the generic exclusion of structures and components based on performance or condition monitoring. An applicant may exclude from an AMR components or structures that are replaced on the basis of specific performance or condition monitoring activities if the following two conditions are met: 1) that the applicant identifies those structures and components in the LRA that are being excluded based on performance and condition monitoring, and 2) that the applicant submit a site-specific justification for the exclusion of these components.

Note: This above open item is repeated for the turbine building roof in Section 2.2.3.6.7.2.1 of this SER.

Duke Reponse to SER Open Item 2.2.3.6.4.2.1-1

Rather than generically excluding the roofs based on performance or condition monitoring, Duke has reevaluated whether the roofs are subject to aging management review based on function. Upon further investigation, Duke has determined that the roof systems for the Keowee Powerhouse and the Oconee Turbine Buildings are not subject to aging management review because they do not perform a §54.4 intended function. The Keowee Powerhouse and Oconee Turbine Building are within the scope of license renewal in accordance with the criteria in §54.4. Certain structural components of these two structures perform a structural intended function as identified in Tables 2.7-4 and 2.7-7 of Exhibit A of the Application. The roofs are components of the Powerhouse and Turbine Building, but the roofs do not perform an intended function. Degradation or loss of either the Keowee Powerhouse or Turbine Building roof will not result in loss of any structural, mechanical or electrical system or component intended function.

If the Keowee Powerhouse roof were to degrade, equipment located on the operating floor would remain sheltered/protected. Table 2.7-4 of Exhibit A of the Application identifies those components that perform the intended function of shelter/protection of safety-related equipment. The breakers are protected by a reinforced concrete breaker vault. The breaker vault is constructed of reinforced concrete walls, floor slab and roof slab. Electrical equipment is protected by the switchgear cabinets. Switchgear cabinets are included with electrical panels and enclosures. The turbine generator is protected with a metal cover. Therefore, degradation of the Powerhouse roof would not result in the loss of any component intended function.

The Turbine Building contains safety-related equipment located in the basement. If the Turbine Building roof were to degrade, equipment located in the basement would remain sheltered/protected. The equipment is located beneath several reinforced concrete floors. These floors provide shelter/protection to the equipment located beneath them (See Table 2.7-7 of Exhibit A of the Application for identification of the Turbine Building components which provide shelter/protection). Therefore, degradation of the Turbine Building roof would not result in the loss of any component intended function.

In summary, the Keowee Powerhouse roof and the Turbine Building roof do not perform an intended function within the scope of license renewal. Degradation of these components has been evaluated and it has been determined that degradation would not result in the loss of any system, structure, or component intended function. Therefore, aging management review of the Keowee Powerhouse and Turbine Building roofs is not required.

The requirements of §54.21(a)(1) are to list and identify within the application those structures and components subject to aging management review. Since the roofs are not subject to aging management review, the roofs are not listed in Tables 2.7-4 and 2.7-7 of Exhibit A of the Application.

SER Open Item 2.2.3.7-1
In Section 2.6.6.1.2 of the application, the applicant identified insulated cables and connections used for fire detectors as part of the fire detection system and excluded them from an AMR because they are replaced based on a performance or condition program. In response to RAI 2.6-4, the applicant referenced SOC Section III.f.(l)(b) and 10 CFR 54.21 (a)(1)(ii) as the basis for excluding fire detector cables and connections from an AMR. However, the applicant also stated that the fire detector cables are not physically different from other insulated cables. There is no generic exclusion for components that are replaced based on performance or condition. An applicant may exclude from an AMR components or structures that are replaced on the basis of specific performance and condition monitoring activities if the following two conditions are met: 1) that the applicant identifies those structures and components in the LRA that are being excluded based on performance and condition monitoring, and 2) that the applicant submit a site-specific justification for the exclusion of these components. The applicant should either provide a plant-specific justification for excluding these components from an AMR or include them in an AMR.

Response to SER Open Item 2.2.3.7-1
Duke agrees with the staff position regarding the Oconee fire detector cables. Following discussion with staff during the on-site inspections, Duke now understands that detection of degradation is a necessary aspect of excluding a component based on performance or condition. The fire detector cables as part of the fire detection system are provided with failure detection, not detection of degradation. Therefore, Duke now includes the fire detector cables in the aging management review of cables and connections. The aging management review for fire detector cables is detailed below.

The function of a fire detector cable is to electrically connect the fire detector with the rest of the fire detection system so that a fire alarm can be initiated on the fire detection panel. Radiation and heat are the two applicable stressors for fire detector cable insulation and the potential aging effects are reduced insulation resistance and electrical failure. Extreme aging of insulation could lead to the formation of cracks that may expose the electrical conductors within the cable to a common ambient environment. If enough moisture is introduced into this common ambient environment, the electrical conductors in the cable could short to each other or to ground. The fire detection system provides constant supervision of the fire detector wiring. Upon the detection of a short circuit or ground condition on a fire detector cable, a trouble alarm is initiated on the fire detection panel. When the fire detection panel receives either a fire alarm or a trouble alarm, a statalarm is initiated in the control room to inform the plant operators. There is no fire detector cable failure caused by aging that would result in a panel alarm and control room statalarm not being initiated. Therefore, the fire detector cables are fail-safe and no aging management is required.

SER Open Item 2.2.3.7-2
During a plant walkdown at the ONS, the staff identified a generic renewal issue regarding exclusion of equipment from an AMR that meets the scoping criteria of 10 CFR 54.4 but is kept

in storage. Specifically, this issue focuses on the replacement of pump motors, switchgear, and electrical cables associated with the low-pressure injection, high-pressure injection, or low-pressure service water that may be required for cold shutdown in order to comply with Appendix R to 10 CFR Part 50, which requires the reactor to be in cold shutdown within 72 hours after a fire accident. The identification of the structures and components that are excluded in 10 CFR 54.21(a)(1)(i) presumes that they are installed in the plant and are challenged by routine operation or periodic testing. The logic that was used to screen out systems, structures, and components that perform active functions does not apply to motors and switchgear stored in warehouses because they are not challenged by routine operation or periodic testing. Therefore, pump motors and switchgear that are stored in warehouses should be subject to an AMR.

