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ONS-2015-072

June 30, 2015

Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2746

10 CFR 50.71(e)
10 CFR 50.59(d)
10 CFR 54.37(b)

Subject: Duke Energy Carolinas, (LLC) (Duke Energy)
Oconee Nuclear Station, Units 1, 2, and 3
Docket Nos. 50-269, 50-270, 50-287
Updated Final Safety Analysis Report, Revision 24

Pursuant to 10 CFR 50.71(e), and in accordance with 10 CFR 50.4, Duke Energy hereby submits the Oconee Nuclear Station Updated Final Safety Analysis Report (UFSAR), Revision 24. The effective date of the UFSAR revision is December 31, 2014, as indicated at the bottom of each page. Changes made in Rev. 24 are indicated by side bars.

The Oconee UFSAR, Revision 24, is enclosed on one compact disk (CD). The contents are in Adobe Acrobat Portable Document Format (pdf). As required by NRC guidance for electronic submissions, Attachment 1 provides a listing of the document components that comprise the enclosed CD. Attachment 2 provides the List of Effective Pages for Tables and Figures. Attachment 3 provides insertion instructions for those receiving hardcopy distribution. Attachment 4 provides a listing of items removed in the 2014 UFSAR update.

Attachment 5 provides the report of changes, tests, and experiments performed pursuant to 10 CFR 50.59.

In addition, 10 CFR 54.37(b) requires that after the renewed license is issued, the UFSAR update must include any systems, structures and components (SSCs) newly identified that would have been subject to an aging management review or evaluation of time-limited aging analysis in accordance with 10 CFR 54.21. The UFSAR update must describe how the effects of aging are managed such that the intended function(s) in 10 CFR 54.4(b) will be effectively maintained during the period of extended operation. A review was completed to determine whether any newly-identified SSCs existed in support of submitting UFSAR Revision 24. As a result of this review, there were no newly-identified SSCs for which aging management reviews or time-limited aging analyses would apply.

This submittal document contains no new or revised regulatory commitments. If you have any questions regarding this submittal, please contact Susan Perry at (864) 873-4370.

A053
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 30, 2015.

Sincerely,

Scott L. Batson
Vice President
Oconee Nuclear Station

Attachments:

1. Document Components on CD
2. List of Effective Pages (LOEP) for Tables and Figures
3. Update Insertion Instructions (for hardcopy distribution only)
4. List of Removed Items
5. 10 CFR 50.59 Report

Enclosure:
CD: Oconee Nuclear Station Updated Final Safety Analysis Report, 2014 Update - Rev 24

cc: Mr. Victor McCree, NRC Region II Administrator (CD)
U.S. Nuclear Regulatory Commission
Marquis One Tower
245 Peachtree Center Ave., NE, Suite 1200
Atlanta, GA 30303-1257

Mr. James R. Hall, Project Manager (CD)
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
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Rockville, MD 20852

Mr. Eddy Crowe (hardcopy)
Senior Resident Inspector
Oconee Nuclear Station

Attachment 1
Document Components on CD
(2 pages)

Filename	File Size
000 ONS UFSAR Rev 24 Title Page	82 KB
001 ONS UFSAR Rev 24 Ch 1 Text	309 KB
002 ONS UFSAR Rev 24 Ch 1 Tables	159 KB
003 ONS UFSAR Rev 24 Ch 1 Figures	4,668 KB
004 ONS UFSAR Rev 24 Ch 2 Text	911 KB
005 ONS UFSAR Rev 24 Ch 2 Tables	4,788 KB
006 ONS UFSAR Rev 24 Ch 2 Figures	7,501 KB
007 ONS UFSAR Rev 24 Ch 3 Text	2,259 KB
008 ONS UFSAR Rev 24 Ch 3 Tables	1,727 KB
009 ONS UFSAR Rev 24 Ch 3 Figures	10,260 KB
010 ONS UFSAR Rev 24 Ch 4 Text	1,006 KB
011 ONS UFSAR Rev 24 Ch 4 Tables	1,202 KB
012 ONS UFSAR Rev 24 Ch 4 Figures	1,323 KB
013 ONS UFSAR Rev 24 Ch 5 Text	1,000 KB
014 ONS UFSAR Rev 24 Ch 5 Tables	1,686 KB
015 ONS UFSAR Rev 24 Ch 5 Figures	5,074 KB
016 ONS UFSAR Rev 24 Ch 6 Text	950 KB
017 ONS UFSAR Rev 24 Ch 6 Tables	1,964 KB
018 ONS UFSAR Rev 24 Ch 6 Figures	13,914 KB
019 ONS UFSAR Rev 24 Ch 7 Text	1,433 KB
020 ONS UFSAR Rev 24 Ch 7 Tables	366 KB
021 ONS UFSAR Rev 24 Ch 7 Figures	8,618 KB
022 ONS UFSAR Rev 24 Ch 8 Text	548 KB
023 ONS UFSAR Rev 24 Ch 8 Tables	441 KB
024 ONS UFSAR Rev 24 Ch 8 Figures	32,654 KB
025 ONS UFSAR Rev 24 Ch 9 Text	1,482 KB
026 ONS UFSAR Rev 24 Ch 9 Tables	1,029 KB
027 ONS UFSAR Rev 24 Ch 9 Figures	24,022 KB

