OCONEE NUCLEAR STATION

LICENSE RENEWAL - TECHNICAL INFORMATION

TOPICAL REPORT

OLRP-1001

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1. INTRODUCTION

1.1 PURPOSE

The License Renewal Rule [1] (hereafter referred to as Part 54) requires that an application for a renewal license contain certain technical information. The purpose of this topical report, "Oconee Nuclear Station, License Renewal - Technical Information, OLRP-1001" (hereafter referred to as OLRP-1001), is to provide the technical information required by §54.21(a) - "An integrated plant assessment"; §54.21(b) - "CLB changes during NRC review of the application"; and §54.21(c) - "An evaluation of time-limited aging analyses."

Individual sections of OLRP-1001 will be submitted for NRC review and approval by the dates noted in each section. A final safety evaluation covering the entire contents of OLRP-1001 was requested in the transmittal letter of the initial submittal dated July 31, 1996. Following issuance of a final safety evaluation report approving OLRP-1001 by the NRC, Duke Power will use the information contained therein, as well as other considerations, in deciding whether to continue to pursue license renewal for Oconee Nuclear Station.

In the event Duke Power decides to submit an application for a renewal license for Oconee Nuclear Station, the approved version of OLRP-1001 would be incorporated by reference into the application for a renewal license as permitted by 10 CFR §54.17(e). Referencing OLRP -1001 in an application for a renewal license, addressing any applicant required action items identified in the NRC safety evaluation report, and summarizing the credited aging management programs and time-limited aging analyses in the final safety analysis report supplement should provide the NRC with sufficient information to make the finding required by 10 CFR §54.29(a).

1.2 ORGANIZATION

OLRP-1001 is organized in the following manner:

Chapter 1 contains introductory information and identifies the Current Licensing Basis (CLB) changes which occur during the NRC review of OLRP-1001 and the application for a renewal license, pursuant to the requirements of §54.21(b). It also identifies time-limited aging analyses and the results of the plant specific exemptions review, conducted pursuant to §54.21(c).

Chapter 2 describes the structure and component selection process as well as the results of the Integrated Plant Assessment (IPA), pursuant to §54.21(a)(1) and (2).

Chapter 3 discusses the aging effects review activities of the IPA, pursuant to §54.21(a)(3), as well as the evaluations of time-limited aging analyses that have been identified pursuant to §54.21(c).

Chapter 4 includes the overall conclusion determined from the Chapter 3 reviews.

Chapter 5 includes the references which are identified throughout OLRP -1001 by square brackets, '[]'.

1.3 CLB Changes During NRC Review of Application

This section documents CLB changes that materially affect the contents of OLRP-1001 and the application for a renewal license, including the Final Safety Analysis Report (FSAR) supplement, as required by §54.21(b).

1.4 TIME-LIMITED AGING ANALYSIS REVIEW

1.4.1 Identification of Time-Limited Aging Analyses

Part 54 requires a list of time-limited aging analyses (TLAA) be provided as part of the application for a renewal license. TLAA are defined in §54.3 as those licensee calculations and analyses that meet six specific criteria. The process used to identify the Oconee specific TLAA is consistent with the guidance provided in NEI 95-10, Section 5 [2].

Oconee-specific source documents that were reviewed for TLAA include the Oconee licensing correspondence file, the Oconee FSAR [3], BWNT Topical Reports referenced in correspondence and the FSAR, and ASME Section XI Summary Reports.

Generic source documents which were reviewed include the Standard Review Plan, various codes and standards, and certain NRC generic regulatory compliance documents including Bulletins, Generic Letters, Regulatory Guides, and 10 CFR Part 50 and its Appendices. This review was conducted as part of an industry effort in which Duke participated. The review of generic source documents confirmed the results from the review of Oconee-specific source documents.

The information developed from the review of both Oconee-specific source documents and generic source documents was reviewed to determine which calculations and analyses met all six criteria of §54.3. Those analyses and calculations that met all six criteria were identified as Oconee-specific TLAA. The list of Oconee-specific TLAA is contained in Table 1.4-1.

As required by §54.21(c)(1), an evaluation of each Oconee-specific TLAA must be performed. The evaluation must demonstrate that:

1) the analyses remain valid for the period of extended operation; or
2) the analyses have been projected to the end of the period of extended operation; or
3) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The evaluation conclusions are presented in Table 1.4-1. The supporting information for each of these conclusions is presented in the sections as noted.

1.4.2 Exemptions

Part 54 requires that the application for a renewal license include a list of plant-specific exemptions that were granted pursuant to §50.12, are currently in effect, and are based on time-limited aging analyses as defined in §54.3. A review of the Oconee docket has been performed and the results of this review identified that no §50.12 exemptions were granted on the basis of a time-limited aging analysis as defined in §54.3.