Response to SER Open Item 2.2.3.7-2
The Appendix R equipment stored in the warehouse (motors, switchgear) are routinely inspected, tested and maintained. Since the Appendix R motor and switchgear are treated similarly to the motors and switchgears installed in the plant, they are appropriately considered active and are not subject to an aging management review.

SER Open Item 3.2.12-2
For the decay heat removal coolers and the reactor building cooling units, the applicant determines heat removal capacity (based on flow rates and temperature difference) and compares the test results to the acceptance criteria. For the SSF heat exchangers, the applicant verifies acceptable cooling-water flow rates through these heat exchangers. The staff requests the applicant to state specifically what the acceptance criteria are for each of these heat exchangers and provide the basis for the acceptance criteria. The applicant should discuss in its response how the acceptance limits ensure sufficient heat transfer capacity under both normal operating and accident conditions. Also, for the decay heat coolers, the applicant implements corrective actions if the heat transfer capacity degrades more than 4% from the last test. The staff requests the applicant to state if similar criteria are in place for the reactor building cooling units and the SSF heat exchangers. If not, the applicant should discuss why this is not needed. The applicant should also discuss in its response the basis for implementing corrective actions upon measuring a 4% degradation in heat transfer capacity. The insufficient specificity on the acceptance limits and corrective actions for the heat exchangers are identified as an Open Item.

Notes on 3.2.12-2

There is a lot in this question. Here is what I grasped as the question within this OI:

1) what are acceptance criteria and how do you extrapolate them to normal and accident conditions
2) Is 4% criteria in place for RBCUs. If not, why not?
3) Why is 4% okay?

I answered them out of order.

Duke Response to SER Open Item 3.2.12-2
Please see response to Open Item ?? for SSF heat exchanger information. The remainder of this Open Item response pertains to the Decay Heat Coolers and Reactor Building Cooling Units only.

Total containment heat removal capability is determined by the combined heat removal capabilities of the Decay Heat Coolers and Reactor Building Cooling Units in conjunction with the Reactor Building Spray System. The acceptance criteria for the Decay Heat Coolers and the Reactor Building Cooling Units is that total containment heat removal capability exceed the design basis required containment heat removal capability. Test results are used in calculations that project the heat removal capability through the upcoming fuel cycle and compare it to the design basis required heat removal capability. Heat removal capabilities of the Reactor Building Cooling Units and Decay Heat Coolers are interdependent. When heat exchanger performance test data are collected for the Reactor Building Cooling Units and Decay Heat Coolers, the data is used to effectively calculate a heat transfer coefficient for each cooler. This heat transfer coefficient is then used to calculate heat removal capability in normal and accident conditions using variables that apply to normal and accident conditions, such as lake temperature and Reactor Building atmospheric temperature. These calculations use the test data to prove that total required containment heat removal capability of the Reactor Building Cooling Units, Decay Heat Coolers, and Reactor Building Spray System during normal and accident conditions exists. If calculations show acceptance criteria are not met, the coolers are cleaned. For more information on cleaning of these coolers, refer to response to RAI 4.17-1. Details on performance relative to conditions of the test setup and the models used to analyze test data can be found in Section 6.2.1.1.3, Subsection "Containment Heat Removal Systems", of the Oconee UFSAR.

Because the decay heat coolers can only be tested during a refueling outage, an additional conservatism is applied to heat transfer value acceptance criteria to ensure operation under all design basis scenarios. A 4% degradation in heat transfer capacity is imposed on the Decay Heat Coolers above and beyond the requirement that the total containment heat removal exceed its design basis requirement. This 4% criteria is based on testing and operating experience. Historically, testing has shown that less than 4% degradation is expected during a normal operating cycle. Therefore, if 4% degradation is detected, corrective action to chemically clean the coolers is implemented. This degradation criteria is not part of the design basis of the coolers, it is simply in place as a conservative measure to ensure that corrective actions are taken before the heat removal capabilities of the containment heat removal systems approaches the design basis limit. This 4% degradation criteria is not applied to the Reactor Building Cooling Units. While the Decay Heat Coolers can only be tested during a refueling outage, the Reactor Building Cooling Units can be tested any time during unit operation and are tested quarterly. Since frequent testing allows for more effective trending on the Reactor Building Cooling Units, the additional 4% degradation criteria is not imposed upon them.

SER Open Item 3.3.3.1-1

In Section 3.3.4.2 of Exhibit A of the LRA, Duke emphasizes that in spite of the water infiltration and high humidity in the ONS tendon galleries, the tendon components are well protected. Based on the information contained in the database on the condition of the tendon grease caps and the bearing plates in tendon galleries (see Plates 2, 7, and 11 in Appendix A of NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures"), the staff does not agree with the applicant's conclusion. The intended function of the post-tensioning system is to impose compressive forces on the concrete containment structure to resist the internal pressure resulting from a design-basis accident with no loss of structural integrity. Operational experience, as documented in NUREG-1522, has shown that water infiltration and high humidity in the tendon gallery can be a significant aging effect on the vertical tendon anchorages that could potentially result in loss of the ability of the post-tensioning system to perform its intended function. Therefore this aging effect needs to be adequately considered.

Duke Response to SER Open Item 3.3.3.1-1
Loss of material due to corrosion was determined to be an applicable aging effect for the tendon anchorage as described in Section 3.3.4.5 of Exhibit A of the Application. The *Containment Inservice Inspection Program* is credited with managing loss of material due to corrosion of the tendon anchorage for the extended period of operation. The *Containment Inservice Inspection Program* is discussed in Section 4.8 of Exhibit A. As part of the *Containment Inservice Inspection Program*, the tendon anchorage are inspected for loss of material due to corrosion in accordance with ASME Section XI Subsection IWL. Based on the discussion in Section 4.8 of Exhibit A of the Application, the implementation of the *Containment Inservice Inspection Plan* provides reasonable assurance that the aging effects of the tendon anchorage, including those located in the gallery, will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

SER Open Item 3.4.3.3-1
The surface examination will be a one-time inspection performed when a heater bundle is removed. If the results are not acceptable, they may be used as a baseline for establishing a longer term programmatic action covering all ONS pressurizer heater bundles. However, Duke has not stated when the heater bundle will be removed for examination and the basis for scheduling the inspection.