Filename	File Size
028 ONS UFSAR Rev 24 Ch 10 Text	611 KB
029 ONS UFSAR Rev 24 Ch 10 Tables	141 KB
030 ONS UFSAR Rev 24 Ch 10 Figures	734 KB
031 ONS UFSAR Rev 24 Ch 11 Text	838 KB
032 ONS UFSAR Rev 24 Ch 11 Tables	479 KB
033 ONS UFSAR Rev 24 Ch 11 Figures	285 KB
034 ONS UFSAR Rev 24 Ch 12 Text	406 KB
035 ONS UFSAR Rev 24 Ch 12 Tables	210 KB
036 ONS UFSAR Rev 24 Ch 13 Text	676 KB
037 ONS UFSAR Rev 24 Ch 13 Tables	74 KB
038 ONS UFSAR Rev 24 Ch 13 Figures	422 KB
039 ONS UFSAR Rev 24 Ch 14 Text	449 KB
040 ONS UFSAR Rev 24 Ch 14 Tables	130 KB
041 ONS UFSAR Rev 24 Ch 15 Text	1,904 KB
042 ONS UFSAR Rev 24 Ch 15 Tables	2,210 KB
043 ONS UFSAR Rev 24 Ch 15 Figures	5,528 KB
044 ONS UFSAR Rev 24 Ch 16 Text	98 KB
045 ONS UFSAR Rev 24 Ch 17 Text	101 KB
046 ONS UFSAR Rev 24 Ch 18 Text	785 KB
047 ONS UFSAR Rev 24 Ch 18 Tables	104 KB

Attachment 2
List of Effective Pages (LOEP) for Tables and Figures
(40 pages)

OCONEE UFSAR - 2014 UPDATE

List of Effective Pages (LOEP) for Tables

The purpose of this list is to assure that the pages in the Tables section of your manual match the most recent issue, as well as to show a full accounting of all tables, including those that have been deleted. The earliest effective date, 12/31/00, was used when all tables were re-issued.

Effective Date	Table No.	Table Title
Chapter 1		
12/31/00	1-1	Key Dates in Oconee History
12/31/00	1-2	Engineered Safeguards Equipment
12/31/00	1-3	Deleted Per 1997 Update
Chapter 2		
12/31/00	2-1	1970 Population Distribution 0-10 Miles
12/31/00	2-2	2010 Projected Population Distribution 0-10 Miles
12/31/00	2-3	1970 Population Distribution 0-50 Miles
12/31/00	2-4	2010 Projected Population Distribution 0-50 Miles
12/31/00	2-5	1970 Cumulative Population Density 0-50 Miles
12/31/00	2-6	2010 Projected Cumulative Population Density 0-50 Miles
12/31/08	2-7	Frequency of Tropical Cyclones in Georgia, South Carolina and North Carolina Plus Coastal Waters
12/31/08	2-8	Mean Monthly Thunderstorm Days and Thunderstorms for Nuclear Plant Site
12/31/08	2-9	Duration and Frequency (in Hours) of Calm and Near-Calm Winds Average of Three Locations (1/59 - 12/63)
12/31/08	2-10	Annual Surface Wind Rose For Greenville, South Carolina (1/59 - 12/63)
12/31/08	2-11	Percent Frequency of Wind Speeds at Various Hours Through the Day - Greenville, S.C. (1/59 - 12/63)
12/31/08	2-12	Duration and Frequency of Calm and Near-Calm Winds Average of Three Locations (1/59 - 12/63)
12/31/08	2-13	Percentage Distribution of Athens, Georgia Annual Winds at 0630 Eastern Standard Time (800-1300 Feet Above Ground)
12/31/08	2-14	Percentage Distribution of Athens, Georgia Annual Winds at 0630 Eastern Standard Time (2300-2800 Feet Above Ground)
12/31/08	2-15	Average Wind Direction Change with Height, Athens, Georgia, by Lapse Rates in the Lowest 50 Meters-Two Years of Record

