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November 30, 1999

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Subject: License Renewal
Response to NRC Letter dated October 8, 1999
Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

By letter dated July 6, 1998, Duke Energy Corporation submitted an Application for Renewed Operating Licenses for Oconee Nuclear Station, Units 1, 2, and 3 (Application). Exhibit A of the Application contains the technical information required by 10 CFR Part 54. On June 16, 1999, the staff issued its Safety Evaluation Report (SER) related to the review of the Oconee application.

Within the SER, Open Item 2.1.3.1-1 concerns the scoping methodology used by Duke to comply with the requirements of 10 CFR §54.4. By letter dated June 22, 1999, Duke Energy provided a response to SER Open Item 2.1.3.1-1. Subsequently, the staff conducted an Oconee site visit from August 16-18, 1999, the results of which are documented in a meeting summary dated August 27, 1999.

By letter dated October 8, 1999, the staff provided its proposed plan for the resolution of SER Open Item 2.1.3.1-1. The proposed plan consists of Duke performing an assessment of an event set that is broader than the set of events that was used in the Oconee License Renewal Scoping Methodology. On October 28, 1999, Duke met with the staff to discuss its understanding of the purpose of performing the assessment and some preliminary results. The results of this meeting are contained in a staff meeting summary dated November 4, 1999. At this meeting, Duke agreed to provide the results of this assessment by the end of November. Accordingly, please find attached the results of the assessment requested by the October 8, 1999 NRC letter.

Duke believes that these results provide a validation that the NRC Staff can rely upon in making the finding that there is reasonable assurance that the Oconee License Renewal Scoping Methodology described in the Application, in various RAI and SER Open Item responses (specifically in the Duke response to SER Open Item 2.1.3.1-1 provided in the Duke letter dated June 22, 1999) has identified all systems, structures and components relied upon to remain functional to ensure the functions identified in 10 CFR §54.4.

Commitments are docketed statements that establish requirements or actions to be performed. There are no commitments contained within this letter.

If there are any questions, please contact Bob Gill at 704-382-3339.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

M. S. Tuckman, being duly sworn, states that he is Executive Vice President, Nuclear Generation Department, Duke Energy Corporation, that he is authorized on the part of said Company to sign and file with the U. S. Nuclear Regulatory Commission this supplemental response to an NRC request for additional information concerning the Application to Renew the Facility Operating Licenses of Oconee Nuclear Station submitted by letter dated July 6, 1998; and that all statements and matters set forth herein are true and correct to the best of his knowledge and belief. To the extent that these statements are not based on his personal knowledge, they are based on information provided by Duke employees and/or consultants. Such information has been reviewed in accordance with Duke Energy Corporation practice and is believed to be reliable.

M. S. Tuckman

M. S. Tuckman, Executive Vice president
Duke Energy Corporation

Subscribed and sworn to before me this 29TH day of NOVEMBER 1999.

Mary P. Nelms
Notary Public

My Commission Expires:

JAN 22, 2001

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Attachment 1

Oconee Nuclear Station
Application for Renewed Operating Licenses
Response to NRC Letter dated October 8, 1999

November 30, 1999

Attachment 1
Oconee Nuclear Station
Application for Renewed Operating Licenses
Response to NRC Letter dated October 8, 1999

November 30, 1999

OVERVIEW

The Nuclear Regulatory Commission (NRC) Staff issued the Safety Evaluation Report (SER) related to the renewal of the operating licenses of Oconee 1, 2 and 3 on June 16, 1999. Over the course of preparation of this SER, the NRC staff had questions regarding aspects of the Oconee license renewal scoping process required by 10 CFR §54.4. At the time of issuance of the SER, the scoping issue remained open and was captured as Open Item 2.1.3.1-1. To address this issue, a series of meetings, both at technical and management levels, as well as written correspondence, have occurred between Duke Energy (Duke) and the NRC Staff. Most recently, in an October 8, 1999 letter, the NRC Staff proposed a plan for the resolution of the scoping issue which could constitute the basis upon which the NRC Staff could find the results of Duke's mechanical scoping methodology acceptable. The plan involved performing an assessment of ten events outside of the set Duke had considered as a part of Oconee license renewal. Duke and NRC management met on October 28, 1999 to further clarify management expectations associated with this plan. A meeting summary was published in an NRC letter dated November 4, 1999.

At that meeting, Duke described its understanding that the purpose of performing this assessment was not to redefine the Oconee License Renewal Scoping Methodology as presented in the Oconee License Renewal Application, but to provide reasonable assurance that the methodology identifies all systems, structures and components relied upon to remain functional to ensure the functions identified in 10 CFR §54.4. At the October 28, 1999 meeting, Duke presented preliminary results of its ongoing assessment in the form of examples grouped into a number of categories. As requested by the NRC Staff, these groupings have been used in the presentation of the assessment results herein. When clarifying its management intent for the outcome of the assessment, the NRC Staff requested, for any instance where the assessment identified specifically-credited plant hardware that is outside of the scope of license renewal, that Duke discuss if and how the hardware is relied on to function in order to accomplish its committed function for the event. A specific grouping for this topic is provided in the presentation of the assessment results herein.

The NRC Staff also requested that Duke's response to the October 8, 1999 letter include links to the Oconee License Renewal Application wherever appropriate. The links to the application materials, including the Duke responses to NRC Requests for Additional Information (RAIs) and to SER Open Items, are provided in the presentation of the assessment results herein. To further clarify management expectations, the NRC Staff requested and Duke agreed to include a discussion of the additions to the scope of license renewal that have occurred since the original application submittal on July 6, 1998.

These additions were identified as part of the 10 CFR §54.21(b) annual application update and were provided in the Duke letter dated September 30, 1999. Other additions were identified as a result of responses to various safety evaluation open items and were provided in the Duke letter dated October 15, 1999. The scope additions from both Duke responses are discussed in the presentation herein.

It is Duke's conclusion that the findings of this assessment provide a validation that the NRC Staff can rely upon in making a finding that there is reasonable assurance that the Oconee License Renewal Scoping Methodology identifies all systems, structures and components relied upon to remain functional to ensure the functions identified in 10 CFR §54.4.

OCONEE LICENSE RENEWAL SCOPING METHODOLOGY

To put the results of the Duke assessment into perspective, an understanding of the Oconee License Renewal Scoping Methodology is helpful. The Oconee License Renewal Scoping Methodology contains seven separate features and does not rely solely on the scope of design basis events which are defined in Chapter 15 of the Oconee UFSAR to establish renewal scope. The seven features of the Oconee License Renewal Scoping Methodology are:

1. *Functional flow path identification* - All mechanical systems and their functions that are listed in the Oconee event mitigation calculations are included within the scope of license renewal. (The scope of these events is the subject of SER Open Item 2.1.3.1-1.) Using the same criteria as those described in § 54.4(a)(1)(i), (ii), and (iii) as the success criteria for the events, these calculations document comprehensive technical evaluations to identify all mechanical flowpaths required to fulfill or support the fulfillment of those criteria. The event mitigation calculations are Oconee site documents that are used in a manner similar to how other sites may use a "Q-list" to identify those systems and components that fulfill or support the fulfillment of those functions described in §54.4(a)(1)(i),(ii), and (iii). Further information about these calculations and about the process of functional flow path identification are contained in the Duke response to Open Item 2.1.3.1-1 provided by the Duke letter dated June 22, 1999.

2. *Fluid pressure boundary determination* - All passive pressure boundaries required for mechanical systems identified in Feature 1 above are included within the scope of license renewal.

3. *Physical interference identification* - Portions of mechanical systems whose failure to maintain their pressure boundary or to remain structurally intact would result in

impacting the function of any essential system and component (e.g. seismic II/I) are included within the scope of license renewal.

4. *Other designated item identification (safety-related, seismic)* - Mechanical systems or portions of systems that contain safety-related and seismically designed piping that have not otherwise been included in the previous three categories are included within the scope of license renewal.

5. *Structures* - All Oconee structures that are designated as either Class 1 or 2 as defined in the UFSAR are included within the scope of license renewal.

6. *Electrical Components* - All Oconee electrical components are included within the scope of license renewal, except for the electrical components in the 525 kV Switchyard, the Jocassee, Calhoun, Oconee, and Dacus 230 kV transmission lines, the Radwaste Facility and the Oconee Retail Substation. These exceptions are described in the Duke response to RAI 2.6-1 found in the Duke letter dated February 17, 1999.

7. *Five regulated programs* - All structures and mechanical systems required to demonstrate compliance with NRC regulations for events identified in 10 CFR §54.4(a)(3) are included within the scope of license renewal.

OCONEE LICENSE RENEWAL SCOPING RESULTS

The Application for Renewed Operating Licenses for Oconee Nuclear Station, Units 1, 2, and 3 (Application) was submitted to the NRC by letter dated July 6, 1998. Exhibit A of the Application contains the technical information required by 10 CFR §54.21. Chapter 2 of Exhibit A describes the first major activity of the Oconee Integrated Plant Assessment – the identification of structures and components subject to aging management review. Inherent in the identification of structures and components subject to an aging management review is the identification of the systems and structures within the scope of license renewal required by §54.4.

The methodology used to identify structures and mechanical systems at Oconee that are within the scope of license renewal is described in Section 2.2 of Exhibit A of the Application. The results of applying this methodology are provided in Sections 2.3, 2.4, 2.5 and 2.7 of Exhibit A. The methodology used to identify electrical components and the results of the application of the methodology are described in Section 2.6 of Exhibit A.

The Staff review of the Application commenced in the fall of 1998. Staff requests for additional information were made in several letters sent to Duke from October through

December 1998. Duke responses to these letters were provided in letters dated December 14, 1998, January 25, 1999, February 8, 1999, February 17, 1999, March 18, 1999, March 29, 1999, April 6, 1999, and May 10, 1999. The SER related to the license renewal of Oconee Nuclear Station was published by the Staff on June 16, 1999. Since the issuance of the SER in June 1999, Duke has submitted letters dated June 22, 1999 (which provided a partial response to SER Open Item 2.1.3.1-1), September 30, 1999 (which provided the update of the Application as required by §54.21(b)), and October 15, 1999 (which provided responses to several SER Open and Confirmatory Items and Duke comments on the SER).