Duke Response to SER Open Item 3.4.3.3-1
As described in Section 4.3.7.2 of Exhibit A of the Application, the surface examination of a sample population of peripheral pressurizer heater bundle penetration structural welds on one Unit 1 heater bundle will be performed when a heater bundle is replaced. Replacement of the bundle would occur as a result of the inability to meet operational requirements for heater operation owing to inoperable heater elements. Heater bundle replacement may occur either prior to the period of extended operation or during the period of extended operation.

Duke believes that this inspection may be aligned to when a Unit 1 heater bundle is replaced whenever that may occur due to the impractical nature of such an inspection otherwise. The failure of a structural weld that attaches the heater sheath to the Alloy 600 heater sleeve or failure of the weld that attaches the heater sleeve to the Alloy 600 diaphragm plate would result in leakage within the make-up system capacity and the integrity of the heater bundle bolted closure would not be compromised. No loss of pressurizer function would occur due to leakage at either of these welds.

It is unreasonable to remove an operable heater bundle assembly that remains functional. In order to inspect the heater sheath-to-heater sleeve structural welds and heater sleeve-to-diaphragm plate structural welds, the seal welds for the diaphragm plate made from Alloy 82/182 welds must be cut and the bundle removed. The cutting and rewelding of the Alloy 82/182 seal weld may increase the likelihood of cracking by PWSCC (implies that seal weld is a structural weld but is not—essentially a permanent gasket). Ultimately, leakage at the seal weld can have a higher consequence on the functionality of the pressurizer and may lead to steam cutting and loss of material at or near the bolted connection to the pressurizer shell. Because the pressurizer external surface examination in accordance with ASME Section XI would detect leakage at any location associated with the heater bundles, more immediate action will be taken should it be necessitated by something other than inoperable heater elements.

SER Open 3.4.3.3-2

For ONS Unit 1, Duke proposes to inspect the heater sheath-to-heater sleeve penetration welds, but not the heater-sleeve-to-heater-bundle diaphragm plate. The ONS Unit 1 heater sleeves and heater bundle diaphragm plates are fabricated from Alloy 600, which is susceptible to PWSCC. Hence, both the heater-sheath-to-sleeve plate and the heater-sleeve-to-bundle diaphragm plate need to be inspected to determine whether the Alloy 600 materials in the heater bundle have experienced PWSCC. The heater sheaths and heater bundle diaphragm plates in ONS Units 2 and 3 are stainless steel. Therefore, they are not susceptible to PWSCC. The ONS Unit 1 heater bundles are susceptible to PWSCC and the ONS Unit 2 and 3 heater bundles are not. Therefore, the scope of the inspection of Unit 1 should be expanded to include the heater sheath-to-sleeve plate and the heater-sleeve-to-bundle diaphragm plate.

Duke Response to SER Open Item 3.4.3.3-2

The Unit 1 heater bundle inspection will be revised to include inspections of the heater sheath-to-heater sleeve structural weld and the heater sleeve-to-heater bundle diaphragm plate structural weld (Reference Figure 2-8 of BAW-2244A). Inspections of Unit 2 or Unit 3 heater bundle welds are not required.

SER Open Item 3.6.3.3.2-1

As stated earlier, the staff found the program scope and parameters monitored to be acceptable. The applicant analyzes the oil samples following industry guidance; specifically, ASTM D95-83, "Water in Petroleum and Bitumens." This standard provides a widely used and accepted method of determining the amount of water in a sample of oil, but it does not provide recommendations for sampling frequency. The applicant plans to take oil samples every six months for analyses. The applicant also stated that the program will be implemented by February 6, 2013. The applicant did not provide the basis for the six month sampling interval, nor did the applicant justify delaying the implementation of the program until possibly February 6, 2013. The relatively frequent oil sampling of every six months indicates to the staff that there is a need to perform this testing on a fairly aggressive schedule. The staff requests the applicant provide the basis for the 6-month sampling interval as well as the basis for implementing the program by the end of the current operating period.

Duke Response to SER Open Item3.6.3.3.2-1

The Keowee Oil Sampling Program is a new program for License Renewal because it has not been previously performed by Oconee Nuclear Station. The sampling activity described in the program has been performed for a number of years by Duke Energy's Hydroelectric Facility Department. The activity has historically been set up to be conducted every six months, although it has not always been performed rigorously on schedule or documented in a retrievable fashion. Since the activity is needed to manage aging for license renewal, Duke decided to bring the program under the nuclear department within Oconee control and designate it a new program. Since the frequency has historically been every six months, and the Hydroelectric Facility Department experience has shown this frequency to be acceptable, it was decided to leave that frequency unchanged. Steps have already been taken to establish this new program, such as taking the oil sample by Oconee personnel and procedure rather than by the Hydroelectric Facility Department. Although this new program, like the other new programs credited for license renewal, was originally committed in the License Renewal Application to an implementation date of February 6, 2013, Duke is now revising its commitment with respect to the timing of the implementation of this new program. The Keowee Oil Sampling Program will be implemented concurrent with the UFSAR update required by 10 CFR 50.71(e) after the Oconee renewed operating license is issued by the NRC.

SER Open Item 3.8.3.1-1

In the discussion of the environment around the steel components in a fluid environment, Duke stated that the ONS UFSAR limits the spent fuel pool temperature to 183° F. A review of Section 9.1.3 of the UFSAR shows a limit of 150° F for normal heat load and abnormal heat load when the three-pump-cooler configuration is in operation. It also shows a temperature limit of 205° F for abnormal heat loads when the two-pump-cooler configuration is in operation. From the standpoint of aging effects assessment, sustained effects under normal heat load are important. The staff requests that the applicant clarify the discrepancy between the above-noted UFSAR temperature limits. If the real normal load limit is above 150° F, the staff is concerned that, although the temperature of 183° F may have no effect on the steel components, it could have an aging effect on the concrete of the spent fuel pool walls and slabs. The applicable code (ACI 349) limits the concrete temperature to 150° F. This limit of 150° F does not guard against additional cracking. However, it assures that the concrete properties, such as compressive strength and modulus of elasticity, would not be significantly affected. The applicant should discuss the aging effects of the temperature (183° F) on the concrete cracking and concrete properties.

Duke Response to SER Open Item 3.8.3.1-1

Normal operating temperature for the spent fuel pool is below 150°F [Reference Oconee UFSAR Section 9.1.3.1]. Control room operators monitor spent fuel pool temperatures at all times in accordance with a periodic surveillance procedure, which is required by Improved Technical Specifications (ITS 5.4.1). Spent fuel pool normal operating temperatures range from approximately 90°F to 120F. These temperatures are well below the ACI 349 threshold where degradation would occur to concrete. Therefore, there are no applicable aging effects resulting from the temperature of the spent fuel pool.