Table 1.4-1 Time-Limited Aging Analyses

Structure, Component / TLAA	Evaluation Conclusion	Supporting Information
1. **Reactor Building Liner Plate and Penetrations /** Thermal Fatigue	Existing analyses remain valid for the period of extended operation.	See Section 3.3.2.2
2. **Reactor Building Tendons /** Loss of Prestress	Evaluation in progress. Results to be provided at a later date.	See Section 3.3.3.2
3. **Reactor Vessel /** Embrittlement, Upper Shelf Energy Toughness, Thermal Shock, PTS, under clad crack growth.	Evaluation in progress. Results to be provided at a later date.	
4. **Reactor Vessel Internals** / Loss of fracture toughness, fatigue	Evaluation in progress. Results to be provided at a later date.	
5. **Reactor Coolant System /** Fatigue	Evaluation in progress. Results to be provided at a later date.	
6. **Class 1 Components /** Fracture Mechanics Analyses for ISI reportable indications (fatigue)	Evaluation in progress. Results to be provided at a later date.	
7. **Once Through Steam Generator /** Vibration, lifetime	Evaluation in progress. Results to be provided at a later date.	
8. **Reactor Coolant Pump Flywheel /** Fatigue	Evaluation in progress. Results to be provided at a later date.	
9. **Non-Class 1 Piping /** Thermal Fatigue	Evaluation in progress. Results to be provided at a later date.	
10. **Electrical Equipment /** Environmental Qualification	Evaluation in progress. Results to be provided at a later date.	
11. **Polar Crane /** Fatigue, heavy load cycles	Evaluation in progress. Results to be provided at a later date.	
12. **Spent Fuel Rack Boraflex** / Aging of non-metallic material	Evaluation in progress. Results to be provided at a later date.	

2. INTEGRATED PLANT ASSESSMENT - STRUCTURE/COMPONENT IDENTIFICATION

2.1 INTRODUCTION

Part 54 requires that for those systems, structures, and components within the scope of this part, as delineated in §54.4, those structures and components subject to an aging management review be identified and listed. §54.21(a)(1). Part 54 further requires that the methods used to identify and list these structures and components be described and justified. §54.21(a)(2). The methods use by Duke to accomplish these requirements of Part 54 follow the guidance that has been provided in NEI 95-10, Section 4.1 [2].

The structures and components that are within the scope of license renewal and subject to an aging management review have been identified. The following sections describe and justify the methods used to identify the structures and components subject to an aging management review. The sections also provide the lists these structures and components. The process described provides reasonable assurance that for all systems, structures and components within the scope of license renewal, as delineated in §54.4, those structures and components that are subject to an aging management review have been identified.

2.2 STRUCTURE / COMPONENT IDENTIFICATION PROCESS OVERVIEW

Part 54 states that all structures and components that are long-lived and passive are subject to aging management reviews. §§54.21(a)(i), (ii). The methods that were used to identify structures and components that are subject to aging management reviews are summarized in the following paragraphs. Additional descriptive information is provided in the sections that follow. Guidance contained in NEI 95-10, Sections 3 and 4, has been used as an aid in making the above determinations.

Oconee source documents describe the design basis events and responses to the five regulated events specified in Part 54. Oconee source documents include, but are not limited to, the Oconee FSAR [3], Oconee engineering documents, and docketed correspondence between NRC and Duke Power. These documents were used to identify the systems, structures, and components that are within the scope of Part 54 and their intended functions.

A review of the Oconee source documents was performed to identify those systems, structures, and components that meet the requirements of Criteria (1) and (2) of §54.4. The Oconee safety-related systems, structures, and components and non-safety-related systems, structures, and components whose failure could prevent a safety function from being fulfilled are listed in Oconee engineering documents.

Oconee source documents were also reviewed to identify systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the requirements of five regulations, pursuant to Criterion (3) in §54.4. Submittals made by Duke, the NRC evaluation of these submittals, as well as Oconee engineering documents have been reviewed to identify the systems, structures, and components relied on to perform a function that demonstrates compliance with the following regulations: Fire Protection (10 CFR §50.48), Environmental Qualification (10 CFR §50.49), Pressurized Thermal Shock (10 CFR §50.61), Anticipated Transients Without Scram (10 CFR §50.62) and Station Blackout (10 CFR §50.63). Systems, structures, and components required to comply with these regulations have been identified and are listed in Oconee engineering documents.

The Reactor Building was determined to be within the scope of license renewal. Its components that are within the scope of license renewal and are long-lived and passive are subject to an aging management reviews. The method used to identify the Reactor Building components that are subject to aging management review is described and the list of reviewable components is provided in Section 2.3

Likewise, the Reactor Coolant System was determined to be within the scope of license renewal. Its components which are within the scope of license renewal and which are long-lived and passive are also subject to an aging management reviews. The method used to identify the Reactor Coolant System components that are subject to aging management review is described and the list of reviewable components is provided in Section 2.4.

Based on the review of Oconee source documents, a number of mechanical systems were identified as being within the scope of license renewal. A review of the Oconee drawings for each mechanical system in scope was performed to identify those components that are long-lived and passive and subject to an aging management reviews. The methods used to identify these components and the list of mechanical components that have been determined to be subject to aging management review are described in Section 2.5.

In addition, a number of electrical, instrumentation and control systems have also been identified as being within the scope of license renewal. Duke has elected to use the approach described in the Sandia Cable Aging Management Guideline (AMG) [4] as the method to perform the IPA on electrical components. Section 2.6 describes the method used to identify the electrical component groups and lists the groups that have been determined to be subject to aging management reviews.