Effective Date	Table No.	Table Title
12/31/08	2-38	Supplemental Data Oconee Meteorological Survey (Tower Data) For Period of June 1, 1968 Thru May 31, 1969
12/31/08	2-39	Supplemental Data - Joint Frequency Distribution
12/31/08	2-40	Deleted Per 2008 Update
12/31/08	2-41	Deleted Per 2008 Update
12/31/08	2-42	Deleted Per 2008 Update
12/31/08	2-43	Deleted Per 2008 Update
12/31/08	2-44	Supplemental Data - SF_6 Detector Readings - Test Date: January 28, 1970
12/31/08	2-45	Deleted Per 2008 Update
12/31/08	2-46	Deleted Per 2008 Update
12/31/08	2-47	Deleted Per 2008 Update
12/31/08	2-48	Deleted Per 2008 Update
12/31/08	2-49	Deleted Per 2008 Update
12/31/08	2-50	Deleted Per 2008 Update
12/31/08	2-51	Deleted Per 2008 Update
12/31/08	2-52	Deleted Per 2008 Update
12/31/08	2-53	Deleted Per 2008 Update
12/31/08	2-54	Deleted Per 2008 Update
12/31/08	2-55	Deleted Per 2008 Update
12/31/08	2-56	Deleted Per 2008 Update
12/31/08	2-57	Deleted Per 2008 Update
12/31/08	2-58	Deleted Per 2008 Update
12/31/08	2-59	Deleted Per 2008 Update
12/31/08	2-60	Deleted Per 2008 Update
12/31/08	2-61	Deleted Per 2008 Update
12/31/08	2-62	Deleted Per 2008 Update
12/31/08	2-63	Deleted Per 2008 Update
12/31/08	2-64	Deleted Per 2008 Update
12/31/08	2-65	Deleted Per 2008 Update
12/31/08	2-66	Deleted Per 2008 Update

Effective Date	Table No.	Table Title
12/31/08	2-67	Deleted Per 2008 Update
12/31/08	2-68	Deleted Per 2008 Update
12/31/08	2-69	Deleted Per 2008 Update
12/31/08	2-70	Deleted Per 2008 Update
12/31/08	2-71	Deleted Per 2008 Update
12/31/08	2-72	Deleted Per 2008 Update
12/31/08	2-73	Deleted Per 2008 Update
12/31/08	2-74	Deleted Per 2008 Update
12/31/08	2-75	Deleted Per 2008 Update
12/31/08	2-76	Deleted Per 2008 Update
12/31/08	2-77	Deleted Per 2008 Update
12/31/08	2-78	Deleted Per 2008 Update
12/31/08	2-79	Deleted Per 2008 Update
12/31/08	2-80	Deleted Per 2008 Update
12/31/08	2-81	Deleted Per 2008 Update
12/31/08	2-82	Deleted Per 2008 Update
12/31/08	2-83	Deleted Per 2008 Update
12/31/08	2-84	Deleted Per 2008 Update
12/31/08	2-85	Deleted Per 2008 Update
12/31/08	2-86	Deleted Per 2008 Update
12/31/08	2-87	Deleted Per 2008 Update
12/31/08	2-88	Deleted Per 2008 Update
12/31/08	2-89	Deleted Per 2008 Update
12/31/08	2-90	Deleted Per 2008 Update
12/31/08	2-91	Deleted Per 2008 Update
12/31/08	2-92	Deleted Per 2008 Update
12/31/00	2-93	Soil Permeability Test Results
12/31/00	2-94	Significant Earthquakes in the Southeast United States (Intensity V or Greater)
12/31/00	2-95	Velocity Measurements
12/31/00	2-96	Core Measurements

Effective Date	Table No.	Table Title
Chapter 3		
12/31/04	3-1	System Piping Classification
12/31/14	3-2	System Component Classification
12/31/04	3-3	Summary of Missile Equations
12/31/00	3-4	List of Symbols
12/31/14	3-5	Properties of Missiles - Reactor Vessel & Control Rod Drive
12/31/03	3-6	Properties of Missiles - Steam Generator
12/31/00	3-7	Properties of Missiles - Pressurizer
12/31/00	3-8	Properties of Missiles - Quench Tank and Instruments
12/31/00	3-9	Properties of Missiles - System Piping
12/31/00	3-10	Missile Characteristics
12/31/00	3-11	Depth of Penetration of Concrete
12/31/00	3-12	Containment Coatings
12/31/00	3-13	Service Load Combinations for Reactor Building
12/31/00	3-14	Accident, Wind, and Seismic Load Combinations and Factors for Class 1 Concrete Structures
12/31/00	3-15	Inward Displacement of Liner Plate
12/31/03	3-16	Stress Analysis Results
12/31/00	3-17	Stress Analysis Results
12/31/03	3-18	Stress Analysis Results
12/31/03	3-19	Stress Analysis Results
12/31/03	3-20	Stress Analysis Results
12/31/03	3-21	Stress Analysis Results
12/31/00	3-22	Bent Wire Test Results
12/31/09	3-23	Auxiliary Building Loads and Conditions
12/31/00	3-24	Mark-BZ Fuel Assembly Seismic and LOCA Results at 600°F
12/31/00	3-25	Deleted Per 1996 Update
12/31/00	3-26	Stress Limits for Seismic, Pipe Rupture and Combined Loads
12/31/00	3-27	Deleted Per 1999 Update
12/31/00	3-28	Deleted Per 2004 Update