The temperature limit of 183°F in Section 3.7.1 of Exhibit A of the Application is incorrect. Bulk spent fuel pool temperatures for the spent fuel pools remain at or below 150°F (UFSAR Sections 9.1.3.1 and 3.8.4.4). As discussed in Section 3.8.4.4 of the Oconee UFSAR, the spent fuel pool walls were analyzed for thermal loads.

SER Open Item 3.8.3.1-2

The discussion of the industry and ONS-specific experience database in Sections 3.7.1 and 3.7.2 of Exhibit A of the LRA does not capture (1) the essence of the results of the ONS baseline inspections that would have been performed during the implementation of the Maintenance Rule, and (2) the instances of the reported unusual events, such as the water leakage from the spent fuel pool liners. The conclusions drawn from this information could affect the applicable aging effects.

Duke Response to SER Open Item 3.8.3.1-2
Industry and Oconee specific operating experience are included in the discussion in Sections 3.7.1 and 3.7.2 of Exhibit A of the Application. More detailed information concerning the findings during the implementation of the Maintenance Rule are included in Section 4.19 of Exhibit A of the Application. Review of the findings of the Maintenance Rule inspections did not result in the identification of any aging effects other than those that were identified. Therefore, the conclusions of the Maintenance Rule civil inspections were taken into consideration during the aging effect evaluation and it was determined that the conclusions of the inspections did not change the applicable aging effects.

SER Open Item 3.8.3.1.9-1
Regarding the consideration of the applicability of the loss of material resulting from the aging effect to the ONS cable tray and conduit category, Duke determined that the aging effect applies to those cable trays and conduits located within the reactor building; however, the same aging effect is not considered plausible for cable trays and conduits located in other parts of the ONS plants (refer to Tables 3.7-1 through 3.7-6 of the LRA). Duke is requested to provide additional information to justify this differential treatment of the aging effect covering cable trays and conduits located in structures other than the reactor building.

Duke Response to SER Open Item 3.8.3.1.9-1
As stated in Section 3.7.2.2.2 of Exhibit A of the Application, cable tray is constructed of galvanized sheet metal that does not have a tendency to age with time. In addition, a review of industry experience did not identify any aging effects for cable tray systems. The staff has agreed with this conclusion in Section 3.8.3.1.2 (page 3-192) of the Oconee License Renewal SER.

A review of Oconee specific operating experience has not identified any aging effects for cable trays located in any structures except for those located in the Reactor Building. Loss of material of cable trays due to boric acid corrosion has been identified in the Reactor Building. The cable trays in the Reactor Building are located in areas that are susceptible to boric acid leakage. Therefore, loss of material due to corrosion is an applicable aging effect for the cable trays in the Reactor Buildings.

Since loss of material due to corrosion has not been identified in Oconee or industry experience for cable tray located in environments other than the Reactor Building, loss of material is not an applicable aging effect for cable tray in those locations.

SER Open Item 3.8.3.2.5-1
ONS UFSAR Section 3.8.3.3 (related to the internal structures of the steel containment) states that the loads and load combinations considered for the design of the interior structures are described in UFSAR Section 3.8.1.3. Section 3.8.1.3 discusses the "calculated prestressing force" (after consideration of appropriate losses) as a load to be considered in load combinations tabulated in Table 3-14. Thus, the staff believes that the SSW prestressing tendons system is

part of the CLB. The applicant should provide information demonstrating that the prestressing forces in the SSW will be adequately maintained for the period of extended operation.

Duke Response to SER Open Item 3.8.3.2.5-1
Loss of prestress of the SSW tendons is managed by the *Tendon – Secondary Shield Wall – Surveillance Program*. Lift-off forces are measured and compared to established acceptance criteria. Where tendon lift-off readings have fallen below the minimum allowable, adjacent tendons were tested and tendons have been re-tensioned as needed. The prestressing forces in the SSW will be adequately maintained by the *Tendon – Secondary Shield Wall – Surveillance Program* during the period of extended operation. The results of the program (minimum required prestress, lift-off testing results, retensioning, etc.) are maintained on site and are available for staff review. Information concerning the performance of the secondary shield wall (SSW) tendon system is provided in Section 4.28 of Exhibit A of the Application. In addition, response to RAIs 3.7.7-1, 4.28-1 and 4.28-2 provide additional details on the SSW tendon program.

SER Open Item 4.2.3-1
The applicant indicated that these locations would be managed by the ONS FMP. The adequacy of this program to address the flaw evaluation TLAA cannot be determined without additional information. The applicant should provide the following information relating to the locations identified in Section 5.4.1.2 of Exhibit A of the LRA that could not be demonstrated as acceptable for the number of controlling design basis transients:

- Characterize the indications identified by the ISI for each of the locations listed (i.e., nature, length, through-wall extent and through–wall location);
- From the results of successive ISI of the same flaw locations, characterize the extent of growth of the indication(s) as indicated by the successive examinations;
- For each of the fracture mechanics analyses, identify the transient and number of cycles assumed in the analyses, and the ASME Code Section XI, IWB-3600 criteria that was not satisfied at the end of the license renewal period;
- As of January 1, 1999, what is the status of the actual number of transient cycles for each location, the plant status regarding effective-full-power-years (EFPY), and the estimated EFPY at the end of the license renewal period?
- If the transient cycle count approaches or exceeds the allowable design limit, identify the corrective action steps that could be taken.

Duke Response to SER Open Item 4.2.3-1
Section 5.4.1.2 of the license renewal application (LRA) describes time-limited-aging-analyses (TLAA) related to flaw growth acceptance for the reactor coolant system and Class 1 components at Oconee. As described in the LRA, inservice inspection (ISI) at Oconee, in accordance with ASME Section XI ISI requirements, has lead to the identification of crack-like indications, primarily in welds. The fracture mechanics analyses used for flaw acceptance through the current license period have been reviewed for acceptability for the period of extended operation. This review identified several general flaw locations that could not immediately be demonstrated to be acceptable for the number of controlling design basis transients, but which will continue to be managed by the Oconee *Thermal Fatigue Management Program*.