As a consequence of the Staff review of the Application, the changes to the current licensing basis of Oconee and the Duke responses to SER Open Items, the original scope of systems, structures and components provided in the Application has been expanded. The following is a summary of the systems, structures and components added after the initial scoping for license renewal was performed and the reasons for their addition:

1. The Essential Siphon Vacuum System, the Siphon Seal Water System, the Essential Siphon Vacuum Trenches and the Essential Siphon Vacuum Building were added to the scope of license renewal as a result of the completion of a plant modification after the Application was submitted (Amendment 1 – CLB Changes for 1999, Duke letter dated September 30, 1999).

2. Portions of the Component Cooling System were added to the scope of license renewal as a result of a revised steam generator tube rupture analysis after the Application was submitted (Amendment 1 – CLB Changes for 1999, Duke letter dated September 30, 1999).

3. Portions of the Low Pressure Water System were added to the scope of license renewal as a result of a management decision to implement a functional change in the Reactor Building Auxiliary Coolers after the Application was submitted (Amendment 1 – CLB Changes for 1999, Duke letter dated September 30, 1999).

4. The Chilled Water System and portions of the Condenser Circulating Water System and of the Control Room Pressurization and Filtration system were added to the scope of license renewal due to a change to the Oconee current licensing basis as a result of implementing Improved Technical Specifications in 1999 (Response to SER Open Item 2.2.3.4.3.2.1-1 contained in Duke letter dated October 15, 1999).

5. Two complete systems and portions of two other systems (which other parts were already within the scope of license renewal) were added to the scope of license renewal as a result of a revised component evaluation boundary definition of the Standby Shutdown Facility (SSF) diesel generator. The Diesel Jacket Water Cooling

System and the SSF Diesel Generator Lube Oil System were added due to the revised component evaluation boundary definition. Additional components in the SSF Diesel Generator Fuel Oil System and the Starting Air System were added to the scope of license renewal due to a change of the component boundary definition from the skid connection to the engine connection (Response to SER Open Item 2.2.3.4.8.2.1-1 contained in Duke letter dated October 15, 1999).

DUKE ASSESSMENT OF THE TEN EVENTS IDENTIFIED IN THE OCTOBER 8, 1999 LETTER

In its October 8, 1999 letter, the NRC Staff asked Duke to perform an assessment of ten events not considered by Oconee to be scoping events for the purposes of license renewal. These ten events are thus not included in Duke's scoping process associated with 10 CFR §54.4 and the Oconee License Renewal Scoping Methodology. The purpose of this assessment is to provide the NRC Staff with additional assurance that the Oconee License Renewal Scoping Methodology identifies those systems, structures and components that are relied upon to remain functional to ensure the functions identified in 10 CFR 54.4.

The ten events requested to be included in the assessment are (1) high energy line break, (2) loss of decay heat removal, (3) loss of spent fuel pool cooling – heat transfer function, (4) loss of control room, (5) steam generator overfill, (6) steam generator dryout, (7) loss of instrument air, (8) internal flooding (Auxiliary Building), (9) control of heavy loads, and (10) loss of condensate.

As requested by the NRC in the October 8, 1999 letter, the review of each of the ten events is limited to a review of licensing commitments in five defined document sets that are a subset of the Oconee current licensing basis. The five defined document sets are (a) the Oconee UFSAR, (b) license conditions, (c) Commission orders, (d) Commission regulations, and (e) exemptions granted by the Commission.

The process to review the ten events was performed by members of the Oconee License Renewal Team. The Oconee current licensing basis was reviewed to determine how each of the ten events may be identified within each of the five defined document sets. For each event, the document sets containing any reference to the event were reviewed and an evaluation of the Oconee response to that event was compiled and analyzed. The compiled Oconee response to each event was then reviewed by an expert panel consisting of both Oconee engineering and licensing experts in order to confirm the adequacy of the review and its results. This review identified the specific plant capabilities relied on for these ten events as described in the five defined document sets. The results of this review are presented herein.

RESULTS OF THE DUKE ASSESSMENT

A review of the results of the Duke assessment has identified four groupings among the ten events that helps put the results into perspective. These groupings were discussed at the October 28, 1999 management meeting between Duke and the NRC Staff, and the assessment results here will be provided by these groupings. The four groupings are: (i) events not described in the five defined document sets, (ii) events that solely credit plant systems, structures and components that are already within the scope of license renewal, (iii) events that credit only operator action and mention non-specific plant hardware, and (iv) events that credit plant hardware in addition to systems, structures and components already within the scope of license renewal.

(i) Events Not Described In The Five Defined Document Sets

In this first group, Duke's assessment revealed three events identified in the Staff's October 8, 1999 letter that were not described in the five defined document sets. The control of heavy loads, the loss of condensate and internal flooding (Auxiliary Building) are not described in the current licensing basis information contained in the Oconee UFSAR, plant license conditions, Commission orders, Commission regulations, and exemptions granted by the Commission.

While these three events could be excluded from the scope of this review, an additional perspective can be offered related to the control of heavy loads and internal flooding in the Turbine Building. Because of the association of heavy loads with cranes, Duke did note that, though the mechanical portion of the scoping process did not consider control of heavy loads to be included, the structural screening process completed as a part of the integrated plant assessment did capture crane rails and girders in Application Section 2.7. These non-safety related components were included since their failure could prevent satisfactory accomplishment of a required safety related function.

Internal flooding related to the Auxiliary Building is not discussed in the five defined document sets. Internal flooding is discussed in the Oconee UFSAR related to the Turbine Building. Mitigation of an internal flood in the Turbine Building does involve the exterior of a common wall shared between the Turbine Building and the Auxiliary Building. This common wall was identified as being within the scope of license renewal as part of the Auxiliary Building in Application Section 2.7. One of its identified functions was to provide a protective barrier for such an internal flood event.

(ii) Events Solely Crediting Plant SSCs Already Within The Scope of License Renewal

In the second group, five events solely credit plant systems, structures and components that were already within the scope of license renewal. These events include high energy

line break, loss of control room, steam generator overfill, steam generator dryout, and loss of instrument air.

High energy line break is discussed in the five defined document sets. The Oconee UFSAR references a study performed as a result of a 1972 request from the predecessor of the NRC to provide analyses and other relevant information needed to determine the consequences of a postulated pipe failure outside Reactor Building containment. This event is not described in the current licensing basis information contained in plant license conditions, Commission orders, Commission regulations, or exemptions granted by the Commission. The 1972 Duke study includes references to non-specific shutdown paths as well as plant modifications required to comply with the NRC request. For the purposes of this assessment, the references to non-specific shutdown paths were not areas of further focus in a manner similar to the other event falling within category (iii) herein. The specific plant modifications resulting from the 1972 study were further examined in this assessment. The hardware associated with each of the modifications was previously identified as being within the scope of license renewal and is included in the Application by several of the seven scoping methodology features previously described herein. The hardware associated with the modifications was included in license renewal as a result of the mechanical scoping process described in Application Section 2.5, the structures screening process described in Application Section 2.7, and the electrical scoping feature described, among other places, in the Duke response to RAI 2.6-1 found in the Duke letter dated February 17, 1999.

Loss of control room is discussed in the five defined document sets. The Oconee UFSAR describes the capabilities of the control room in several sections. This event is not described in the current licensing basis information contained in plant license conditions, Commission orders, Commission regulations, or exemptions granted by the Commission. The Oconee UFSAR describes the non-safety related auxiliary shutdown panel as a means to shut down the plant and to maintain it in a safe shutdown condition in the event of loss of the control room. The auxiliary shutdown panel was installed at the time of original Oconee construction. Since original licensing and operations began, the safety-related Standby Shutdown Facility has been added to the plant, in part, to mitigate the consequences of fire in the control room. The Standby Shutdown Facility makes available an alternate means to attain hot shutdown in the event of loss of control room.

Though the capability of the auxiliary shutdown panel is mentioned in the defined document sets, no specific requirement exists for the operator to use the auxiliary shutdown panel in the event of loss of control room. Further, no specific requirement exists to have two methods of attaining hot shutdown in the event of loss of control room. The ability to maintain hot shutdown via the auxiliary shutdown panel was not identified by the Oconee License Renewal Scoping Methodology. However, the physical portions of the auxiliary shutdown panel, including the panel itself along with the instrumentation

and cabling are included within the scope of license renewal by other features of the scoping methodology. The structural portions of the panel are included in Application Section 2.7, while the electrical portions of the panel are included by the electrical component scoping feature described, among other places, in the Duke response to RAI 2.6-1 found in the Duke letter dated February 17, 1999. The Standby Shutdown Facility contains mechanical systems, structures and electrical components that were also identified as being within the scope of license renewal in various sections of the Application.

It should be noted that the status of the auxiliary shutdown panel's relationship to the Application has changed from the preliminary results presented by Duke at the October 28, 1999 meeting and described in the November 4, 1999 NRC written summary of this meeting. This change was needed since the panel was incorrectly noted as being excluded from the scope of license renewal in the Duke presentation materials.

Steam generator overfill is discussed in the five defined document sets. The Oconee UFSAR describes how overfill of water to the steam generator that results in a high water level will automatically trip the main feedwater pumps and subsequently initiate the Emergency Feedwater System. This event is not described in the current licensing basis information contained in plant license conditions, Commission orders, Commission regulations, or exemptions granted by the Commission. The instrumentation associated with the high water level indication as well as the electrical initiation circuitry associated with this event has been included within the scope of license renewal via the electrical component scoping feature described, among other places, in the Duke response to RAI 2.6-1 found in the Duke letter dated February 17, 1999. The Emergency Feedwater System was identified as being within the scope of license renewal in Application Section 2.5.9.3.