Finally, there are many structures and structural components that were identified as being within the scope of license renewal. A review of Oconee source documents was performed and it was determined that structures and structural components generally are long-lived and passive and that they could be placed into several structure and structural component groups. The method used to identify these groups and the list of structures and structural components subject to aging management reviews are described in Section 2.7

2.3 REACTOR BUILDING

The Oconee Reactor Buildings are identified as Class 1 structures in the Oconee FSAR, Section 3.2. Class 1 structures are those which prevent uncontrolled release of radioactivity and are designed to withstand all loadings, including seismic, without loss of function. All Class 1 structures fulfill §54.4(a)(1). In addition, it has been determined that the Reactor Buildings perform functions that demonstrate compliance with the Commission's fire protection regulation (10 CFR §50.48) and the station blackout regulation (10 CFR §50.63).

The Reactor Building and its components support one or more of the following intended functions:

- Provides essentially leaktight barrier to prevent uncontrolled release of radioactivity.
- Provides structural and/or functional support to safety related systems, structures, and components.
- Provides shelter/protection to safety related systems, structures, and components (including radiation shielding).
- Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
- Serves as an external missile barrier.
- Provides structural and/or functional support to non-safety related systems, structures, and components where failure of this structural component could directly prevent satisfactory accomplishment of any of the required safety related functions.
- Provides heat sink during station blackout.

Each Oconee Reactor Building is a post-tensioned, reinforced concrete structure with a shallow dome roof, a cylinder wall, a floor, and a flat foundation slab. The interior of the concrete structure is lined with a steel liner plate. The guidance provided in NEI 95-10, Section 4.1.2, indicates that structures within the scope of license renewal are long-lived and passive. The Reactor Building long-lived and passive components which are subject to aging management reviews are listed in Table 2.3-1, as required by §54.21(a)(1).

The Reactor Coolant System equipment supports attached to the Reactor Building are addressed in Sections 2.4 and 3.4, Reactor Coolant System. The integrated plant assessments of process piping passing through the mechanical penetrations are addressed in Sections 2.5 and 3.5, Mechanical Components. Reactor Building internal structures, including the polar crane, are addressed in Sections 2.7 and 3.7, Structures and Structural Components.

The Reactor Buildings are founded on bedrock and include a number of components that are exposed to several environments which are applicable to the aging management review within this report:

- The Reactor Building concrete foundation slab and the portion of the external cylinder wall below grade are exposed to backfill and groundwater. The groundwater chemistry plays a major role in the degradation of the below grade components.
- The Reactor Building external concrete dome, cylinder wall above grade, and some of the post-tensioning components are exposed to the external environment.
- The post-tensioning system components and the cylinder wall above grade enclosed by adjacent buildings are exposed to controlled environments which protects them from external weather and temperature changes.
- The top of the concrete floor, the liner plate, and other select steel components are exposed to the internal environment of the Reactor Building. High temperature, humidity and radiation play a role in the degradation of the components located within this environment.

The following sub-sections provide descriptions of the three Reactor Building component groups that are within the scope of license renewal. Additional descriptions of the Reactor Building are contained in the Oconee FSAR, Section 3.8 [3].

2.3.1 Concrete Components

The Reactor Building concrete components that are within the scope of license renewal and which are subject to an aging management review are listed in Table 2.3-1. The codes and standards, including applicable editions, which were used for the original design and fabrication are given in the Oconee FSAR, Section 3.8.

The Reactor Building reinforced concrete components were designed in accordance with the American Concrete Institute (ACI)318-63 and constructed in accordance with ACI 301. The concrete ingredients consist of Type II cement American Society of Testing and Materials(ASTM) C-150, Solar 25 air entraining agent (ASTM C-260), Plastiment water-reducing agent (ASTM C-494), aggregate (ASTM C-33), and reinforcing steel (ASTM A615 Grade 40 and Grade 60). In addition, resistance to surface deterioration is enhanced by the application of prestress to the concrete sections. The prestressed concrete design places the cylinder and dome concrete in compression for all normal loading conditions over the current and extended period of operation. The precompression minimizes the number and width of shrinkage, temperature, or load induced cracks.

The reinforced concrete dome and cylinder walls are prestressed by a post-tensioning system. The combined strength provided by the concrete, conventional reinforcing steel, and the post-tensioning system is utilized to account for the design loads. Although these three material components act together as one composite system, the post-tensioning system is addressed as a separate component because it is installed and stressed after the reinforced concrete components are complete and because of the tendon surveillance program.

Conventional reinforcing is provided near the surface of the cylinder walls and dome primarily to resist local moment and shear loads at discontinuities and for temperature and shrinkage crack control. The conventional reinforcing is accounted for in the strength design of the concrete sections for the internal shear forces and moments resulting from the design loadings.

The reinforced concrete floor is provided in the Reactor Building interior above the bottom portion of the liner plate to protect the liner plate from punctures and corrosion. The conventionally reinforced concrete foundation slab serves as the structural foundation support for the Reactor Building. The vertical tendons extend through the foundation slab thickness and are anchored on the underside of the foundation slab.

2.3.2 Steel Components

The Reactor Building steel components that are within the scope of license renewal and which are subject to an aging management review are listed in Table 2.3-1. The codes and standards, including applicable edition, used for original design and fabrication are given in the Oconee FSAR, Section 3.8. The Reactor Building steel component design complies with the codes and standards of the American Society of Mechanical Engineers (ASME Section III - 1965) for the pressure boundary, the American Institute of Steel Construction (AISC, 6th Edition) for the structural steel, and the American Welding Society (AWS).