Since the submittal of the application in July 1998, most of the locations identified in Section 5.4.1.2 of Exhibit A of the LRA that could not be demonstrated to be acceptable have been reanalyzed and found acceptable for the number of controlling design basis transients. The following locations were not reanalyzed: flaw in the pressurizer upper head to shell weld at Unit 2, flaw in the weld that connects the OTSG upper head to tubesheet at Unit 2, and the discontinuities in the CRDM motor tube housings at Units 1 and 2. At present, the pressurizer upper head to shell weld at Unit 2 is being evaluated for the period of extended operation. The current design cycle limit and any revised design cycle limit resulting from reanalysis will be managed for this location by the Oconee *Thermal Fatigue Management Program*. The OTSGs at Unit 2 and CRDMs at Units 1 and 2 are in the process of being replaced, thus reanalysis was not needed. Therefore, all locations, with the exception of the flaw in the pressurizer upper head to shell weld on Unit 2, are acceptable for the period of extended operation. Table 1 provides a range of updated information (i.e., controlling transients and extent of growth) associated with each of the locations.

Note that Table 1 does not contain information associated with the CRDM motor tube housing. The CRDM motor tube housing indications are described in the BAW- topical report entitled "A Study of Discontinuities in Control Rod drive Motor Tube Extensions," BAW-10047, Revision 1, August 1972. A fracture mechanics analysis, which applies to the Type A drives at Oconee Units 1 and 2, was performed to show that the CRDM motor tube extension fabrication discontinuities were acceptable for the design life of the plant. The CRDM fracture mechanics analysis will not be updated for license renewal since the Type A CRDMs at Units 1 and 2 will all be replaced with Type C drives prior to the end of the current term of operation. The Type C drives do not contain the subject fabrication discontinuities.

Table 1. Summary of Specific Oconee Fracture Mechanics Calculations of Flaw Indications									
Component Flaw Location	**Unit**	**Flaw Size Per IWA-3000**			**Controlling Transients**	**Inservice Inspection Results**	**Controlling design transient cycle limit for location**	**Total number of Transient Cycles the indication is acceptable for**	**Accumulated Transient Cycles when indication was observed**
		a(in.)	**t (in.)**	**l(in.)**					
Pressurizer near heater bundle	1	1.075	7.0	2.15	Heatup and Cooldown	B&W first reported this indication in Outage 7 in 1983. The indication was sized at 12.33% a/t. Monitoring of the indication in three successive outages, (Outage 9 March 1986, Outage 11 February 1989 and Outage 13 September 1991) showed no increase in size.	360	448	88
Pressurizer support lugs Larger of two similar indications	1	0.7	3.5	1.35	Heatup and Cooldown	B&W first reported two unacceptable indications in Outage 6 in 1981. Fracture mechanics analysis accepted these indications. The support lug welds were reexamined in Outage 7 in 1983 and a second fracture mechanics analysis performed. The indications were acceptable. No monitoring was performed.	360	425	65
Steam generator at the upper head to tubesheet region	1	0.57	8.05	6.1	Heatup and Cooldown	B&W first reported this indication in Outage 12 in 1990. The indication was sized at 7.1% a/t. Monitoring of the indication in Outage 14, 1992 and Outage 17, 1997 showed no increase in size.	120	207	87

Reactor vessel at the reactor vessel flange to shell region	1	1.15	12.0	4.4	Heatup and Cooldown Inservice Leak and Hydro	B&W first reported thirteen unacceptable indications in Outage 9 in 1986. The indications were re-evaluated in 1987 and were finally resolved as geometry. No further action was taken.	Not Applicable	Not Applicable	Not Applicable
Core flood tank dump valve to nozzle Largest flaw reported is a=1.30 inches and 1.65 inches	2	1.30	2.7	1.65	Heatup and Cooldown	B&W first reported fifteen unacceptable indications in Outage 6 in 1983. Fracture mechanics analysis accepted these indications. The indications were re-sized in Outage 7 in 1985 with different transducers and consequently, twelve indications were found unacceptable. A second fracture mechanics analysis accepted these indications. Nuclear Energy Services (NES) was contracted to monitor these indications in Outage 8 in 1986. NES recorded nineteen indications of which only four were unacceptable. These four indications were correlated with the previous B&W data. Review of both sets of data show that the flaws had not increased in size.	360	455	95
Pressurizer upper head to shell region Larger of two similar indications	2	0.55	4.75	7.0	Heatup and Cooldow LOCA	B&W first reported two unacceptable indications in Outage 1 in 1976. Fracture mechanics analysis performed on the larger of the two indications shows it acceptable. Subsequent examinations in Outage 2 (1977), Outage 3 (1979) and Outage 4 (1980) show no flaw growth.	240	279	39

Open Item 4.2.3-2
Since GSI–190 has not been resolved, the staff requested, in RAI 1.5.5-1, that the applicant discuss how it satisfies the relevant portion of paragraph 54.29 of the license renewal rule as discussed in the statement of considerations (SOC) (60 FR 22484, May 8, 1995) in the absence of the staff's endorsement of EPRI Report TR-105759. The applicant did not provide a technical rationale addressing the adequacy of components in the RCP boundary considering environmental fatigue effects pending the resolution of GSI–190. In its response to the RAI, the applicant stated that the concerns of GSI–190 are not directly related to the ONS thermal fatigue design and licensing basis. The applicant further indicated the application contains its technical rationale for concluding that the effects of thermal fatigue will be adequately managed for the period of extended operation or until GSI–190 is resolved. On this basis, the applicant concluded that the relevant portions of 50.29 of the license renewal rule as discussed in the statement of considerations (60 FR 22484, May 8, 1995) are met by the ONS FMP. The staff does not agree with the applicant's reasoning. As discussed above, the staff assessment for GSI–166 found that there is sufficient conservatism in the CLB for the 40–year design life. However, this conclusion could not be extrapolated beyond the current facility design life. As a consequence, the staff recommended that a sample of components with high usage factors be evaluated using the latest available environmental fatigue data for any proposed period of extended operation. The staff also initiated GSI–190 to further evaluate this issue for license renewal.