Effective Date	Table No.	Table Title
12/31/00	3-29	Deleted Per 2004 Update
12/31/00	3-30	Deleted Per 2004 Update
12/31/00	3-31	Deleted Per 2004 Update
12/31/00	3-32	Deleted Per 2004 Update
12/31/00	3-33	Deleted Per 2004 Update
12/31/00	3-34	Deleted Per 2004 Update
12/31/00	3-35	Deleted Per 2004 Update
12/31/00	3-36	Deleted Per 2004 Update
12/31/00	3-37	Deleted Per 2004 Update
12/31/00	3-38	Deleted Per 2004 Update
12/31/00	3-39	Deleted Per 2004 Update
12/31/00	3-40	Deleted Per 2004 Update
12/31/00	3-41	Deleted Per 2004 Update
12/31/00	3-42	Deleted Per 2004 Update
12/31/00	3-43	Deleted Per 2004 Update
12/31/00	3-44	Deleted Per 2004 Update
12/31/00	3-45	Deleted Per 2004 Update
12/31/00	3-46	Deleted Per 2004 Update
12/31/00	3-47	Deleted Per 2004 Update
12/31/00	3-48	Deleted Per 2004 Update
12/31/00	3-49	Deleted Per 2004 Update
12/31/00	3-50	Deleted Per 2004 Update
12/31/00	3-51	Deleted Per 2004 Update
12/31/00	3-52	Deleted Per 2004 Update
12/31/00	3-53	Deleted Per 2004 Update
12/31/00	3-54	Deleted Per 2004 Update
12/31/00	3-55	Deleted Per 2004 Update
12/31/00	3-56	Deleted Per 2004 Update
12/31/00	3-57	Deleted Per 2004 Update
12/31/00	3-58	Deleted Per 2004 Update

Effective Date	Table No.	Table Title
12/31/00	3-59	Deleted Per 2004 Update
12/31/00	3-60	Deleted Per 2004 Update
12/31/00	3-61	Deleted Per 2004 Update
12/31/00	3-62	Deleted Per 2004 Update
12/31/00	3-63	Deleted Per 2004 Update
12/31/00	3-64	Deleted Per 2004 Update
12/31/00	3-65	Deleted Per 2004 Update
12/31/00	3-66	Deleted Per 2004 Update
12/31/00	3-67	Deleted Per 2004 Update
12/31/13	3-68	Electrical Equipment Seismic Qualification
Chapter 4		
12/31/09	4-1	Core Design, Thermal, and Hydraulic Data
12/31/09	4-2	Fuel Assembly Components
12/31/12	4-3	Nuclear Design Data
12/31/12	4-4	Typical Fuel Cycle Excess Reactivity, HFP Samarium
12/31/00	4-5	Effective Multiplication Factor
12/31/12	4-6	Shutdown Margin Calculation for Typical Oconee Fuel Cycle
12/31/09	4-7	Moderator Temperature Coefficient (For the First Cycle)
12/31/00	4-8	BOL Distributed-Temperature Moderator Coefficients, 100% Power, 1200 ppm Boron (O1C01)
12/31/12	4-9	BOL Distributed-Temperature Moderator Coefficients, vs. Power, No Xenon
12/31/12	4-10	BOL Distributed-Temperature Moderator Coefficient, 100% Full Power
12/31/12	4-11	Power Coefficients of Reactivity
12/31/00	4-12	pH Characteristics
12/31/00	4-13	Design Methods
12/31/00	4-14	Deleted Per 1999 Update
12/31/00	4-15	Deleted Per 1997 Update
12/31/00	4-16	Internals Vent Valve Materials
12/31/00	4-17	Vent Valve Shaft & Bushing Clearances
12/31/00	4-18	Control Rod Assembly Data