The interior of the Reactor Building is lined with steel liner plates that are welded together. The liner plate covers the dome, the cylinder wall and also runs between the floor and the foundation slab to form an essentially leaktight boundary. Steel anchors and embedments are welded to the liner and serve to anchor the liner plate to the Reactor Building concrete. In addition, other anchors and embedments are provided to transfer loads from attachments to the liner plate to the concrete wall or the concrete basemat. The liner plate is coated on the interior surface with inorganic zinc primer and Phenoline 305 for corrosion protection. There is no coating on the side of the liner plate in contact with the concrete.

Anchors/embedments are steel commodities, such as angles and anchor studs, that are welded to the liner and serve to anchor the liner to the Reactor Building concrete shell. The anchors/embedments serve to maintain the essentially leaktight barrier by preserving the integrity of the liner.

Attachments to the liner that are integral with the liner and concrete structure (i.e., attachment has corresponding anchor in concrete), include those equipment or system supports that are connected to the inside face of the liner and are thus exposed to the interior of the Reactor Building. The polar crane brackets are examples of attachments to the liner. The polar crane brackets consist of welded carbon steel plate construction of the same material, fabrication, and inspection practices as used for the liner.

Two personnel hatches are provided in each Reactor Building for personnel access and egress. One is used for normal passage, the other is used in emergency situations. The hatches are constructed of carbon steel which has been painted. Each hatch consists of a double door, welded steel assembly. Each hatch door has a flexible seal.

The equipment hatch is provided for moving large equipment into and out of the Reactor Building. The hatch is fabricated from carbon steel and is painted. The hatch is provided with a double sealed flange which is bolted to the Reactor Building during normal operation.

The penetrations of the Reactor Building include mechanical penetrations, electrical penetrations, spare penetrations, and the two fuel transfer tubes. Mechanical penetrations provide the means for passage of process piping across the Reactor Building boundary. The integrated plant assessment of process piping is addressed in Sections 2.5 and 3.5, Mechanical Components. Spare penetrations consist of a steel sleeve with welded end cap closure(s) or bolted blind flange plate(s) with gaskets at both ends. These spare penetrations are addressed as part of the Reactor Building steel components.

Electrical penetrations provide the means for electrical conductors to cross the Reactor Building boundary while maintaining the essentially leaktight barrier. The scope of review contained in this section includes all metallic components of the electrical penetration that are part of the Reactor Building pressure boundary. The integrated plant assessment of the enviornmentally qualified portions of the electrical penetration assembly are addressed in Sections 2.6 and 3.6, Electrical / Instrumentation & Control Components.

Two fuel transfer tubes penetrate the Reactor Building and link the refueling canal with the fuel transfer canal in the Spent Fuel Pool and serve as the underwater pathway for moving fuel into and out of the Reactor Building during refueling operations. The scope of review in Sections 2.3 and 3.3 includes the closure weld between the transfer tubes and the Reactor Building liner plate. The transfer tube, blind flange, and gate valve are part of the spent fuel pool system and are addressed in Sections 2.5 and 3.5, Mechanical Components.

2.3.3 Post-Tensioning System

The Reactor Building design incorporates a post-tensioning system that provides prestress forces to counteract forces resulting from design loads. The codes and standards used for the Reactor Building design and fabrication including applicable edition are given in the Oconee FSAR, Section 3.8. The Reactor Building post-tensioning buttresses and anchorage zone complies with the ACI 318-63. The dome is prestressed by 162 dome tendons. The cylinder wall is prestressed by 176 vertical tendons and by 632 hoop tendons. Each tendon is comprised of 90 wires bundled together. The wire material is ASTM A421-65, Type BA steel.

The primary components of the post-tensioning system are the tendon wires and associated anchorage. The tendons are continuous from anchorage to anchorage, and are deflected around penetrations. Conduits and bearing plates are cast into the concrete shell to receive the tendons which are installed after construction of the reinforced concrete is complete. The conduits are ungalvanized spiral wrapped corrugated thin wall sheathing. The tendon conduit provides the channel in the concrete through which the tendon is pulled, and contains the bulkfill grease. The uncoated conduit is protected from corrosion following construction by the alkaline environment provided by the concrete against the exterior of the conduit and by the bulkfill grease on the inside of the conduit. The bulkfill grease serves as an anti-corrosion medium for the tendon and anchorage.

Table 2.3-1 List of Reactor Building Components

Components Subject to Aging Management Review	Aging Management Review
Concrete Components • Dome • Cylinder Wall • Floor • Foundation Slab	See Section 3.3.1
Steel Components • Liner • Anchors/Embedments/Attachments • Personnel Hatch • Emergency Personnel Hatch • Equipment Hatch • Mechanical Penetrations • Electrical Penetrations • Fuel Transfer Tubes	See Section 3.3.2
Post-Tensioning System • Tendon Wires • Tendon Anchorage	See Section 3.3.3

2.4 REACTOR COOLANT SYSTEM

This section will be provided by April 1, 1997.

2.5 Mechanical Components

This section will be provided by April 1, 1997.