On the basis of the preceding discussion, the staff concludes that the applicant's TLAA of the RCS is not adequate to address the fatigue concerns for operation beyond the current design life of 40 years. The applicant must either develop an aging management program that incorporates a plant-specific resolution of GSI–190 or submit a technical rationale which demonstrates that the CLB will be maintained until some later point in time in the period of extended operation, at which point one or more reasonable options would be available to adequately manage the effects of aging. If GSI–190 is resolved prior to the period of extended operation, the applicant may follow the resolution of the GSI.

Duke Response to Open Item 4.2.3-2:

Overview
In response to SER Open Item 4.2.3-2, the Oconee Thermal Fatigue Management Program can be modified in the future to incorporate a plant-specific resolution of GSI-190. GSI –190 pertains to the adequacy of fatigue design life when environmental effects are considered for light water reactor components beyond a 40 year operating period. The purpose of proposing this future modification to the Thermal Fatigue Management Program is to offer a technically feasible option for resolving GSI-190 for Oconee which would assure that the current licensing basis is maintained during extended period of operation when modified design rules may show that operational fatigue is no longer bounded.

The current Oconee Thermal Fatigue Management Program relies on cycle counting to assure compliance with the current licensing basis. This technique has been accepted by NRC in the Oconee Safety Evaluation Report in Section 4.2.3.4. Any modifications to the Oconee Thermal Fatigue Management Program to account for environmental effects cn fatigue life should consider the practical and reliable aspects of cycle counting. Section 5.4.1 of the Oconee License Renewal Application provides further specifics on the Oconee design cycle basis.

Future modifications to the Oconee Thermal Fatigue Management Program will adjust the allowable design cycles limits being tracked by the program. Further details on the process to modify the design cycle limits and the Thermal Fatigue Management Program are provided in the following discussion.

Details

The current Oconee Thermal Fatigue Management Program relies on cycle counting to assure compliance with the current licensing basis. This technique has been accepted by NRC in the Oconee Safety Evaluation Report in Section 4.2.3.4. Any modifications to the Oconee Thermal Fatigue Management Program to account for environmental effects on fatigue life should consider the practical and reliable aspects of cycle counting.

<u>Current Practice</u> - In the analysis basis, a piecewise linear relationship exists between design cycles and cumulative usage factor which is defined by the Code. The relationship between design cycles and cumulative usage factor is considered piecewise linear because each stress range pair associated with a given transient produces its own linear rate. When pieced together these sets of stress range pairs add up to a value for an incremental usage at an allowable number of design cycles for that transient. The sum of all of the incremental usage factors for all of the transients will equal the cumulative usage factor (CUF) at the design number of occurrences of all of the transients.

Figure 1 shows an illustrative transient where two stress range pairs are included in the transient of interest, giving a bi-linear relation between the number of cycles and usage factor. When pieced together the usage factor value at a given number of cycles is determined. Also it should be noted that the cumulative usage factor base for this transient is not zero. The base is the sum of the usage factors for all other transients. Each individual transient thus contributes to the overall cumulative usage factor through its own set of piecewise rates. In Figure 1, by including the usage factor value for this transient with all others, the overall component design cumulative usage factor is determined. (For convenience, the transient of interest is shown as the last in the sum producing the CUF. This is an allowed technique since no modification is being made at this time to any stress range magnitudes or postulated occurrences, hence there is no order dependency.)

From the design, the postulated cycles for each appropriate design transient are known. These cycles are the actual evaluated cycles and hence are the analysis allowable limit and are tracked via the Thermal Fatigue Management Program. Figure 2 shows the actual accumulation of cycles for an illustrative transient over time. The original analysis allowable cycle limit is noted. Figure 2 also shows a conservative projected cycle accumulation rate for 60 years of operation. In this example, the projection does not exceed the allowable number of cycles for the 60 year period.

<u>Proposed Modified Practice</u> - Specifically, a modification to the number of allowable design cycles will accommodate issues associated with environmental effects on fatigue life (also referred to as environmentally assisted fatigue). Accounting for environmental effects on fatigue life requires the application of an environmental penalty factor to the design. Current industry thinking on the analysis basis associated with environmental effects on fatigue life causes penalties to be applied to cumulative usage factor values determined during design analysis.

This environmental penalty factor only needs to be applied to the usage factors for specific transients for selected components in order to find the lowest number of acceptable cycles for a given transient. The final focus will be to develop a modified analysis allowable cycle count limit to cover all RCS components that are managed by the Oconee Thermal Fatigue Management Program. The determination of the most limiting component for each transient can be made by examining the sensitivity of the individual transient on a component's CUF. Several components may require analytical examination in order to thus determine the most limiting component for each transient.

Figure 3 illustrates how the environmental penalty factor affects the incremental usage factor for an example transient. The penalty factor results in an increase in the rate of usage per cycle, thus for a given number of cycles, the incremental usage factor with the penalty factor becomes greater.

In this manner, this new Environmental Penalty CUF can be compared to the 1.0 Code allowable, and if greater, a new limit for the transient in question can be set such that this does not occur. Further details and different possible outcomes of this exercise are given below.

The exercise of applying the environmental penalty factor and comparing the projected cycles to any modified analysis allowable cycle limit for years 40 through 60 would need to be performed at an appropriate time before year 40. If the projections will exceed the reduced allowables prior to 60 years of operation, further detailed planning and management is required within the Thermal Fatigue Management Program. The remaining margins identified by the projections will establish the urgency of required corrective actions. As with the current Oconee Thermal Fatigue Management Program, it remains the responsibility of the utility to maintain operation within analyzed limits.

<u>Examples of Proposed Modified Practice</u> - Several likely outcomes exist when the environmental penalty factor is applied:

Example 1: (Illustrated in Figure 3) Apply the environmental penalty factor to all of the appropriate stress range pairs (for all of the transients) comprising the original overall component design CUF and find that the cumulative usage factor with the penalty (Environmental Penalty CUF) is less than or equal to 1.0. Make no adjustments to the original analysis allowable cycle limit being tracked against (example shown in Figure 2) and continue to track the accumulation of these cycles through the Oconee Thermal Fatigue Management Program.

Example 2: (Illustrated in Figure 4) Apply the environmental penalty factor to all of the appropriate stress range pairs of the original overall component design CUF and find that the Environmental Penalty CUF is greater than 1.0. This case has several steps in order to resolve.