Steam generator dryout is discussed in the five defined document sets. The Oconee UFSAR describes how a low level indication associated with dryout in the steam generators will automatically initiate the Emergency Feedwater System. This event is not described in the current licensing basis information contained in plant license conditions, Commission orders, Commission regulations, or exemptions granted by the Commission. The low-level initiation electrical circuitry associated with this event has been included within the scope of license renewal via the electrical component scoping feature described, among other places, in the Duke response to RAI 2.6-1 found in the Duke letter dated February 17, 1999. The Emergency Feedwater System was identified as being within the scope of license renewal in Application Section 2.5.9.3.

Loss of instrument air is described in the five defined document sets. The Oconee UFSAR describes specifically how the Low Pressure Service Water System fulfills its function following a loss of coolant accident where a loss of offsite power leading to the

loss of instrument air is also assumed to occur. The loss of instrument air event is not described in the current licensing basis information contained in plant license conditions, Commission orders, Commission regulations, or exemptions granted by the Commission. The specific Low Pressure Service Water System functions and hardware are included within the scope of license renewal and identified in Application Section 2.5.6.5. The UFSAR also includes loss of instrument air as a typical situation addressed by emergency procedures, but does not mention specific plant capability credited for the event. For the purposes of this assessment, the references to non-specific shutdown paths were not areas of focus in a manner similar to the other event falling within category (iii) herein.

The consideration of a loss of instrument air function is included within the Oconee License Renewal Scoping Methodology by virtue of the fact that the Instrument Air System is assumed to be lost coincident with other evaluated events. The Instrument Air System loses function any time a loss of offsite power is experienced. With regard to the components within the Instrument Air System, the portion of the system that passes through the Reactor Building containment boundary was identified as being within the scope of license renewal in the Application Section 2.5. The appropriateness of this scope for the Instrument Air System was confirmed by the onsite NRC license renewal inspection documented in NRC Inspection Report 50-269/99-11, 50-270/99-11 and 50-287/99-11.

(iii) Events Crediting Only Operator Action and Mentioning Non-specific Plant Hardware

In a third group, one event – loss of spent fuel pool cooling – credits only operator action and mentions non-specific plant hardware as part of the plant capabilities.

Loss of spent fuel pool cooling is discussed in the five defined document sets. The Oconee UFSAR describes the results of an analysis of loss of spent fuel pool cooling. This event is not described in the current licensing basis information contained in plant license conditions, Commission orders, Commission regulations, or exemptions granted by the Commission. During normal operation, pool heat is removed by the Spent Fuel Pool Cooling System which, in turn, is cooled by the non-safety related Recirculating Cooling Water System. In the event of the loss of spent fuel pool cooling, only operator actions and non-specific sources of water to restore pool inventory are credited to restore the pool inventory, to keep the fuel covered and to preclude releases that would exceed Part 100 guideline values. The operator would depend on spent fuel pool level instrumentation to manage pool level.

Although only operator actions and non-specific sources of water are credited to restore pool inventory, the spent fuel pool itself and the spent fuel racks within the pool must remain in tact for these actions to be effective. The spent fuel pool was identified as,

being within the scope of license renewal in Application Section 2.7.3. The spent fuel racks within the pool were also identified as being within the scope of license renewal in Application Section 2.7. All instrumentation required to monitor pool level is within the scope of license renewal via the electrical component scoping feature described, among other places, in the Duke response to RAI 2.6-1 found in the Duke letter dated February 17, 1999.

(iv) Events Crediting Plant Hardware In Addition To SSCs Within The Scope of License Renewal.

In the fourth group, one event – loss of decay heat removal – credits plant capabilities in addition to systems, structures and components already within the scope of license renewal.

Loss of decay heat removal is discussed in the five defined document sets. The Oconee UFSAR as well as the license conditions described in the plant technical specifications describe the actions required to mitigate loss of decay heat removal. This event is not described in the current licensing basis information contained in Commission orders, Commission regulations, or exemptions granted by the Commission. The normal action to mitigate loss of decay heat removal is operator action to restore the operation of the safety-related Low Pressure Injection System - the Oconee system that provides decay heat removal capabilities. The Low Pressure Injection System was identified in Application Section 2.5.5.3 as being within the scope of license renewal.

In response to NRC Generic Letter 88-17, "Loss of Decay Heat Removal," a number of administrative and programmatic actions, as well as one plant hardware addition, were implemented at Oconee. The hardware component consists of an instrument that is installed during certain shutdown modes to assist the operator in detecting low inventory levels in the Reactor Coolant System so that he can take appropriate corrective action, such as that mentioned above, to restore the Low Pressure Injection System. This instrument is stored in a warehouse when not in use and is verified to be functional when it is calibrated prior to being placed into service. The electrical component scoping feature described, among other places, in the Duke response to RAI 2.6-1 found in the Duke letter dated February 17, 1999 includes all instrumentation and associated electrical lines within the scope of license renewal. A related issue on equipment within the scope of license renewal and normally stored in the warehouse is discussed in Duke response to SER Open Item 2.2.3.7-2 provided in the Duke letter dated October 15, 1999. The mechanical fluid pressure boundary determination scoping feature also includes the Reactor Coolant System tubing associated with this instrument within the scope of license renewal in Application Section 2.5.

One of the administrative actions associated with the Duke response to NRC Generic Letter 88-17 was to implement several commitments to further reduce the risk of loss of decay heat removal. These commitments are described in the Oconee UFSAR. They involve operator actions to establish and maintain specific plant conditions. The hardware associated with establishing and maintaining these conditions includes the Containment structural components, components in the Reactor Coolant System, mechanical system containment penetrations and electrical power and instrumentation components. The Containment structural components were identified as being within the scope of license renewal in Application Section 2.3. The components in the Reactor Coolant System were identified as being within the scope of license renewal in Application Section 2.4. Components associated with mechanical system containment penetrations were identified as being within the scope of license renewal in Application Sections 2.3 and 2.5. Electrical power and instrumentation components were identified as being within the scope of license renewal via the electrical component scoping feature described, among other places, in the Duke response to RAI 2.6-1 found in the Duke letter dated February 17, 1999.

One other commitment is related to adding inventory to the Reactor Coolant System. Specifically, three additional means of adding inventory to the Reactor Coolant System were identified. First, a gravity flow path from the borated water storage tank in the Low Pressure Injection System was identified. Second, the capability for adding inventory via a high pressure injection pump in the safety-related High Pressure Injection System was identified. Third, inventory addition via the use of a non-safety related reactor coolant bleed transfer pump and connecting piping was identified. The gravity flow path from the borated water storage tank and the high pressure injection pump are safety-related and were thus identified as being within the scope of license renewal in Application Section 2.5. The non-safety related reactor coolant bleed transfer pump and connecting piping were not identified by the Oconee License Renewal Scoping Methodology and are not included in the Application.

CONCLUSIONS FROM DUKE ASSESSMENT

Duke's assessment of the ten events identified in the October 8, 1999 letter revealed only one instance where specific plant hardware falling outside the scope of license renewal was identified as a result of the implementation of the methodology contained in the Staff's October 8, 1999 letter. For three of the events, no information exists in the five defined document sets. For five of the events, the five defined document sets credit only plant systems, structures and components that are already within the scope of license renewal. For one event, the five defined document sets credit only operator action and mention non-specific plant capabilities.

The one event that mentions specific hardware not within the scope of license renewal is the loss of decay heat removal. For this event, the Oconee UFSAR credits the availability of the non-safety related reactor coolant bleed transfer pumps and connecting piping in addition to two other safety-related systems and components as means of adding inventory to the Reactor Coolant System. The two safety-related systems and components are within the scope of license renewal.

A review of the UFSAR discussion indicates that the reactor coolant bleed transfer pumps and connecting piping are not uniquely relied upon to function in order to add inventory to the Reactor Coolant System should a loss of decay heat removal occur. Rather, each of the three means of adding inventory to the Reactor Coolant System serves as an alternative available to the operator. Since this non-safety hardware alternative is not uniquely relied upon to function and specifically designated components that were within the scope of license renewal had already been identified, no further investigation was warranted into how it would accomplish its function.

The overall plan for resolution of the scoping issue related to the NRC review of the Oconee Application involved performing an assessment of ten events outside of the set of events Duke had considered as input to the Oconee License Renewal Scoping Methodology. The purpose of this assessment was to provide the NRC Staff with additional assurance that the Oconee License Renewal Scoping Methodology identified all systems, structures and components relied upon to remain functional to ensure the functions identified in 10 CFR §54.4. The results of the assessment provided herein identified no hardware uniquely relied on to perform a function associated with any of the ten events that was not already within the scope of license renewal. Duke believes that these results provide a validation that the NRC Staff can rely upon in making the finding that there is reasonable assurance that the Oconee License Renewal Scoping Methodology described in the Application, in various RAI and SER Open Item responses (specifically in the Duke response to SER Open Item 2.1.3.1-1 provided in the Duke letter dated June 22, 1999) has identified all systems, structures and components relied upon to remain functional to ensure the functions identified in 10 CFR §54.4.

It is Duke's view that these materials previously provided on the docket contain sufficient information to describe the scoping methodology in the SER. This SER description should, in turn, be sufficient to provide a technical basis from which the NRC Staff can conclude that there is reasonable assurance that the Oconee License Renewal Scoping Methodology identified those SSCs that are relied upon to remain functional to ensure the functions identified in 10 CFR §54.4. The validation of the Oconee license renewal scoping results by this assessment can serve to support this conclusion.

Duke's SER Comments From 10/15/99 Submittal

Attachment 1
Comments on the Safety Evaluation Report Related to the License Renewal Application of Oconee Nuclear Station, Units 1, 2, and 3 (SER) June 1999

October 15, 1999

1. Clarify Basis for Program Evaluation Conclusions

Duke used one set of attributes to evaluate aging management programs and activities. This set of attributes was derived from several sources as described in Section 4.2 of Exhibit A of the Application. The staff in its review uses a different set of attributes to evaluate the aging management programs and activities. The link between the two sets of attributes is not so apparent. In its conclusion for each program or activity reviewed, the staff typically states that the applicant has demonstrated (emphasis added) that the program or activity is effective at managing the aging effects of concern. It would seem then that the fundamental basis for the staff conclusion on the specific program is the information provided by Duke in the Application and in responses to staff requests for additional information.