2.6 ELECTRICAL / INSTRUMENTATION & CONTROL COMPONENTS

This section will be provided by December 31, 1996.

2.7 STRUCTURES AND STRUCTURAL COMPONENTS

This section will be provided by December 31, 1996.

3. INTEGRATED PLANT ASSESSMENT - AGING MANAGEMENT REVIEW

3.1 INTRODUCTION

Part 54 requires that for each structure and component requiring an aging management review, the applicant must demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. §54.21(a)(3). Part 54 further requires that identified TLAA need to be evaluated for the period of extended operation. §54.21(c). The methods used by Duke to accomplish these requirements of Part 54 follow the guidance that has been provided in NEI 95-10, Sections 4 and 5, [2].

The structures and components that had been identified as requiring aging management reviews in Chapter 2 have been evaluated to determine if the effects of aging will be adequately managed for the period of extended operation. The following sections provide the demonstration required by §54.21(a)(3).

In addition, TLAA that were identified in Section 1.4 are evaluated concurrently with the aging effects evaluation on each structure, component, or component grouping. The results, as required by §54.21(c), are provided in the applicable sections of Chapter 3.

3.2 AGING MANAGEMENT REVIEW PROCESS OVERVIEW

Duke Power has performed the aging management reviews for the systems, structures, and components within the scope of Part 54 consistent with the guidance provided in NEI 95-10, Section 4.2. Applicable aging effects were determined by reviewing materials of construction, operating environment, operating stresses, industry experience, and NRC generic communications. Once the applicable aging effects were identified, existing programs and activities were reviewed to determine how they manage the applicable aging effects. Objective evidence of the effectiveness of existing programs and activities in managing the applicable aging effects provides reasonable assurance that the applicable aging effects are being managed. When necessary, enhancements to existing programs or new programs have been identified.

In addition to the above aging management review process, Duke Power has used the results of previous reviews. In particular, Duke Power has used the technical reports developed by a B&W Owners Group activity which are associated with the Reactor Coolant System. These technical reports also used the approach described above and, in several instances, have been reviewed and approved by the NRC.

The components of the Reactor Building that were identified as being subject to aging management review have been described in Section 2.3. The identification of the aging effects that are applicable to these components and the demonstration that these aging effects will be managed for the period of extended operation are described in Section 3.3.

Likewise, the components of the Reactor Coolant System that were identified as being subject to aging management review have been described in Section 2.4. The identification of the aging effects that are applicable to these components and the demonstration that these aging effects will be managed for the period of extended operation are described in Section 3.4.

Mechanical system components that are subject to aging management review have been identified in Section 2.5. The identification of the aging effects that are applicable to these components and the demonstration that these aging effects will be managed for the period of extended operation are described in Section 3.5.

The aging management review for electrical components is based on the plant spaces approach that is described in the Sandia Cable Aging Management Guideline (AMG) [4]. Electrical system components that are subject to aging management review were identified in Section 2.6. The identification of the aging effects that are applicable to these components and the demonstration that these aging effects will be managed for the period of extended operation are described in Section 3.6.

Finally, the structures and structural components that are subject to aging management review were identified in Section 2.7. The identification of the aging effects that are applicable to these structures and structural components and the demonstration that these aging effects will be managed for the period of extended operation are described in Section 3.7.

3.3 Reactor Building

Three Reactor Building component groups that are within the scope of license renewal and require aging management reviews were identified in Section 2.3. The aging management review consists of identifying the applicable aging effects for each of these Reactor Building component groups and then evaluating how existing programs and activities manage those effects. By reviewing the materials of construction and the environmental conditions, the applicable aging effects that can challenge the intended functions have been identified for each of the component groups. A review of industry operating experience relative to aging effects of the Reactor Building was also performed for each of the component groups in order to validate the identified aging effects. In addition, the industry reports on PWR Containment Structures [5] and Class 1 Structures [6] were also utilized to determine the applicable aging effects.

Identified aging effects were evaluated against the existing plant programs at Oconee. The demonstration process consists of evaluating the applicable existing programs with the guidance of NEI 95-10, Section 4.2 and reviewing actual Oconee specific results obtained from the implementation of these existing programs. These actual results provide objective evidence that these existing programs are effective in managing the identified effects of aging for the scope of components that are subject to aging management reviews.

If an aging effect was not found to be adequately managed by existing programs for the period of extended operation, then program enhancements were identified to manage these aging effects. While new programs may be required as a result of other aging management reviews, no new aging management programs were identified as a result of this aging management review of Oconee Reactor Building components. Table 3.3-1 identifies the credited Reactor Building aging management programs for Oconee.

The aging management review is complete when the credited programs provide reasonable assurance that the applicable aging effects are managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation. The process described in this section is intended to meet the requirements of §54.21(a)(3) and to permit the staff to make the finding identified in §54.29(a).

3.3.1 Concrete Components

3.3.1.1 Aging Management Review

3.3.1.1.1 Applicable Aging Effect

The Reactor Building concrete components were evaluated for the aging effects of loss of material, cracking, and change in material properties. The Reactor Building reinforced concrete components were designed in accordance with ACI 318-63 and constructed in accordance with ACI-301 using ingredients conforming to ACI and ASTM standards which provide a good quality, dense, low permeability concrete that precludes aging effects. Concrete mix designs incorporated low water-cement ratios, adequate air entrainment, water curing, and high quality materials. The protective concrete cover provided over the reinforcing bars met or exceeded the minimum requirements specified in ACI 318.