1. The first step is to determine the modified number of cycles needed to keep the Environmental Penalty CUF equal to 1.0. This can be done mathematically by using the linear relationships between the cumulative usage factor and the number of cycles. Here, a new piecewise, diagonal line, labeled [1] in Figure 4, defines the intersection of the original design cycles and the Environmental Penalty CUF portion attributable to this transient. Then associated lines are drawn at CUF = 1.0 and a new modified analysis

allowable cycle limit (labeled [2] in Figure 4) is established. This modified analysis allowable cycle limit can now be used for comparison to actual accumulated and project cycles. As stated previously, the most limiting adjusted allowable for each transient will be determined by examining the sensitivity of this exercise on all of the individual components affected by that transient. This may be an iterative procedure involving spreading the cycle adjustments among several transients to keep all component CUFs less than or equal to 1.0.

2. The next step is review current actual accumulated thermal cycles previously recorded by the Thermal Fatigue Management Program and to project them at a conservative rate through 60 years of operation. The conservative rate is illustrated in Figure 2 by the increased slope of the cycle accumulation in the future.

3. The third step is to check the modified analysis allowable cycle limit against the projected cycle count. If the conservatively projected 60 year cycles are less than modified analysis allowable cycle limit, then the Thermal Fatigue Management Program simply needs to be adjusted to track against the modified analysis allowable cycle limit for the years 40 through 60. If the conservatively projected 60 year cycles are greater than the modified analysis allowable cycle limit, then a fourth step is needed.

4. The fourth step occurs only when the 60 year projected cycles are greater than the modified analysis allowable cycle limit. In this step, the Thermal Fatigue Management Program is adjusted to manage the reduced number of design cycles for the years 40 through 60. The program manager uses the amount of margin shown by the projections to gain a sense of urgency on when corrective actions are required to assure a valid design envelope remains for plant operation. Corrective actions, as currently established within the Thermal Fatigue Management Program are taken if the number of events is expected to exceed the limits of design within a manageable time period. A manageable time period is the time needed to complete actions to ensure the affected components stay within acceptable limits.

   Step 4 Note: - The number of cycles projected at step 2 may exceed the modified analysis allowable cycle limit for an Environmental Penalty CUF prior to 40 years of operation. This is simply a recognized discontinuity in the logic of applying the environmental penalty factor only to the 40 to 60 year operating period. This does not mean that the initial 40 year design envelope is somehow substandard.

Conclusion - Modifying the Oconee Thermal Fatigue Management Program to account for environmental effects on fatigue life is a feasible option to resolve GSI-190 for Oconee. Modifying the allowable design cycles managed within the program provides a specific means to account for environmental effects and to maintain plant operation within analyzed limits.

Duke Response to SER Confirmatory Item 4.2.1.3-1

The NRC staff issued RAI 3.3-6 in their letter dated November 14, 1997. The RAI requested additional information on the periodic Type A Integrated Leak Rate tests as part of the design loads considered in the liner plate fatigue analysis. In response to this RAI, Duke revised the final paragraph in Section 5.3.1 of Exhibit A of the Application (See Footnote 9). The final paragraph in UFSAR Supplement Section 3.8.1.5.3 of Exhibit B of the Application was also revised accordingly.

Confirmatory Item 4.2.1.3-1 identified additional considerations associated with the Type A tests that should be included in the UFSAR supplement to provide completeness. The final paragraph of the UFSAR supplement Section 3.8.1.5.3 will be revised as follows (revised text is underlined for emphasis):

Periodic Type A Integrated Leak rate tests are additional major sources of load changes. These Type A loads are considered within the set of design loads whose cumulative total was assumed to be 500 cycles. Seven Type A tests have been performed per unit to date (June 1998). <u>Based on the frequency of Type A tests (according to performance-based Option B of 10 CFR Part 50, Appendix J), four more tests may be performed per unit through the period of extended operation if the results of earlier tests have not shown problems. Additional Type A tests may be performed if major modifications or repairs are made to the containment pressure boundary</u>. The additional load cycles on the liner due to Type A testing are considered to be insignificant.

FIGURE 1
Illustrative Transient

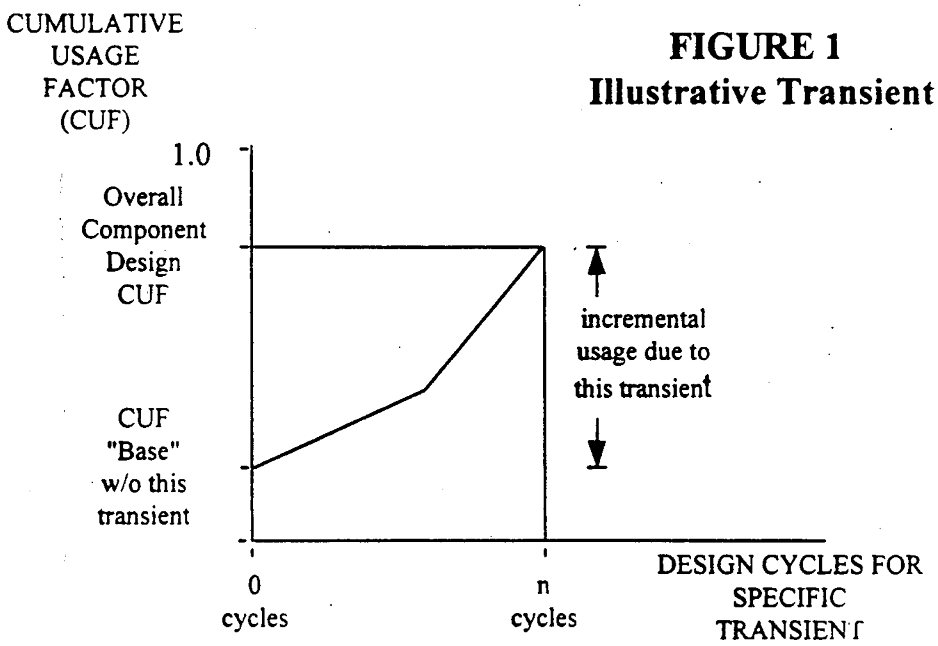
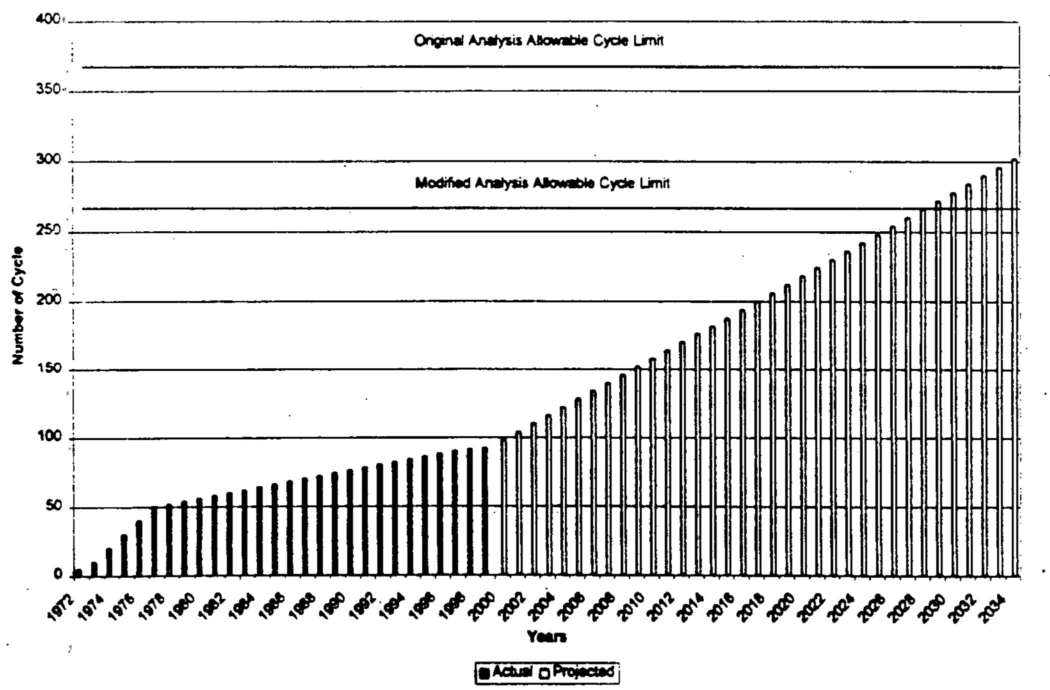