Since the program attributes in the Application form the basis of the current UFSAR Supplement (Exhibit B of the Application), a clearer link between the staff review and Duke attributes would allow the UFSAR Supplement to be maintained in a manner that avoids differing interpretations in the future.

2. Revise Pressurized Thermal Shock Discussion for Oconee Unit 2

The Unit 2 reactor vessel pressurized thermal shock discussion in SER Section 4.2.4.3.3 does not contain the most current results. Duke letter dated February 17, 1999, Attachment 1, response to RAI 5.4.2-1 provides the most current results based on materials data from B&W Owners Group Topical Report BAW-2325. The RT PTS value for Unit 2 is 296.8 °F. In addition, because the value is now below the PTS screening criterion, Duke withdrew the commitment that is described in the SER on page 4-19. Attachment 7 of the above letter, Item #8, clearly states that the commitment has been withdrawn.

3. Discuss Leak-Before-Break Evaluation in SER Section 4.2

In response to RAI 5.4.1-1, Duke provided a substantial discussion of leak-before-break (LBB) for Oconee. Subsequent to the submittal of the Application in July 1998, Duke identified LBB as a TLAA and provided the results of the evaluation in the above response. Section 4.2 does not include any mention of LBB.

4. Clarify Administrative Controls for Preventive Maintenance Activities

Section 3.2.10.3 of the SER evaluates the elements of the Preventive Maintenance Activities collectively. In particular, the section states that the corrective actions, confirmation process, and administrative controls for these activities are in accordance with the site quality assurance plan pursuant to 10 CFR Part 50, Appendix B. As noted in Inspection Report IR 99-12, Section E.8.3.p.1, many of the activities in the Preventive Maintenance Activities program are performed on non-safety equipment and are not controlled by the site quality assurance program. The corrective actions, confirmation process, and administrative controls elements for the Preventive Maintenance Activities should be evaluated for the individual activities given the information provided on these activities in response to RAI 4.3.8-1. For more information on

control of non-safety equipment within license renewal aging management programs, see response to SER Open Item 3.2.3.3-1

5. Clarify Discussion of Auxiliary Service Water

At the time the technical documentation was being developed to support the Oconee License Renewal Application, there was a misunderstanding about the normal system alignment of the Auxiliary Service Water (ASW) System. This system is normally in a standby mode with parts of the system normally wetted and other parts normally dry and exposed to an internal air environment. This misunderstanding caused the environment/ aging effect/ aging management program information in the Application to be misconstrued. The ASW System is depicted on OLRFD 121D-1.2. The portion of the system upstream of valve CCW-101 is normally exposed to raw water. This portion of the system contains pipe, valves, tubing, an annubar tube, and the ASW pump. The portion of the system downstream of CCW-101 is normally exposed to air because drain valve CCW-309 is normally open. At one time, the portion of the system now exposed to air did contain stagnant, raw water, but a change in valve alignment a few years ago to leave CCW-309 normally open caused this portion of the system to be drained and exposed the internal environment of the piping to air. This portion of the system contains pipe and valves.

5.1 Raw Water Portion of the System

The aging effects in the raw water portion of the system are loss of material and fouling. Fouling is managed by System Performance Testing Activities, as stated in Application Section 3.5.6.2.3 and Table 3.5-4. Table 4.3-1 of the Application incorrectly lists fouling as being managed by the *Auxiliary Service Water Piping Inspection* of the Preventive Maintenance Activities. Inclusion of this program name in the table is an error and should be disregarded. System Performance Testing Activities alone are credited with managing fouling in the raw water portion of the Auxiliary Service Water System.

Loss of material in the raw water portion of the system is managed by the Service Water Piping Corrosion Program, as stated in Application Section 3.5.6.2.3 and Table 3.5-4. It was noted in Inspection Report 99-12 that the Service Water Piping Corrosion Program inspection location in this system is located at the discharge of valve CCW-101, not at the pump discharge check valve. At the time the inspection location was chosen, the location was expected to be a susceptible location because the piping did contain stagnant, raw water. Since the valve realignment, however, the piping now normally contains air and the inspection location is no longer useful.

As pointed out in the Application, the Service Water Piping Corrosion Program is not a system-specific program and performs ultrasonic testing across a number of raw water systems at sample locations of various flow regimes. The program does contain inspection location points of carbon steel piping in stagnant portions of other raw water systems which will provide information to allow aging management of the similar materials in the raw water portion of the

Auxiliary Service Water System. The staff review of the Service Water Piping Corrosion Program is provided in Section 3.2.13 of the Safety Evaluation Report. Based on its review, the staff concluded that, pending acceptable resolution of the Open Items identified, the applicant has demonstrated that the Service Water Piping Corrosion Program will adequately manage the aging effects associated with a loss of material from corrosion for those systems that have components exposed to a raw water environment for the period of extended operation. Duke believes the staff's conclusion supports the concept of the Service Water Piping Corrosion Program covering the full range of materials and flow regimes in the raw water systems. Elimination of this ASW location does not invalidate this concept , since similar materials and flow regimes are covered elsewhere within the program.

5.2 AIR PORTION OF THE SYSTEM

The aging effect in the air portion of the system is loss of material. The Preventive Maintenance Activity that is credited in Application Section 3.5.6.2.3 and Table 3.5-4 and expanded upon in response to RAI 4.3.8-1 (12/14/98) as the *Auxiliary Service Water Piping Inspection* was credited for managing loss of material of the components exposed to air. The activity involves a visual inspection of the interior of the pipe when check valve CCW-100 is disassembled. The RAI response reads, "Conditions at this location make it a leading indicator of the condition of the piping that is normally opened to atmosphere downstream of the closed pump discharge isolation valve."

As can be seen from OLRFD 121D-1.2, the location of CCW-100 is not in the portion of the system normally exposed to air. The visual inspection, therefore, does not provide any additional oversight related to the air portion of the system than that which the *Service Water Piping Corrosion Program* provides. The loss of material of the components in an air environment are applicable because of the components' possible exposure to moisture from the raw water portion of the system. Therefore, the aging effects are expected to be most prominent in the raw water portion of the system and sampling those locations would serve as a leading indicator of the air portion of the system. The Service Water Piping Corrosion Program can therefore be credited with managing aging of all system components, both those exposed to raw water and to the less susceptible ones exposed normally to an air environment.

The original commitment to perform the *Auxiliary Service Water Piping Inspection* as part of the Preventive Maintenance Activities is withdrawn. Duke has determined that the *Service Water Piping Corrosion Program* is effective in managing loss of material for the components in the Auxiliary Service Water System.

The following is a revised excerpt of Table 3.5-4 of the Application that shows the component/ material/ environment/ aging effects/ aging management program combinations for the entire Auxiliary Service Water System.

Updated Portion of Application Table 3.5-4 for the Auxiliary Service Water System

Mechanical Component	Material	Internal Environment	Applicable Aging Effects	Aging Management Program/Activity
Auxiliary Service Water System				
Annubar Tube	Stainless Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Fouling	System Performance Testing Activities
Pipe	Carbon Steel	Air	Loss of Material	Service Water Piping Corrosion Program
Pipe	Carbon Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection
			Fouling	System Performance Testing Activities
Pipe	Stainless Steel	Air	Loss of Material	Service Water Piping Corrosion Program
Pipe	Stainless Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Fouling	System Performance Testing Activities
Pump Casing	Cast Iron	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection Cast Iron Selective Leaching Inspection
Tubing	Stainless Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Fouling	System Performance Testing Activities
Valve Bodies	Carbon Steel	Air	Loss of Material	Service Water Piping Corrosion Program
Valve Bodies	Carbon Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection
			Fouling	System Performance Testing Activities
Valve Bodies	Stainless Steel	Air	Loss of Material	Service Water Piping Corrosion Program
Valve Bodies	Stainless Steel	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Fouling	System Performance Testing Activities

6. Clarify Discussion of Cast Austenitic Stainless Steel (CASS)

Section 3.4.3.3 (starting on page 3-110) of the SER discusses embrittlement of CASS reactor vessel internals items. This discussion needs to be revised to reflect the discussion held between Duke and the staff during a meeting on August 24, 1999 and as provided in our response to SER Open Item 3.4.3.3-5.

7. Revise the Evaluation of the Chemistry Control Program

The staff's evaluation of the Chemistry Control Program is contained in Section 3.2.2 of the SER. The staff's evaluations of the SSF Fuel Oil Surveillances, which are described in Section 4.6.5 of Exhibit A of the Application, are not included in its SER.

8. Revise the Description of the "Technical Information for Identifying Systems, Structures, and Components within the Scope of License Renewal"

The staff's description of the "Technical Information for Identifying Systems, Structures, and Components within the Scope of License Renewal" is presented in Section 2.1.2.1 of the SER. This section does not include the description of several important features of the Oconee scoping methodology. These features are described in Section 2.2 of Exhibit A of the Application. Additional information was provided in Duke letter dated March 18, 1999 and June 22, 1999 relative to the mechanical system scoping features. As a convenience to the staff, these important features are summarized below:

1. All mechanical systems and their functions that are listed in Oconee event mitigation calculations are included within the scope of license renewal. (The scope of these events is the subject of SER Open Item 2.1.3.1-1.)
2. All passive pressure boundaries required for mechanical systems identified in Feature 1 above are included within the scope of license renewal.
3. Portions of selected mechanical systems whose failure to maintain their pressure boundary or to remain structurally intact would result in impacting the function of any essential system and component (seismic II/I) are included within the scope of license renewal.
4. Mechanical systems or portions of systems that contain safety-related and seismically designed piping that have not otherwise been included are included within the scope of license renewal.
5. All Oconee structures that are designated as either Class 1 or 2 are included within the scope of license renewal.
6. All Oconee electrical components are initially assumed to be within the scope of license renewal.
7. All structures and mechanical systems required to demonstrate compliance with NRC regulations for events identified in §54.4(a)(3) are included within the scope of license renewal.