In addition, resistance to surface deterioration is enhanced by the application of prestress to the concrete sections. The prestressed concrete design places the cylinder wall and dome concrete in compression for all normal loading conditions over the period of extended operations. The precompression minimizes the number and width of shrinkage, temperature, or load induced cracks.

Inspections performed to date of the accessible exposed concrete surfaces have indicated a few instances of minor cracking and spalling. These concrete components have been repaired. The aging effects identified were minor and would not preclude Reactor Building components from performing their intended functions consistent with the CLB for the period of extended operation.

Based on the review performed, no aging effects were identified which would cause either above-grade and below-grade Oconee Reactor Building concrete components to lose their capability to perform their intended functions consistent with the CLB for the period of extended operation.

3.3.1.1.2 Aging Management Demonstration

Based on the aging management review that has been performed, Duke Power has determined that no aging management programs are required for Reactor Building concrete components for the period of extended operation.

3.3.1.2 Time-Limited Aging Analyses

There are no time-limited aging analyses associated with the Reactor Building concrete components.

3.3.2 Steel Components

3.3.2.1 Aging Management Review

3.3.2.1.1 Applicable Aging Effect

The Reactor Building steel components were evaluated for the aging effects of loss of material, cracking and change in material properties. The design of the steel components minimizes the potential for occurrence of detrimental aging effects. The applicable aging effect for Reactor Building steel components for the period of extended operation is loss of material due to corrosion for the liner, hatches, and penetrations if the coatings are not maintained; for the liner below the concrete floor if expansion joint sealants are not maintained; and for the liner behind welded attachments if the cavity formed between the attachment and the liner is not sealed. Cracking and change in material properties are precluded based on original design and the absence of an aggressive environment.

3.3.2.1.2 Aging Management Demonstration

The following three existing programs are being credited in license renewal to manage the potential loss of material due to corrosion of Reactor Building steel components for the period of extended operation:

- Reactor Building Civil Inspection
- Reactor Building Integrated Leak Rate Test
- Reactor Building Local Leak Rate

REACTOR BUILDING CIVIL INSPECTION

The Oconee Reactor Building Civil Inspection is a visual inspection which has been in effect since the initial startup of Oconee and is required pursuant to 10 CFR 50, Appendix J, Section V, Inspection and Reporting of Tests. The existing Reactor Building visual inspection program includes the key elements of an effective program, as identified in NEI 95-10, Section 4.2, necessary to assure that the Reactor Building maintains its intended function for the period of extended operation. The inspection includes all accessible areas of the Reactor Building including the liner, personnel hatches, the equipment hatch, and penetrations.

A total of 18 Reactor Building Civil Inspections have been performed to date at Oconee. Prior Reactor Building Civil Inspections have identified minor degradation including some minor local coating failures such as peeling or flaking. A small area of corrosion on the liner was observed at the interface of the base slab and the liner in a few locations, particularly under the equipment hatch. Local minor corrosion was also observed on a few isolated welded attachments to the liner and on a few penetrations. Coating degradation and minor local corrosion on steel components were repaired by an existing coatings maintenance procedure applicable to steel surfaces inside the Reactor Building. Previously identified locations of deterioration have been reinspected and a determination of the effectiveness of previous corrective actions has been made.

Based on the aging management review that has been performed, Duke Power has determined that the existing Reactor Building Civil Inspection Program provides reasonable assurance that the applicable aging effects will be managed so that the intended functions of the steel components of the Reactor Building will be maintained consistent with the CLB for the period of extended operation.

In addition to the above existing inspection program, NRC is in the final stages of rulemaking that will require plants to implement the requirements of ASME Section XI, Subsection IWE, 1992 Edition with the 1992 Addenda.

A summary description of the Reactor Building Civil Inspection program will be provided in the FSAR Supplement at the time of application, pursuant to the requirements of §54.21(d).

REACTOR BUILDING INTEGRATED LEAK RATE TEST

While the primary method for detection of aging effects is the Reactor Building Civil Inspection described above, additional assurance is provided by the Reactor Building Integrated Leak Rate Test (Type A ILRT) which is required by 10 CFR Part 50, Appendix J, Section III.A. and which would detect severe corrosion of the Reactor Building steel components. The existing Type A ILRT includes the key elements of an effective program, as identified in NEI 95-10, Section 4.2, necessary to assure that the Reactor Building maintains its intended function for the period of extended operation. The test includes all accessible areas of the Reactor Building including liner plate, personnel hatches, the equipment hatch, and penetrations. A total of 21 Type A ILRTs have been successfully performed at Oconee since initial startup. The Type A ILRTs are considered to be effective programs to manage the effects of aging and will continue to be conducted through the period of extended operation.

A summary description of the Type A Integrated Leak Rate Test program will be provided in the FSAR Supplement at the time of application, pursuant to the requirements of §54.21(d).