Figure 2
Illustrative Transient Cycles Projected for 60 Years



FIGURES FOR RESPONSE TO SER OPEN ITEM 4.2.3-2

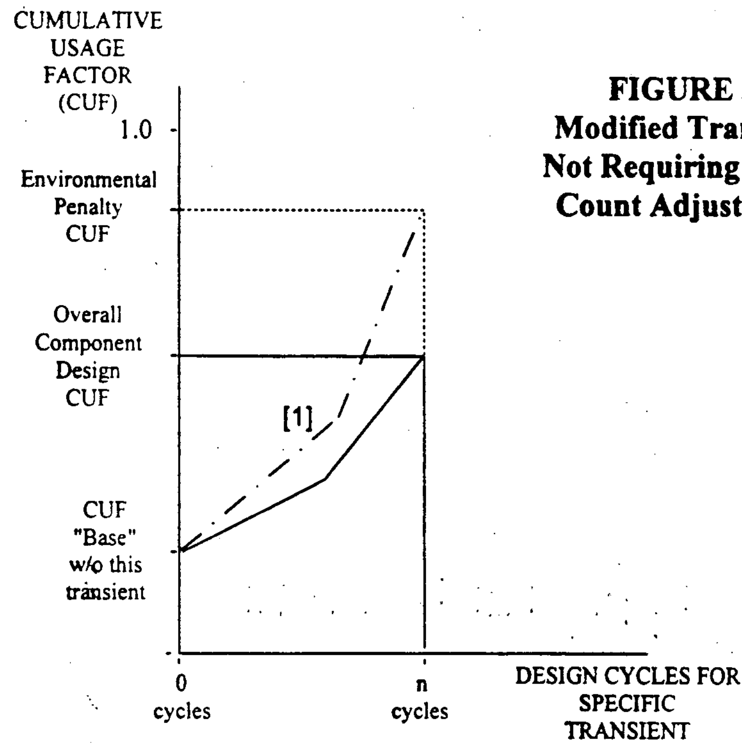


FIGURE 3
Modified Transient Not Requiring Cycle Count Adjustment

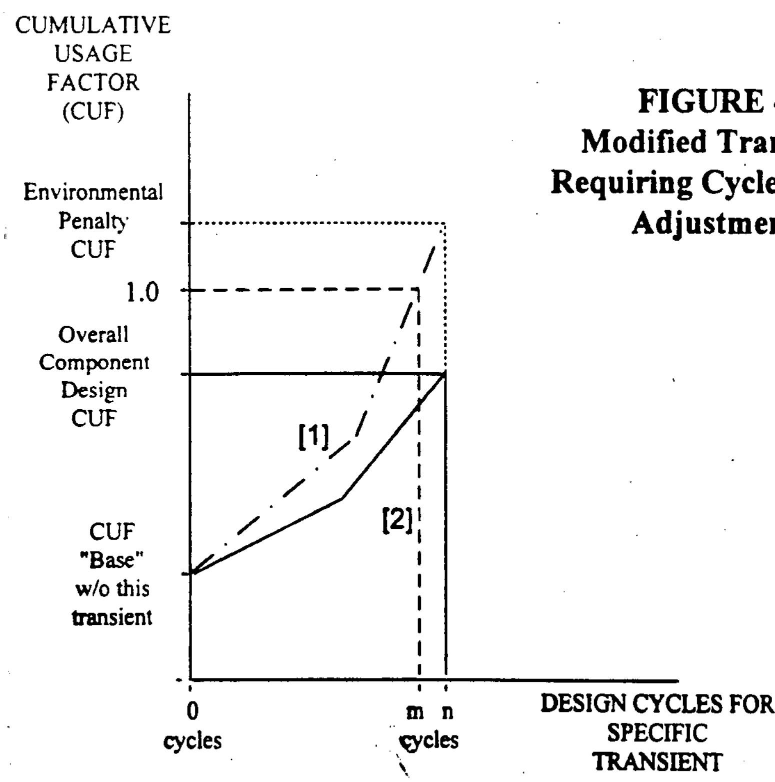


FIGURE 4
Modified Transient Requiring Cycle Count Adjustment

FIGURES FOR RESPONSE TO SER OPEN ITEM 4.2.3-2