Because the staff's description of the entire Oconee scoping methodology requires revision as described above, it follows that its evaluation presented in Section 2.1.3 of the SER, "Evaluation of the Methodology for Identifying Systems, Structures and Components within the Scope of License Renewal" likewise needs to be revised to include its evaluation of these important features of the Oconee scoping methodology.

9. Verify the Appropriateness of Specifically Referencing Documents that are not Part of the Application

Several Oconee engineering documents were reviewed by the staff in support of its review of the Oconee Application that was submitted in July 1998. These engineering documents are specifically referenced in Sections 2.1.2.2 and 4.2.8.3 of the SER, yet they are not docketed. Duke believes that they should not be specifically referenced in the SER as to do so would seem to imply that the documents are part of the Application, which they are not. The staff should verify the appropriateness of including these specific references in the Oconee SER.

10. Revise Discussion of Class E Piping Supports

Class E pipe supports were inadvertently omitted from the scope of license renewal during development of the substantiating information for the application. The application incorrectly states that Class E piping is not within the scope of license renewal. The information should be changed to include Class E pipe supports which are required for seismic structural integrity. The first complete paragraph on Page 2.7-7 of Exhibit A of the Application should be deleted and the discussion concerning Class E piping supports in the second paragraph on Page 2.7-7 of the Application should be revised to read as follows:

Duke Class E, G and H piping supports may be assigned QA Condition 4 to denote requirements for seismic structural integrity to prevent adverse interactions with safety related systems, structures, and components. Duke Class E, G and H pipe supports, which are QA Condition 4, are within license renewal scope.

Duke's Proposed Reactor Vessel Internals Aging Management Program Response

REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM

The purpose of the Oconee *Reactor Vessel Internals Aging Management Program* is to further characterize the applicable aging effects associated with reactor vessel internals items in order to determine whether an ongoing aging management program for the reactor vessel internals items is necessary. As described in BAW-2248, the determination of applicable aging effects was qualitatively assessed based on operating conditions and operating experience. Further understanding of these aging effects is needed and is the focus of this program. The Oconee *Reactor Vessel Internals Aging Management Program* consists of the following four major activities:

- Investigation of Applicable Aging Effects,
- Analyses,
- Inspections, and
- Reports

INVESTIGATION OF APPLICABLE AGING EFFECTS

The Oconee *Reactor Vessel Internals Aging Management Program* includes the investigation of applicable aging effects for bolting, CASS Reactor Vessel Internals (RVI) items, and RVI items other than bolts. The scope of the investigation includes the following applicable aging effects: (1) cracking due to irradiation assisted stress corrosion cracking, (2) reduction of fracture toughness due to irradiation embrittlement and thermal embrittlement, (3) dimensional changes due to void swelling and (4) loss of closure integrity due to stress relaxation.

Duke Energy will participate in the B&W Owners Group Reactor Vessel Internals Aging Management Program and other industry programs, as appropriate, to continue the investigation of applicable aging effects for RVI items (Renewal Applicant Action Item 4.1 (Item 4) in the Safety Evaluation Report for BAW-2248A.)

ANALYSES

The Oconee *Reactor Vessel Internals Aging Management Program* includes several analyses. These analyses will be performed to determine the number of baffle bolts that must remain intact in order to maintain functionality of the reactor vessel internals. Analyses will also be performed to determine critical crack sizes for RVI items other than bolts within the reactor vessel internals. These analyses will address the topics of interest described in SER Open Item 4.2.5.3-1 and Renewal Applicant Action Item 4.1 (Item 12) in the Safety Evaluation Report for BAW-2248A. Finally, analyses will be performed to determine the critical crack sizes of the CASS reactor vessel internals items. Critical locations for baffle bolting, CASS RVI items, and RVI items other than bolts will also be determined by these analyses.

Duke Energy will participate in the B&W Owners Group Reactor Vessel Internals Aging Management Program and other industry programs, as appropriate to establish monitoring and

inspection programs for RVI items (Renewal Applicant Action Item 4.1 (Item 4) in the Safety Evaluation Report for BAW-2248A.)

The above activities will be used in the preparation of the Reactor Vessel Internals inspections described below. Current plans are to complete the above investigations and analyses at least three years prior to the outages in which the inspections are to be performed. Timely completion of these activities will allow the inspection plan to be finalized and the inspection to occur as committed.

INSPECTIONS

The Oconee *Reactor Vessel Internals Aging Management Program* includes the following three interrelated inspections:

1. Baffle Bolt Inspection
2. Inspection of Cast Austenitic Stainless Steel (CASS) Reactor Vessel Internals(RVI) Items
3. Inspection of Non-CASS RVI Items

The purpose of the *Reactor Vessel Internals - Baffle Bolt Inspection* is to assess the condition of the baffle bolts in order to confirm the required number of baffle bolts remain functional, ensuring the functionality of the reactor vessel internals. Activities are currently in progress to develop and qualify the inspection method. If the results of the inspection are not acceptable, then the results may be used as a baseline inspection for establishing a longer-term programmatic action to inspect or replace baffle bolts in all three Oconee units.

The purpose of the *Reactor Vessel Internals - Inspection of CASS RVI Items* is to ensure that the reduction of fracture toughness properties of these items will not result in loss of the component intended functions of the reactor vessel internals during the period of extended operation. If the results of the inspection are not acceptable, then the results may be used as a baseline inspection for establishing a longer-term programmatic action to inspect the CASS RVI items in all three Oconee units. The reactor vessel internals items fabricated from CASS include control rod guide tube spacers, vent valve bodies, Unit 3 outlet nozzles, and incore guide tube assembly spiders. The vent valve retaining rings, fabricated from martensitic stainless steel, are also included in this inspection.

The purpose of the *Reactor Vessel Internals - Inspection of Non-CASS RVI Items* is to assess the condition of the non-CASS items (e.g. plates, forgings, welds, core barrel bolts, and thermal shield bolts) in order to ensure aging effects are not adversely affecting the functionality of the reactor vessel internals. Activities are currently in progress to develop and qualify the inspection method. If the results of the inspection are not acceptable, then the results may be used as a baseline inspection for establishing a longer-term programmatic action to inspect non-CASS RVI items in all three Oconee units.

Collectively, these inspections will provide additional assurance that the Oconee reactor vessel internals will remain functional through the period of extended operation. These three inspections are described on the following pages.

All three Oconee unit reactor vessel internals are essentially the same design, number of years of operation and fuel management designs. Current plans are to perform the above inspections on one unit near the end of the 4th 10-year inservice inspection interval. If the results of these inspections indicate that inspections of the other two Oconee unit reactor vessel internals are necessary or that additional inspections of the affected unit are necessary, then the inspections will be performed at the most optimal time during the 5th 10-year inservice inspection interval. This current plan for reactor vessel internals inspection is subject to revision as a result of changes in Oconee unit operation, results of inspections at other plants, or any other factors not currently anticipated.

REPORTS

Duke Energy will provide to the NRC periodic updates (after completion of significant milestones) on the status of the Oconee *Reactor Vessel Internals Aging Management Program*, commencing within one year after the issuance of the renewed license. (Renewal Applicant Action Item 4.1 (Item 4) in the Safety Evaluation Report for BAW-2248A.) A final report will be submitted to the NRC upon completion of the reactor vessel internals inspections on the first Oconee unit.

REACTOR VESSEL INTERNALS – BAFFLE BOLT INSPECTION

Purpose – The purpose of the *Reactor Vessel Internals – Baffle Bolt Inspection* is to assess the condition of the baffle bolts in order to confirm the required number of baffle bolts remain functional, ensuring the functionality of the reactor vessel internals.

Scope – The scope of this inspection consists of the reactor vessel internals baffle bolts for Oconee Units 1, 2 and 3.

Aging Effects – The aging effects of concern are (1) cracking due to irradiation assisted stress corrosion cracking, (2) reduction of fracture toughness due irradiation embrittlement, and (3) dimensional changes due to void swelling.

Method – Current plans are to perform a volumetric inspection of the baffle bolts of one Oconee unit. Activities are in progress to develop and qualify the inspection method.

Sample Size – The sample size for the one Oconee unit inspection will be a selected number of baffle bolts determined as part of the development of the inspection method.

Industry Codes or Standards – No code or standard currently exists to guide or govern this inspection.

Frequency – The *Reactor Vessel Internals – Baffle Bolt Inspection* is a one-time inspection.

Acceptance Criteria or Standard – Any detectable crack indication is unacceptable for a particular bolt. The number of bolts needed to be intact and their locations will be determined by analysis. Acceptance criteria for dimensional changes due to void swelling will be developed prior to the inspection

Corrective Action – The need for subsequent examinations will be determined after the results of the initial examination are available. If the results of the inspection are not acceptable, then the results may be used as a baseline inspection for establishing a longer-term programmatic action to inspect or replace baffle bolts in all three Oconee units. Specific corrective actions will be implemented in accordance with the *Duke Quality Assurance Program.*

Timing of New Program or Activity – Following issuance of the renewed operating licenses for Oconee Nuclear Station, this inspection will be performed on one unit near the end of the 4th 10-year inservice inspection interval.

Administrative Controls – The *Reactor Vessel Internals – Baffle Bolt Inspection* will be implemented by plant procedures in accordance with the *Duke Quality Assurance Program.*

Regulatory Basis – Renewal Applicant Action Item 4.1 (Items 5, 6, 9) in the Safety Evaluation Report for BAW-2248A.

REACTOR VESSEL INTERNALS – INSPECTION OF CASS RVI ITEMS

Purpose – The purpose of the *Reactor Vessel Internals – Inspection of CASS RVI Items* is to ensure that the reduction of fracture toughness properties of these items will not result in loss of the component intended functions of the reactor vessel internals during the period of extended operation.

Scope – The scope of this inspection consists of reactor vessel internals items fabricated from CASS (e.g. control rod guide tube spacers, vent valve bodies, Unit 3 outlet nozzles, and incore guide tube assembly spiders) for Oconee Units 1, 2, and 3. The vent valve retaining rings, fabricated from martensitic stainless steel, are also included in this inspection.

Aging Effects – The aging effects of concern for the reactor vessel internals items fabricated from CASS and martensitic steel are reduction of fracture toughness by thermal embrittlement and irradiation embrittlement.