REACTOR BUILDING LOCAL LEAK RATE TEST

The primary method to detect corrosion is the Reactor Building Civil Inspection described above. The Reactor Building Local Leak Rate Test (Type B LLRT), required by 10 CFR Part 50, Appendix J, Section III.B., provides additional assurance that severe corrosion of certain penetrations including the personnel and equipment hatches is detected. The existing Type B LLRT includes the key elements of an effective program, as identified in NEI 95-10, Section 4.2 necessary to assure that the Reactor Building maintains its intended function for the period of extended operation.

Numerous Type B LLRT have been performed since Oconee start-up over 20 years ago. A review of the test results has shown that when a test failure has occurred, it has been caused by failure of the non-metallic components such as gaskets and seals and not the steel components themselves. Gaskets and seals are replaced when they fail the Type B LLRT. The Type B LLRT are considered to be effective programs to manage the effects of aging and will continue to be conducted through the period of extended operation.

A summary description of the Type B Local Leak Rate Test program will be provided in the FSAR Supplement at the time of application, pursuant to the requirements of §54.21(d).

3.3.2.2 TIME-LIMITED AGING ANALYSES

REACTOR BUILDING LINER PLATE AND PENETRATIONS - THERMAL FATIGUE

The interior surface of the Reactor Building is lined with welded steel plate to provide an essentially leak tight barrier. The liner plate is thickened at all penetrations to reduce stress concentrations. Design criteria are applied to the liner to assure that the specified leak rate is not exceeded under design basis accident conditions. The following fatigue loads, as described in Section 3.8.1.5.3 of the Oconee FSAR, were considered in the design of the liner plate and are considered to be time-limited aging analyses (TLAA) for the purposes of license renewal:

(a) Thermal cycling due to annual outdoor temperature variations. Number of cycles for this loading is 40 cycles for the plant life of 40 years.

(b) Thermal cycling due to Reactor Building interior temperature varying during the startup and shutdown of the Reactor Coolant System. The number of cycles for this loading is assumed to be 500 cycles.

(c) Thermal cycling due to the loss-of-coolant accident will be assumed to be one cycle.

(d) Thermal load cycles in the piping systems are somewhat isolated from the liner plate penetrations by concentric sleeves between the pipe and the liner plate. The attachment sleeve is designed in accordance with ASME Section III considerations. All penetrations are reviewed for a conservative number of cycles to be expected during the plant life.

Each of the above four TLAA have been evaluated for continued operation for up to 60 years. For item (a), an increase in the number of thermal cycles due to annual outdoor temperature variations from 40 to 60 cycles is considered to be insignificant in comparison to the assumed 500 thermal cycles due to Reactor Building interior temperature varying during heatup and cooldown of the Reactor Coolant System. Thus, this TLAA is considered to be valid for the period of extended operation.

For item (b), with respect to the assumed 500 thermal cycles due to startup and shutdown of the Reactor Coolant System, a more limiting number of thermal cycles is contained in Section 5.2 of the Oconee FSAR. Table 5.2 indicates a design limit of 360 heatup cycles and 360 cooldown cycles. The projected number of cycles for each Oconee unit through 60 years of operation has been determined to be less than the original 360 cycle design limits. This TLAA is considered to be valid for the period of extended operation because actual operating cycle values fall well within the assumed 500 thermal cycles due to startup and shutdown of the Reactor Coolant System.

For item (c), the assumed value for thermal cycling due to loss-of-coolant accident remains valid. None have occurred and none are expected to occur. This TLAA is considered to be valid for the period of extended operation.

Finally, for item (d), the design of the Reactor Building penetrations has been reviewed. The designs meet the general requirements of ASME Boiler and Pressure Vessel Code (1965), Section III, Nuclear Vessels. The only high temperature lines penetrating the Reactor Building wall and liner plate are the main feedwater and main steam lines. The design number of thermal load cycles in these two systems is bounded by the number of design heatup and cooldown cycles of the Reactor Coolant System. The projected number of cycles for each Oconee unit through 60 years of operation has been determined to be less than these original design limits. Thus, based on a review of the existing fatigue analysis, this TLAA is considered to be valid for the period of extended operation.

In conclusion, the existing analyses addressing thermal fatigue of the Reactor Building liner plate and penetrations is considered to be valid for the period of extended operation. The evaluation of this TLAA will be contained in the FSAR Supplement at the time of application, as required by §54.21(d).

3.3.3 Post-Tensioning System

3.3.3.1 Aging Management Review

3.3.3.1.1 Applicable Aging Effect

The Reactor Building post-tensioning systems components were evaluated for loss of material. Loss of material due to corrosion was determined to be an applicable aging effect for the tendon anchorage. Leakage of the anti-corrosion grease and moisture are some of the potential causes of the corrosion. Material loss at the tendon anchorage can ultimately lead to tendon failure if the corrosion progresses to the point of cracking of the tendon anchorage. Loss of prestress is evaluated as a TLAA in Section 3.3.3.2.

3.3.3.1.2 Aging Management Demonstration

Reactor Building Tendon Surveillance Program

The Oconee Reactor Building Tendon Surveillance Program has been in effect since the initial startup of Oconee and is required by the Oconee Technical Specifications. The existing tendon inspection and surveillance programs include the key elements of an effective program, as identified in NEI 95-10, Section 4.2, necessary to assure the post-tensioning system maintains its intended function for the extended period of operation.