Effective Date	Table No.	Table Title
12/31/04	5-21	Reactor Coolant Piping Design Data
12/31/06	5-22	Pressurizer Design Data
12/31/00	5-23	Operating Design Transient Cycles for Pressurizer Surge Line
12/31/00	5-24	Evaluation of Reactor Vessel Pressurized Thermal Shock Toughness Properties at 48 EFPY - Oconee Unit 1
12/31/00	5-25	Evaluation of Reactor Vessel Pressurized Thermal Shock Toughness Properties at 48 EFPY - Oconee Unit 2
12/31/00	5-26	Evaluation of Reactor Vessel Pressurized Thermal Shock Toughness Properties at 48 EFPY - Oconee Unit 3
12/31/00	5-27	Evaluation of Reactor Vessel Extended Life (48EFPY) Charpy V-Notch Upper-Shelf Energy - Oconee Unit 1
12/31/00	5-28	Evaluation of Reactor Vessel Extended Life (48EFPY) Charpy V-Notch Upper-Shelf Energy - Oconee Unit 2
12/31/00	5-29	Evaluation of Reactor Vessel Extended Life (48EFPY) Charpy V-Notch Upper-Shelf Energy - Oconee Unit 3
Chapter 6		
12/31/00	6-1	Deleted Per 1995 Update
12/31/00	6-2	Deleted Per 2000 Update
12/31/00	6-3	Quality Control Standards for Engineered Safeguards Systems
12/31/04	6-4	Engineered Safeguards Piping Design Conditions
12/31/00	6-5	Single Failure Analysis Reactor Building Spray System
12/31/00	6-6	Single Failure Analysis For Reactor Building Cooling System
12/31/14	6-7	Reactor Building Penetration Valve Information
12/31/05	6-8	High Pressure Injection System Component Data
12/31/09	6-9	Low Pressure Injection System Component Data
12/31/00	6-10	Core Flooding System Components Data
12/31/05	6-11	Single Failure Analysis - Emergency Core Cooling System
12/31/00	6-12	Oconee Nuclear Station Analysis of Valve Motors Which May Become Submerged Following A LOCA
12/31/00	6-13	Equipment Operational During An Accident and Located Outside Containment
12/31/00	6-14	Equipment Operational During an Accident and Located Within the Containment

Effective Date	Table No.	Table Title
12/31/05	6-15	Emergency Core Cooling Systems Performance Testing
12/31/00	6-16	Deleted Per 1999 Update
12/31/00	6-17	Deleted Per 1999 Update
12/31/00	6-18	Inventory of Iodine Isotopes in Reactor Building (at t = o)
12/31/05	6-19	Single Failure Analysis for Reactor Building Penetration Room Ventilation System
12/31/00	6-20	Parameters for Boron Precipitation Analysis
12/31/03	6-21	Summary of Calculated Containment Pressures and Temperatures for LOCA Cases
12/31/13	6-22	Containment Response Analyses Initial Conditions
12/31/13	6-23	Containment Structural Heat Sink Data
12/31/13	6-24	Accident Chronology for Limiting Break for Equipment Qualification
12/31/13	6-25	Minimum Acceptable Combinations of Containment Heat Removal Equipment Performance
12/31/13	6-26	Engineered Safety Feature Assumptions in Containment Response Analyses
12/31/08	6-27	Summary of Calculated Containment Pressures and Temperatures for Secondary System Pipe Rupture Cases
12/31/00	6-28	Steam Generator Compartment Pressure Response Flowpath Discharge Coefficients
12/31/03	6-29	Peak Pressure Mass and Energy Release Data
12/31/03	6-30	RELAP5 Long-Term Mass and Energy Release Data
12/31/13	6-31	BFLOW/FATHOMS Long-Term Mass and Energy Releases
12/31/03	6-32	Steam Line Break Mass and Energy Releases for Double-Ended Guillotine Break
12/31/05	6-33	NPSH Available and Required for LPI and BS Pumps (Limiting Flow Case)
12/31/08	6-34	Deleted Per 2008 Update
12/31/03	6-35	ROTSG Peak Pressure Mass and Energy Release Data
Chapter 7		
12/31/12	7-1	Reactor Trip Summary
12/31/12	7-2	Engineered Safeguards Actuation Conditions
12/31/12	7-3	Engineered Safeguards Actuated Devices

Effective Date	Table No.	Table Title
12/31/13	7-4	Characteristics of Out-of-Core Neutron Detector Assemblies
12/31/06	7-5	NNI Inputs to Engineered Safeguards
12/31/05	7-6	ICS Transient Limits
Chapter 8		
12/31/02	8-1	Loads to be Supplied from the Emergency Power Source
12/31/00	8-2	Single Failure Analysis for 125 Volt DC Switching Station Power Systems
12/31/14	8-3	Single Failure Analysis for the Keowee Hydro Station
12/31/14	8-4	Single Failure Analysis for the Emergency Electrical Power Systems
12/31/00	8-5	Single Failure Analysis for 125 Volt DC Instrumentation and Control Power System
12/31/00	8-6	Single Failure Analysis for the 120 Volt AC Vital Power System
12/31/00	8-7	125 Volt DC Panelboard Fault Analysis
Chapter 9		
12/31/00	9-1	Spent Fuel Cooling System Data, Units 1, 2
12/31/00	9-2	Spent Fuel Cooling System Data, Oconee 3
12/31/00	9-3	Component Cooling System Performance Data (For Normal Operation on a Per Oconee Basis)
12/31/00	9-4	Cooling Water Systems Component Data (Component Data on a Per Unit Basis)
12/31/06	9-5	Chemical Addition and Sampling System Component Data
12/31/11	9-6	High Pressure Injection System Performance Data
12/31/05	9-7	High Pressure Injection System Component Data
12/31/00	9-8	Low Pressure Injection System Performance Data
12/31/09	9-9	Low Pressure Injection System Component Data
12/31/00	9-10	Coolant Storage System Component Data (Component Quantities for Three Units)
12/31/00	9-11	Ventilation System Major Component Data
12/31/02	9-12	Deleted Per 2002 Update
12/31/00	9-13	Component Cooling System Component Data (Component Data on a Per Unit Basis)
12/31/10	9-14	SSF System Main Components