Method – Reduction of fracture toughness cannot be measured through traditional in-situ examination techniques, thus necessitating an analytical approach to assess the effect of reduction of fracture toughness on the applicable reactor vessel internals items. The specific inspection method will depend on the results of these analyses. The inspection will be performed on one Oconee unit. The Oconee Unit 3 outlet nozzles will be inspected if the results of the analysis indicates such inspection is necessary.

Sample Size – The sample size of the one Oconee unit inspection will be determined concurrently with the determination of the inspection method above.

Industry Codes or Standards – No code or standard currently exists to guide or govern this inspection.

Frequency – The *Reactor Vessel Internals – Inspection of CASS RVI Items* is a one-time inspection.

Acceptance Criteria or Standard – Critical crack size will be determined by analysis. Acceptance criteria for the above aging effects will be developed prior to inspection.

Corrective Action – The need for subsequent examinations will be determined after the results of the initial examination are available. If the results of the inspection are not acceptable, then the results may be used as a baseline inspection for establishing a longer-term programmatic action to inspect the CASS RVI items in all three Oconee units. Specific corrective actions will be implemented in accordance with the *Duke Quality Assurance Program*.

Timing of New Program or Activity – Following issuance of the renewed operating licenses for Oconee Nuclear Station, this inspection will be performed on one unit (and on the Unit 3 outlet nozzles if necessary), near the end of the 4th 10-year inservice inspection interval.

Administrative Controls – The *Reactor Vessel Internals – Inspection of CASS RVI Items* will be implemented by plant procedures in accordance with the *Duke Quality Assurance Program.*

Regulatory Basis – Renewal Applicant Action Item 4.1 (Item 7) in the Safety Evaluation Report for BAW-2248A.

REACTOR VESSEL INTERNALS – INSPECTION OF NON-CASS RVI ITEMS

Purpose – The purpose of the *Reactor Vessel Internals – Inspection of Non-CASS RVI Items* is to assess the condition of the non-CASS items (e.g. plates, forgings, welds, core barrel bolts, thermal shield bolts) in order to ensure aging effects are not adversely affecting the functionality of the reactor vessel internals.

Scope – The scope of this inspection consists of the reactor vessel internals stainless steel non-CASS items for Oconee Units 1, 2 and 3.

Aging Effects – The aging effects of concern are (1) cracking due to irradiation assisted stress corrosion cracking, (2) reduction of fracture toughness due irradiation embrittlement, (3) dimensional changes due to void swelling, and (4) loss of bolted closure integrity due to stress relaxation.

Method – Current plans are to perform a visual inspection of the non-CASS RVI items on one Oconee unit. Activities are in progress to develop and qualify the inspection method.

Sample Size – The sample size of the one Oconee unit inspection will be a selected region of the most-limiting non-CASS RVI item.

Industry Codes or Standards – No code or standard currently exists to guide or govern this inspection.

Frequency – The *Reactor Vessel Internals – Inspection of Non-CASS RVI Items* is a one-time inspection.

Acceptance Criteria or Standard – Critical crack size will be determined by analysis. Acceptance criteria for the above aging effects will be developed prior to inspection.

Corrective Action – The need for subsequent examinations will be determined after the results of the initial examination are available. If the results of the inspection are not acceptable, then the results may be used as a baseline inspection for establishing a longer-term programmatic action to inspect non-CASS RVI items in all three Oconee units. Specific corrective actions will be implemented in accordance with the *Duke Quality Assurance Program.*

Timing of New Program or Activity – Following issuance of the renewed operating licenses for Oconee Nuclear Station. this inspection will be performed on one unit near the end of the 4th 10-year inservice inspection interval.

Administrative Controls – The *Reactor Vessel Internals – Inspection of Non-CASS RVI Items* will be implemented by plant procedures in accordance with the *Duke Quality Assurance Program.*

Regulatory Basis – Renewal Applicant Action Item 4.1 (Items 8, 9) in the Safety Evaluation Report for BAW-2248A.

Duke's Proposed Electrical Cabling Response

SER Open Item 3.9.3-1

This item was added after the SER for Oconee was issued. The item is discussed in a letter that was sent to Duke on October 8, 1999, and involves the aging management program for insulated cables and connections. Duke provided a proposed response to the staff on November 5, 1999, and the staff and Duke had a phone call on November 10, 1999, to discuss the issue. The phone call and Duke's proposed response are discussed in a summary dated November 18, 1999. The following is a brief synopsis of why the staff believes this issue remains open and the questions that need to be answered in order to resolve the issue.

The staff review of the November 5, 1999, draft response to OI 3.9.3-1 identified several concerns that need to be addressed by Duke in order to resolve this open item. The proposed cables and connections inspection program is too limited and needs to be expanded to include non-EQ instrumentation, control, and power cables that are within scope for license renewal and not only those that were identified in NRC Inspection Report 99-12. The proposed inspection program by Duke is limited to black instrumentation cables in the reactor building associated with a feedwater line and those next to a steam generator or the pressurizer.

In addition, heat-shrink tubing on pressurizer cable connectors is included in the inspection program. The program is limited only to a visual inspection for signs of accelerated aging such as discoloration and cracking. Cables that are inaccessible such as those that are directly buried or in conduits will not be inspected for aging with this program. In addition, the inspection program does not contain any provisions for periodic monitoring of changes to the service environments for localized hot spots (radiation or temperature) or moisture/water accumulation in conduits or trenches that may develop and result in unacceptable aging. In summary, the staff believes that the following areas need to be addressed in Duke's proposed cables and connections inspection program:

1. Expansion of the inspection program to include non-EQ cables (instrumentation, control, and power) and connections located in the Reactor Buildings, Auxiliary Buildings, Turbine Buildings, and Standby Shutdown Facility that are subject to the applicable aging effects of heat, radiation, and moisture.

2. Further investigation of the cable condition where visual observations of cable aging have shown cable surface anomalies such as discoloration, cracking or surface contamination. Acceptance criteria need to be established for visual inspections.

3. Electrical measurements on selected cables that are inaccessible or directly buried to detect aging due to radiation, temperature, or moisture.

4. Periodic monitoring of service environments of cables and connections for radiation or temperature hot spots or moisture/water accumulation in conduits or trenches.

Response to Open Item 3.9.3-1

Background

NRC Inspection Report 99-12 stated that *"for electrical cables and connections, the inspection team concluded that the potential aging effects of moisture, radiation, and heat identified in the LRA are applicable at ONS. Based on the evidence of aging effects and the team's review of actual plant experience, the team could not agree with the applicant that no aging management review is needed for electrical cables and connectors for the period of extended operation."* The areas of focus and the inspection team's findings for electrical cables and connectors that support this conclusion were discussed in the inspection report.

From discussions during the inspection, it was recognized that electrical hardware issues can be grouped into two sets. Some issues are clearly design, installation and maintenance problems that are not relevant to license renewal, while other issues are clearly aging problems that are relevant to license renewal. Engineering judgement must be used for items that fall somewhere between these clear distinctions. During the Oconee operating experience review, all electrical hardware issues that were reviewed were identified as design or maintenance problems that would not lead to a loss of function if left unmanaged. During the NRC license renewal inspections, the NRC inspectors expressed their opinion that physical evidence of degradation indicates the need for aging management. The NRC inspector observations were noted. Duke notes that the design of electrical cable at Oconee includes interlocked or braided armor that provides additional protection for the insulated conductors within the cable (See Figure 2.6-5 of Exhibit A of the Application).

Based on the inspection results and the NRC staff conclusions stated in the inspection report, Duke agrees that there are adverse localized environments where surface anomalies such as embrittlement, discoloration or cracking could occur. An adverse localized environment is defined as a condition in a limited plant area that is more severe than the specified service condition for the equipment. When viewed in an extremely conservative manner, these surface anomalies for insulated cables and connections within the scope of license renewal and installed in adverse localized environments may lead to aging effects that could cause a loss of intended function before the end of the extended period of operation. Thus, these surface anomalies can be considered to be relevant conditions requiring aging management in order to eliminate the potential for applicable aging effects that could lead to loss of component intended function.

Duke has decided to manage the aging of the insulated cables and connections that are within the scope of license renewal and that could be affected by these adverse localized environments. On November 5, 1999, Duke proposed an insulated cables and connections aging management program for staff review. In response to the staff comments on the initial proposed aging management program, a revised program description is provided below.

Aging Management Program

The aging management program for the insulated cables and connections at Oconee entitled *Insulated Cables and Connections Inspection* is described in the table below using the program attributes described in Section 4.2 of Exhibit A of the Application. Duke believes that the proposed program will provide reasonable assurance that the identified effects of aging of insulated cables and connections will be managed during the period of extended operation.

Based on NRC feedback on the version of the proposed program provided on November 5, 1999, the description of the program scope has been expanded to include insulated cables and connections within the scope of license renewal that are located in adverse localized environments in the Reactor Buildings, Auxiliary Buildings, Turbine Buildings and Standby Shutdown Facility (SSF) that could be subject to applicable aging effects from heat, radiation or moisture. The specific buildings included are those identified in staff comment 1 as contained in SER Open Item 3.9.3-1. To clarify the meaning of the program scope, the definitions of adverse localized environment and applicable aging effects were also added. The aging management program will be performed by visual inspections that will be performed at least every 10 years. As the aging effects are relatively slow acting, even when accelerated by higher temperatures, an inspection every 10 years is adequate. Staff comment 2 (discussed in SER Open Item 3.9.3-1 above) has been incorporated into the program under method and acceptance criteria. Staff comments 3 and 4 were not incorporated as explained below.

Staff comment 3 is "*Electrical measurements on selected cables that are inaccessible or directly buried to detect aging due to radiation, temperature, or moisture.*" Electrical measurements were not included in the *Insulated Cables and Connections Inspection* because Duke is not aware of any electrical methods currently available to detect aging degradation prior to loss of function. Meggering, which was mentioned several times during the on-site inspections as a possible aging management method, is not a method that can provide trending information for an assessment of cable insulation aging degradation. Meggering is a useful tool for testing standby equipment that must remain deenergized. Therefore, Duke is not aware of any reliable and proven electrical measurement methods that can detect and trend aging, this method was not included in the program.