A total of 18 (six for each of the three Reactor Building units) tendon surveillances have been performed to date at Oconee. The inspections have identified minor pitting corrosion on bearing plates (this condition existed at the time of installation and no deterioration has occurred since installation), minor grease leakage and concrete cracks. Coating degradation and minor local corrosion of the bearing plates were repaired by an existing maintenance procedure. A total of 54 tendon wires have been visually examined and found in excellent condition with no corrosion observed. A total of 162 wires have been tensile tested to date with no significant changes in ultimate strength or elongation of the wire as compared to results obtained during initial acceptance tests. No moisture has been detected and no change in grease coloring or condition has been noted. The observed aging effects are relatively minor and have no impact on the ability of the tendons to withstand the internal pressure resulting from a loss-of-coolant accident with no loss of integrity.

Grease leakage from the conduits or end caps may result in a reduction of the protective grease coverage on the tendon parts. The tendon inspection plan has assured the grease is providing adequate corrosion protection to the tendons. Where grease leakage has occurred, the inspections have shown that a residual protective coating of grease adheres to the metallic wire and anchorage components providing the necessary corrosion protection. Tendon grease leakage has been identified but no evidence exits to show that the bulkfill grease has any detrimental effect on the concrete.

In order to assure that the corrosion protection properties of the grease do not vary over time, the Reactor Building Tendon Surveillance programs include requirements to manage the effect of potential grease contamination by checking for free water in the end caps of the surveillance tendons. These inspections are sufficient to preclude damage due to degradation of the grease during the projected 60 year life of the plant.

Based on the aging management review that has been performed, Duke Power has determined that the existing Reactor Building Tendon Surveillance program provides reasonable assurance that the applicable aging effects will be managed so that the intended function of the post-tensioning system components will be maintained consistent with the CLB for the period of extended operation.

In addition to the above, NRC is in the final stages of rulemaking that will require plants to implement the tendon surveillance requirements of ASME Section XI, Subsection IWL, 1992 Edition with the 1992 Addenda.

A summary description of the tendon surveillance program will be provided in the FSAR Supplement at the time of application, pursuant to the requirements of §54.21(d).

3.3.3.2 TIME-LIMITED AGING ANALYSES

REACTOR BUILDING TENDONS - LOSS OF PRESTRESS

In accordance with ACI 318-63, the design of the Oconee Reactor Building post-tensioning system provides for prestress losses caused by the following:

1. Elastic shortening of concrete
2. Creep of concrete
3. Shrinkage of concrete
4. Relaxation of prestressing steel stress
5. Frictional loss due to intended or unintended curvature in the tendons

By assuming an appropriate initial jacking stress and using appropriate prestress loss parameters, the magnitude of the design losses and the final effective prestress at the end of 40 years for typical dome, vertical, and hoop tendons was calculated at the time of initial licensing. This analysis is considered to be a TLAA for the purposes of license renewal and is presently summarized in Section 3.8.1.5.2 of the Oconee FSAR.

The evaluation of this TLAA is being delayed until:

(a) The NRC review of the Oconee methodology for determining the most accurate minimum required lift-off force for each tendon group is completed; and
(b) The results of the next scheduled tendon surveillance are available (approximately May 1997).

Following completion of the above activities, NRC will be informed when the evaluation of this TLAA for the period of extended operation will be submitted.

3.3.4 Conclusion

Based on the aging management review that has been performed for the Oconee Reactor Building, Duke Power has determined that the programs which are listed in Table 3.3-1 provide reasonable assurance that the applicable aging effects will be managed so that the intended functions of the Reactor Building will be maintained consistent with the CLB for the period of extended operation.

Summary descriptions of the programs listed in Table 3.3-1 will be provided in the FSAR at the time of application, pursuant to the requirements of §54.21(d).

Table 3.3-1 Reactor Building Aging Management Programs

	Regulatory Document		
Aging Management Program	***10 CFR Part 50, §50.55a, (b)(2) (pending)***	***10 CFR Part 50, Appendix J***	***Existing Technical Specifications***
Reactor Building Civil Inspection	(ix) ASME - Subsection IWE, Subsection IWL	Section V.A.	4.4.1
Reactor Building Integrated Leak Rate Test	None	Section III.A.	4.4.1
Reactor Building Local Leak Test	None	Section III.B.	4.4.1
Reactor Building Tendon Surveillance Program	(ix) ASME - Subsection IWL	None	4.4.2

3.4 Reactor Coolant System

This section will be provided by April 1, 1997.

3.6 ELECTRICAL / INSTRUMENTATION & CONTROL COMPONENTS

This section will be provided by December 31, 1996.

3.7 Structures and Structural Components

This section will be provided December 31, 1996.

5. REFERENCES

1. Requirements for Renewal of Operating Licenses for Nuclear Power Plants, 10 CFR Part 54.

2. Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule, NEI 95-10, Revision 0, March 1996.

3. Oconee Nuclear Station, Final Safety Analysis Report, as revised.

4. Aging Management Guidelines for Commercial Nuclear Power Plants - Electrical Cable & Terminations, SAND 96-0344, April 1996.

5. NUMARC Report Number 90-01, " PWR Containment Structures License Renewal Industry Report", Revision 1, September 1991.

6. NUMARC Report Number 90-06, "Class I Structures License Renewal Industry Report", Revision 1, December 1991.