Effective Date	Table No.	Table Title
12/31/10	9-15	SSF Primary Valves
12/31/11	9-16	SSF Instrumentation
12/31/00	9-17	Design Basis Tornado Missiles And Their Impact Velocities
12/31/00	9-18	Design Basis Tornado Missiles Minimum Barrier Thicknesses
12/31/00	9-19	Codes and Specifications For Design of Category I Structures
Chapter 10		
12/31/00	10-1	Condensate/Feedwater Reserves (each unit)
12/31/09	10-2	Parameter Indication Location for EFW System
Chapter 11		
12/31/00	11-1	Potential Radioactive Waste Quantities from Three Units
12/31/00	11-2	Estimated Maximum Rate of Accumulation Radioactive Wastes Per Operation
12/31/00	11-3	Yearly Average Activity Concentrations in the Station Effluent for Three Units, Each Operating with One Percent Defective Fuel
12/31/04	11-4	Escape Rate Coefficients for Fission Product Release
12/31/00	11-5	Reactor Coolant Activity
12/31/12	11-6	Waste Disposal System Component Data (Component Quantities for Three Units)
12/31/04	11-7	Process Radiation Monitors
Chapter 12		
12/31/00	12-1	Parameters Used for Shielding Analyses
12/31/00	12-2	Principal Shielding
12/31/04	12-3	Area Radiation Monitors
Chapter 13		
12/31/00	13-1	Deleted Per 1991 Update
12/31/00	13-2	Deleted Per 1999 Update
Chapter 14		
12/31/00	14-1	Tests Prior to Initial Fuel Loading
12/31/00	14-2	Postcriticality Tests

Effective Date	Table No.	Table Title
Chapter 15		
12/31/10	15-1	Reg. Guide 1.183 Fuel Handling Accident Source Term
12/31/13	15-2	Rod Ejection Accident Analysis Results
12/31/08	15-3	Deleted Per 2008 Update
12/31/04	15-4	Deleted Per 2004 Update
12/31/03	15-5	Steam Line Break Accident - With Offsite Power Case Sequence of Events
12/31/00	15-6	Summary of LOCA Break Spectrum Break Size and Type
12/31/00	15-7	Deleted Per 1997 Update
12/31/00	15-8	Deleted Per 1995 Update
12/31/00	15-9	Deleted Per 1995 Update
12/31/00	15-10	Deleted Per 1995 Update
12/31/00	15-11	Deleted Per 1995 Update
12/31/00	15-12	Deleted Per 1995 Update
12/31/00	15-13	Deleted Per 1995 Update
12/31/00	15-14	Deleted Per 2004 Update
12/31/12	15-15	Total Core Activity for Maximum Hypothetical Accident
12/31/12	15-16	Summary of Transient and Accident Doses Including the Effects of High Burnup Reload Cores with Replacement Steam Generators
12/31/00	15-17	Deleted Per 2000 Update
12/31/00	15-18	Deleted Per 2000 Update
12/31/00	15-19	Deleted Per 1995 Update
12/31/00	15-20	Deleted Per 1995 Update
12/31/00	15-21	Deleted Per 1995 Update
12/31/00	15-22	Deleted Per 1995 Update
12/31/00	15-23	Deleted Per 1995 Update
12/31/00	15-24	Deleted Per 1997 Update
12/31/00	15-25	Deleted Per 2001 Update
12/31/00	15-26	Deleted Per 1995 Update
12/31/00	15-27	Deleted Per 2003 Update
12/31/03	15-28	HPI Flow Assumed in Core Flood Line Small Break LOCA Analyses