Staff comment 4 is "*Periodic monitoring of service environments of cables and connections for radiation or temperature hot spots or moisture/water accumulation in conduits or trenches.*" Service environments of cable and connections need not be periodically monitored for radiation or thermal hot spots. Hot spots by their very nature exist because of (a) inadequate initial design, (b) invalid design assumptions, (c) a change in the operating modes of the plant, or (d) modifications to plant systems or equipment. When a hot spot is identified, the situation is analyzed then corrected or modified to eliminate the problem. The most comprehensive industry document on the identification and treatment of adverse localized equipment environments such as hot spots is the 1999 EPRI report TR-109619. Duke personnel participated in the development of this document and Duke will use it as guidance in the future. The document emphasizes visual inspection walkdowns as the primary method for identifying adverse localized environments and states: "*The basic walkdown guidance presented in Appendix A can*

successfully identify adverse localized environments and their effects ... as these adverse environments are identified, they are managed by such activities as periodic replacement, relocation of the cable, addition of thermal insulation, or improvements to HVAC." Therefore, Duke will use visual inspections, as described below, instead of periodic monitoring of service environments.

Insulated Cables and Connections Inspection
Inspection Attributes

Purpose – The purpose of the *Insulated Cables and Connections Inspection* is to inspect insulated cables and connections within the scope of license renewal to gain reasonable assurance that their intended functions will be maintained through the extended period of operation.

Scope – The *Insulated Cables and Connections Inspection* will include accessible insulated cables and connections within the scope of license renewal that are installed in adverse localized environments in the Reactor Buildings, Auxiliary Buildings, Turbine Buildings and Standby Shutdown Facility (SSF) that could be subject to applicable aging effects from heat, radiation or moisture. An adverse localized environment is defined as a condition in a limited plant area that is significantly more severe than the specified service condition for the equipment. An applicable aging effect is an aging effect that could cause the insulated cables or connection to lose its intended function before the end of the extended period of operation.

Aging Effects – Change in material properties of the cable and connector insulating materials is the applicable aging effect. Surface anomalies such as embrittlement, discoloration or cracking are relevant conditions that can be monitored to preclude the applicable effect.

Method – Insulated cables and connections will be visually inspected for surface anomalies such as embrittlement, discoloration or cracking.

Sample Size – Sample size is not required for this program.

Industry Codes and Standards – EPRI TR-109619, *Guideline for the Management of Adverse Localized Equipment Environments*

Frequency – An inspection of all adverse localized environments containing in-scope insulated cables and connections will be performed at least once every 10 years.

Acceptance Criteria or Standard – No unacceptable visual indications of the relevant conditions for the insulated cables and connections that could lead to applicable aging effects as determined by engineering evaluation.

Corrective Action – Further investigation by engineering will be performed when surface anomalies such as embrittlement, discoloration or cracking are found in order to ensure that insulated cable or connection intended functions will be maintained. Corrective actions may include, but are not limited to, testing, shielding, relocating and replacement of the affected cable or connection. Specific corrective actions will be implemented in accordance with the Problem Investigation Process. The Problem Investigation Process applies to all structures and components within the scope of the *Insulated Cables and Connections Inspection.*

Timing of New Program or Activity – The *Insulated Cables and Connections Inspection* will be initiated during the second refueling outage for each unit following the issuance of renewed operating licenses for Oconee Nuclear Station.

Administrative Controls – The *Insulated Cables and Connections Inspection* will be performed as part of the System Engineering Walkdowns by the engineer responsible for insulated cables. The responsible engineer may adjust the attributes of this inspection provided such changes do not adversely affect the capability of the inspection to manage the effects of aging.

Regulatory Basis – The *Insulated Cables and Connections Inspection* has no current regulatory basis.

Duke's Proposed Response for HVAC Issue

SER Open Item 3.6.1.3.1-1 – In RAI 3.5.8-3, the staff questioned the applicant's identification of applicable aging effects for the HVAC system. The staff raised a concern that, on the basis of its experience, cracking of ductwork occurs from vibration-induced fatigue and loosening fasteners from dynamic loading, especially in the vicinity of attached device types exposed to dynamic loads such as fans. The applicant responded to RAI 3.5.8-3 by letter dated January 25, 1999, stating that cracking of ductwork from vibrational loads and self-loosening of fasteners from dynamic loading were determined not to be applicable aging effects for the HVAC system. The applicant stated that components within the scope of license renewal are equipped with isolators to prevent transmission of vibration and dynamic loading to the rest of the system. Therefore, vibration-induced fatigue and self-loosening of fasteners are not applicable aging effects for the HVAC system. The staff's review of operating experience is that vibration-induced fatigue and self-loosening of fasteners cannot be avoided by installing isolators. The staff, in a subsequent letter dated April 8, 1999, regarding RAI 3.5.8-3 requested that the applicant address these aging effects or present additional justification for not considering them applicable aging effects. The applicant responded in a letter dated May 10, 1999, that the ONS has had good operating experience with respect to isolators in the auxiliary building ventilation system and control room pressurization and filtration system in preventing the transmission of vibration and dynamic loads to surrounding equipment to preclude cracked ductwork and loosened fasteners. A review of the ONS Problem Investigation Process (PIP) database and ONS-specific licensee event reports did not identify any instances of cracking of ductwork or loosening of fasteners in these two ventilation systems. In addition, these two systems have been in service for more than 25 years and cracking of ductwork and loosening of fasteners would have revealed itself as a concern by now. Therefore, the applicant concluded that cracking of ductwork and loosening of fasteners in the auxiliary building ventilation system and control room pressurization and filtration system are not applicable aging effects for these systems. The staff finds the additional justification presented by the applicant not acceptable for the following two reasons:

In general, sub-component parts of isolators are made of elastomers (such as rubber boots, seals, and flexible collars) and elastomers will degrade from relative motion between vibrating equipment, pressure variations, exposure to temperature changes and oxygen. Because of the degradation of isolators, vibration and subsequent dynamic loads applied to the ductwork and fasteners cannot be eliminated. Although no aging effects (cracking of ductwork and loosening of fasteners) were identified after 25 years of operation, one still cannot ensure that there will not be any degradation of the systems within the next 35 years (the remaining design life plus the extended life). The staff believes that these aging effects are applicable because of the nature of the materials involved.

Duke Response to SER Open Item 3.6.1.3.1-1

Duke understands the staff concern for the aging of the ductwork and fasteners. However, for these items to be impacted by dynamic loading, aging must first affect the elastomers designed to preclude the effects of such relative motion. Rubber boots, seals, and flexible collars in question are elastomers that are a part of the air handling units (source of dynamic loads) located in the Auxiliary Building for the Auxiliary Building Ventilation and Control Room Pressurization and

Filtration Systems (Oconee License Renewal Application Exhibit A, Section 3.5.8). The Penetration Room Ventilation System also discussed in Section 3.5.8 of the Application is constructed of pipe and does not contain these items. These elastomers are used to connect air handling units to adjacent ductwork and have a geometry and material composition that allows relative motion between the air handling unit and adjacent ductwork and prevents the transmission of dynamic loads from the air handling unit to the adjacent ductwork. These elastomers are located in the controlled environment of the Auxiliary Building, but could degrade by hardening over time.

Hardening of the elastomers may lead to other aging effects such as cracking of the ductwork and loosening of fasteners. For license renewal, management of the elastomers such that aging of the ductwork is precluded can be performed for a system if failure of the elastomer and, in turn, the ductwork would lead to loss of component intended function. The Auxiliary Building Ventilation System performs the system intended function of smoke removal during certain fire scenarios. Following a fire in either the control rooms, equipment rooms, or cable rooms, smoke is removed using either installed purge fans or portable purge equipment. The smoke is exhausted into areas in the Auxiliary Building for removal by the Auxiliary Building Ventilation System. Hardening of the elastomers that could lead to cracking of the elastomer and perhaps cracking of ductwork and loosening of fasteners during relative motion would not fail the intended function of smoke removal. Smoke removal would be accomplished with cracks in the elastomer or in the ductwork. Since loss of the elastomer flexibility function or ductwork component intended function would not cause loss of the system intended function, aging management of the elastomers in the Auxiliary Building Ventilation System is not required.

The Control Room Pressurization and Filtration System has a system intended function of providing a suitable environment in the control room following postulated design basis events. In addition, the system must maintain a positive pressure in the control room for accident conditions to prevent unfiltered air from entering the control room. For those items located in the Auxiliary Building, hardening of the elastomers that could lead to cracking of the elastomers and perhaps cracking of ductwork and loosening of fasteners during relative motion could fail the intended functions of this system. For license renewal, managing the aging of the Control Room Pressurization and Filtration System elastomers will preclude the possibility of the aging of the ductwork and loosening of fasteners due to relative motion. The aging management of the elastomers in the Control Room Ventilation System Examination is presented in the response to SER Open Item 2.2.3.4.3.2.1-2.

SER open item 3.6.1.3.1-1 (as described in the November 18, 1999 NRC letter)

This issue involves the aging effects of heating ventilation and air conditioning sub-component parts of vibration isolators. In a phone call with Duke on October 27, 1999, the staff questioned Duke's portion of the response regarding the auxiliary building ventilation system. The staff stated that Duke had the following 4 options to resolve the issue for the auxiliary building ventilation system: revisit whether or not the system is within the scope of license renewal, provide an aging management program for the elastomers in the system, provide a more rigorous analysis for why failure of the elastomers would not fail the intended function of smoke removal, or consider the elastomers a consumable.

Duke has not responded to the options presented during the October 27, 1999, phone call, therefore, the staff considers this to be an open item.