Effective Date	Table No.	Table Title
12/31/03	15-29	HPI Flow Assumed in RCP Discharge Small Break LOCA Analyses
12/31/03	15-30	HPI Flow Assumed in HPI Line Small Break LOCA Analyses
12/31/08	15-31	Deleted Per 2008 Update
12/31/14	15-32	Summary of Transient and Accident Cases Analyzed
12/31/13	15-33	Methodology Topical Reports and Computer Codes Used in Analyses
12/31/14	15-34	Summary of Input Parameters for Accident Analyses Using Computer Codes
12/31/12	15-35	Trip Setpoints and Time Delays Assumed in Accident Analyses
12/31/13	15-36	Startup Accident Sequence of Events
12/31/08	15-37	Rod Withdrawal at Power Accident - Peak RCS Pressure Analysis Sequence of Events
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List of Effective Pages (LOEP) for Figures

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12/31/00	15-57	Deleted Per 1995 Update
12/31/00	15-58	Deleted Per 1995 Update
12/31/00	15-59	Deleted Per 1995 Update
12/31/00	15-60	Deleted Per 1995 Update
12/31/00	15-61	Deleted Per 1995 Update
12/31/00	15-62	Deleted Per 1995 Update
12/31/00	15-63	Deleted Per 1995 Update
12/31/00	15-64	Deleted Per 1995 Update
12/31/00	15-65	Deleted Per 1995 Update
12/31/00	15-66	Deleted Per 1995 Update
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12/31/00	15-105	Deleted Per 1995 Update
12/31/00	15-106	Deleted Per 1995 Update
12/31/00	15-107	Deleted Per 1995 Update
12/31/00	15-108	Deleted Per 1995 Update
12/31/00	15-109	Deleted Per 1995 Update
12/31/00	15-110	Deleted Per 2001 Update
12/31/00	15-111	Deleted Per 2003 Update
12/31/14	15-112	Deleted Per 2014 Update
12/31/03	15-113	Rod Withdrawal at Power Accident - Core Cooling Capability Analysis RCS Pressure
12/31/13	15-114	Rod Withdrawal at Power Accident - Core Cooling Capability Analysis DNBR
12/31/08	15-115	Cold Water Accident - Core Average Temperature
12/31/08	15-116	Cold Water Accident - Power
12/31/08	15-117	Cold Water Accident - Cold Leg Temperature
12/31/08	15-118	Cold Water Accident - RCS Pressure
12/31/03	15-119	Loss of Coolant Flow Accidents - Two RCP Coastdown from Four RCP Initial Conditions Analysis - Power
12/31/03	15-120	Loss of Coolant Flow Accidents - Two RCP Coastdown from Four RCP Initial Conditions Analysis - RCS Temperature
12/31/03	15-121	Loss of Coolant Flow Accidents - Two RCP Coastdown from Four RCP Initial Conditions Analysis - Pressurizer Level
12/31/03	15-122	Loss of Coolant Flow Accidents - Two RCP Coastdown from Four RCP Initial Conditions Analysis - RCS Pressure
12/31/13	15-123	Loss of Coolant Flow Accidents - Two RCP Coastdown from Four RCP Initial Conditions Analysis - DNBR
12/31/03	15-124	Loss of Coolant Flow Accidents - One RCP Coastdown from Three RCP Initial Conditions Analysis - RCS Flow
12/31/03	15-125	Loss of Coolant Flow Accidents - One RCP Coastdown from Three RCP Initial Conditions Analysis - Power
12/31/03	15-126	Loss of Coolant Flow Accidents - One RCP Coastdown from Three RCP Initial Conditions Analysis - RCS Temperature

Effective Date	Figure No.	Figure Title
12/31/03	15-127	Loss of Coolant Flow Accidents - One RCP Coastdown from Three RCP Initial Conditions Analysis - Pressurizer Level
12/31/03	15-128	Loss of Coolant Flow Accidents - One RCP Coastdown from Three RCP Initial Conditions Analysis - RCS Pressure
12/31/13	15-129	Loss of Coolant Flow Accidents - One RCP Coastdown from Three RCP Initial Conditions Analysis - DNBR
12/31/03	15-130	Loss of Coolant Flow Accidents - Locked Rotor From Four RCP Initial Conditions Analysis - RCS Flow
12/31/03	15-131	Loss of Coolant Flow Accidents - Locked Rotor From Four RCP Initial Conditions Analysis - Power
12/31/03	15-132	Loss of Coolant Flow Accidents - Locked Rotor From Four RCP Initial Conditions Analysis - RCS Temperature
12/31/03	15-133	Loss of Coolant Flow Accidents - Locked Rotor From Four RCP Initial Conditions Analysis - Pressurizer Level
12/31/03	15-134	Loss of Coolant Flow Accidents - Locked Rotor From Four RCP Initial Conditions Analysis - RCS Pressure
12/31/08	15-135	Loss of Coolant Flow Accidents - Locked Rotor From Four RCP Initial Conditions Analysis - DNBR
12/31/08	15-136	Loss of Coolant Flow Accidents - Locked Rotor From Three RCP Initial Conditions Analysis - RCS Flow
12/31/08	15-137	Loss of Coolant Flow Accidents - Locked Rotor From Three RCP Initial Conditions Analysis - Power
12/31/08	15-138	Loss of Coolant Flow Accidents - Locked Rotor From Three RCP Initial Conditions Analysis - RCS Temperatures
12/31/08	15-139	Loss of Coolant Flow Accidents - Locked Rotor From Three RCP Initial Conditions Analysis - Pressurizer Level
12/31/08	15-140	Loss of Coolant Flow Accidents - Locked Rotor From Three RCP Initial Conditions Analysis - RCS Pressure
12/31/11	15-141	Loss of Coolant Flow Accidents - Locked Rotor From Three RCP Initial Conditions Analysis - DNBR
12/31/00	15-142	Intentionally Blank
12/31/10	15-143	Control Rod Misalignment Accidents - Dropped Rod - RCS Pressure
12/31/13	15-144	Control Rod Misalignment Accidents - Dropped Rod - DNBR
12/31/13	15-145	Turbine Trip Accident - Steam Generator Pressure
12/31/13	15-146	Turbine Trip Accident - RCS Temperatures