Duke Response to the issue as described in the November 18, 1999 letter

In the November 18, 1999 letter, the NRC staff offered four options to address the issue involving aging effects on sub-component parts of vibration isolators. Each option is addressed below along with a fifth option proposed by Duke:

1. Revisit whether or not the system is within the scope of license renewal

Duke has reviewed the current licensing basis associated with the fire protection regulated event identified in 10 CFR §54.4(a)(3) and found that the Auxiliary Building Ventilation System has a system intended function in support of license renewal. Auxiliary Building Ventilation System intended function is to vent smoke from the Auxiliary Building that results from a control room fire. Following the control room fire, the smoke that originates in the control room would be subsequently vented into the Auxiliary Building causing the need for smoke removal. Therefore the portions of the system needed to perform this smoke removal function are within the scope of license renewal and subject to aging management review.

2. Provide an aging management program for the elastomers in the system

This option is somewhat premature in that it presumes applicable aging effects have been identified for the elastomers in the system. This subject is covered in more detail in Duke option 5 below.

3. Provide a more rigorous analysis for why failure of the elastomers would not fail the intended function of smoke removal

The failure of the elastomers is the center point of this option. This failure will be defined as complete severance due to tearing of the neoprene-impregnated woven fiberglass flexible collars on all the Auxiliary Building Ventilation System exhaust fans. Complete severance due to tearing is required since no limits exist for the amount of time required to remove smoke from the Auxiliary Building. Without such a limit, any amount of flow through the ductwork to the unit vents is sufficient to accomplish the system intended function. Any aging, such as cracking, that does not completely sever the fan from the ductwork is not an applicable aging effect because it would not cause loss of system intended function. Further, failure of the neoprene-impregnated woven fiberglass flexible collars on all ventilation exhaust fans must occur since any fan still intact will accomplish the system intended function. Since Duke believes that aging

to the extent required to cause complete severance of the flexible collar is not occurring, the need for a more rigorous analysis is not required. Further discussion of Duke's understanding is provided in option 5 below.

4. Consider the elastomers a consumable

Duke does not consider the neoprene-impregnated woven fiberglass flexible collars to be consumables.

5. Duke Option – Evaluate whether the elastomers have an applicable aging effect

The flexible collars installed on the outlets of fans are neoprene-impregnated woven fiberglass. The woven fiberglass provides structure for the shape of the conduit and connectivity between the fan and duct. The woven fiberglass also adds strength to the flexible collar. Neoprene impregnation seals the woven fiberglass to provide the pressure boundary.

As noted in Option 3 presented earlier, the aging effect of concern for the composite material is tearing (versus cracking) to result in a loss of the intended function of the Auxiliary Building Ventilation System. Before tearing would occur, the neoprene would need to be aged to the extent tearing could occur. Neoprene is evaluated below for the specific conditions in the Auxiliary Building Ventilation System to identify any aging effects that will result in a loss of the intended function during the period of extended operation if left unmanaged.

Neoprene can potentially degrade due to thermal exposure, ozone, ultraviolet light, ionizing radiation, and relative motion (caused by vibration and changes in pressure). Each of these stressors is addressed below.

Prolonged exposure to excessive temperatures will embrittle neoprene. Table 4-2 of the Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations [Reference 3.6-1 in Exhibit A of the Application] recommends maximum ambient temperatures for a sixty-year life for various insulating materials used in instrument and control and power cables. One of these insulating materials is neoprene. The neoprene in the instrument and control cables and the flexible collars would be exposed to very similar environments. Both are exposed to the Auxiliary Building environment externally and are exposed to minimal internal heating. Table 4-2 notes that for retention of 50% absolute elongation for sixty years, ambient temperatures must remain below 117°F. This means that the neoprene can remain in an environment for 60 years at 117°F and still be stretched 50% beyond its original dimensions without losing its integrity (i.e., it doesn't split) or its ability to perform its function.

According to UFSAR Section 9.4.3.1 [Reference 3.6-2 of the Application], the Auxiliary Building Ventilation System maintains the Auxiliary Building temperature between 60°F and 104°F to allow for periodic personnel entry for inspection and maintenance. This temperature range allows for a greater than 60 year service life of the neoprene-impregnated woven fiberglass flexible collars before reaching the retention of 50% absolute elongation. Therefore, aging due to thermal exposure of the neoprene-impregnated woven fiberglass flexible collars is not an applicable aging effect.

Table 4-10 of the Aging Management Guideline for Commercial Nuclear Power Plants - Cable and Termination Components [Reference 3.6-1 of the Application] notes that only styrene-butadiene rubber and Buna-N rubber exposed to ozone and subject to ionizing radiation could

experience surface hardening and embrittlement. Neoprene is not listed as susceptible to degradation due to exposure to ozone. Therefore, aging due to ozone exposure of neoprene-impregnated woven fiberglass flexible collars is not an applicable aging effect.

Neoprene could also degrade when exposed to ultraviolet light for extended periods of time. The source of ultraviolet light is solar radiation. The neoprene-impregnated woven fiberglass flexible collars are located in an interior room inside the Auxiliary Building that precludes exposure to solar radiation. Therefore, degradation of the flexible collars due to extended exposure to ultraviolet light is not an applicable aging effect.

From Figure C-3 of EPRI NP-1558, "A Review of Equipment Aging Theory and Technology," dated September 1980 [Reference 3.6-3 of the Application], neoprene with a cumulative exposure to ionizing radiation up to 2 X 10^6 rads will experience incipient to mild damage but will retain its function. Table 3.6-3 of Exhibit A of the Application provides a bounding cumulative exposure of 1.5 X 10^6 rads for components in the Auxiliary Building. The neoprene-impregnated flexible collars are located in rooms in the Auxiliary Building on the 809, 822, and 838 feet elevation that allows personnel periodic access for routine monitoring and maintenance. In these areas where these flexible collars are located, the radiation exposure is expected to be much less than the bounding value of 1.5 X 10^6 rads which is less than the value of 2 X 10^6 rads that may cause damage to neoprene. Therefore, aging due to ionizing radiation is not an applicable aging effect for the neoprene-impregnated woven fiberglass flexible collars.

Neoprene can fail due to relative motion caused by changes in pressure and vibration. For this aging to occur, neoprene must embrittle such that the pressure changes and vibration result in cracking of the neoprene. As discussed earlier, neoprene can embrittle under certain environmental conditions that reduce the ability of the neoprene to accommodate the pressure changes and vibration. None of these environmental conditions exist that would embrittle the neoprene-impregnated fiberglass flexible collars for the period of extended operation. In addition, the woven fiberglass provides additional strength and resistance to tearing. Therefore, aging due to relative motion is not an applicable aging effect for these flexible collars.

To confirm that that there were no applicable aging effects for the neoprene-impregnated woven fiberglass flexible collars, Duke performed a review of Duke specific and industry operating experience to identify applicable aging effects for these flexible collars. In addition, Duke also performed a review of Duke specific and industry operating experience to identify instances of duct cracking and loosening of fasteners due to hardening of elastomer flexible collars. The staff had previously noted that the industry has experienced cracking of duct and loosening of fasteners in ventilation equipment adjacent to rotating equipment due to hardening of elastomer flexible collars. These reviews identified several instances of duct cracking that were attributed to improper initial design and construction or operating a system outside its original design due to a change in system operation or modification. No instances were identified of loosening of fasteners or flexible collar failures.

The NRC noted in the Safety Evaluation Report Open Item that although no aging effects (cracking of ductwork and loosening of fasteners) were identified after 25 years of operation, one still cannot ensure that there will not be any degradation of the systems within the next 35 years (the remaining design life plus the extended life). The technical evaluation based on industry

literature, industry operating experience, and Oconee specific experience show that aging of neoprene-impregnated woven fiberglass flexible collars is not a concern for license renewal.

Based on the industry literature that shows that neoprene has no aging effects when exposed to the Auxiliary Building Ventilation System internally and the Auxiliary Building environment externally, neoprene-impregnated woven fiberglass flexible collars will also not experience any aging effects that could cause loss of intended function for the period of extended operation. This conclusion is supported by the Duke specific and industry operating experience which did not identify any instances of ventilation system failures attributed to aging of flexible collars. Therefore, neoprene-impregnated woven fiberglass flexible collars have no aging effects for license renewal.

Summary of Issue

The Auxiliary Building Ventilation exhaust fans take suction from the room where they are installed. The room serves as a plenum that collects air from the intake ductwork. The exhaust fans discharge the air to the unit vent for monitoring and release. The discharge duct is connected to the fans with elastomeric flexible collars that are a part of the fan unit. These flexible collars prevent the transmission of vibration generated by fan operation to the adjacent duct where it could lead to duct cracking and loosening of fasteners as well as provide a pressure boundary function in support of the system intended function. The collars are made out of neoprene-impregnated woven fiberglass along with metal parts to connect them to the fan and the ductwork. The material of interest here is the neoprene-impregnated woven fiberglass.

One concern expressed by the NRC staff is that hardening of the neoprene-impregnated woven fiberglass could result in the transmission of vibration to adjacent ductwork that could lead to cracking of the ductwork and loosening of fasteners such that the system intended function is lost. Another concern is that cracking of the neoprene-impregnated woven fiberglass could result in the loss of the component intended function such that the system intended function would be lost.

As discussed in Option 5 above, Duke has investigated the aging of neoprene materials and has established no technical justification to consider either hardening or cracking to be an applicable aging effect. Further, Duke has not identified any operating experience to support this concern. The neoprene-impregnated woven fiberglass portion of the flexible collars has a corrugated (wavy) design. This geometry allows relative motion between the fan unit and the adjacent ductwork and prevents transmission of vibration from the fan unit to the ductwork. In the Duke response to SER Open Item 3.6.1.3.1-1, Duke offered an assumption that hardening of this neoprene-impregnated woven fiberglass portion of the flexible collars may occur that could lead to ductwork aging issues, including cracking of the flexible collar. Additional investigation into the applicability of this aging has demonstrated that this assumption is unfounded. As described in Option 5 above, the neoprene-impregnated woven fiberglass flexible collars on the Auxiliary Building Ventilation exhaust fans have no applicable aging effects. Based on this investigation and substantiated by over 25 years of Oconee operating experience as well as industry experience, Duke has reasonable assurance that any aging of the neoprene-impregnated woven fiberglass flexible collars will not result in a loss of the system intended function in the period of extended operation. Therefore, no aging management program is required.