Effective Date	Figure No.	Figure Title
12/31/13	15-147	Turbine Trip Accident - Pressurizer Level
12/31/13	15-148	Turbine Trip Accident - RCS Pressure
12/31/13	15-149	Turbine Trip Accident - Power
12/31/11	15-150	Steam Generator Tube Rupture - Power
12/31/11	15-151	Steam Generator Tube Rupture - Break Flow
12/31/11	15-152	Steam Generator Tube Rupture - RCS Pressure
12/31/11	15-153	Steam Generator Tube Rupture - Pressurizer Level
12/31/11	15-154	Steam Generator Tube Rupture - Steam Generator Pressure
12/31/11	15-155	Steam Generator Tube Rupture - Steam Generator Level
12/31/11	15-156	Steam Generator Tube Rupture - RCS Temperatures
12/31/03	15-157	Steam Line Break Accident - With Offsite Power - Power
12/31/03	15-158	Steam Line Break Accident - With Offsite Power - RCS Pressure
12/31/03	15-159	Steam Line Break Accident - With Offsite Power - Core Inlet Flow
12/31/03	15-160	Deleted Per 2003 Update
12/31/03	15-161	Steam Line Break Accident - Without Offsite Power - Steam Line Pressure
12/31/03	15-162	Steam Line Break Accident - Without Offsite Power - RCS Temperatures
12/31/03	15-163	Steam Line Break Accident - Without Offsite Power - RCS Flow
12/31/03	15-164	Steam Line Break Accident - Without Offsite Power - Reactivity
12/31/03	15-165	Steam Line Break Accident - Without Offsite Power - Power
12/31/03	15-166	Steam Line Break Accident - Without Offsite Power - RCS Pressure
12/31/13	15-167	Steam Line Break Accident - Without Offsite Power - DNBR
12/31/03	15-168	Small Steam Line Break - Steam Mass Flows
12/31/03	15-169	Small Steam Line Break - Steam Line Pressures
12/31/11	15-170	Small Steam Line Break - Main Feedwater Mass Flows
12/31/03	15-171	Small Steam Line Break - RCS Temperatures
12/31/03	15-172	Small Steam Line Break - Core Average Power
12/31/03	15-173	Small Steam Line Break - RCS Hot Leg Pressure
12/31/14	15-174	Deleted Per 2014 Update
12/31/03	15-175	Oconee - No CHRS Flow

Effective Date	Figure No.	Figure Title
12/31/01	15-176	Deleted Per 2001 Update
12/31/03	15-177	Lower Bound Containment Pressure Used in Large Break LOCA
12/31/14	15-178	Deleted Per 2014 Update
12/31/14	15-179	Deleted Per 2014 Update
12/31/14	15-180	Deleted Per 2014 Update
12/31/14	15-181	Deleted Per 2014 Update
12/31/14	15-182	Deleted Per 2014 Update
12/31/14	15-183	Deleted Per 2014 Update
12/31/14	15-184	Deleted Per 2014 Update
12/31/14	15-185	Deleted Per 2014 Update
12/31/14	15-186	Deleted Per 2014 Update
12/31/14	15-187	Deleted Per 2014 Update
12/31/14	15-188	Deleted Per 2014 Update
12/31/14	15-189	Deleted Per 2014 Update
12/31/14	15-190	Deleted Per 2014 Update
12/31/14	15-191	Deleted Per 2014 Update
12/31/14	15-192	Deleted Per 2014 Update
12/31/14	15-193	Deleted Per 2014 Update
12/31/14	15-194	Deleted Per 2014 Update
12/31/14	15-195	Deleted Per 2014 Update
12/31/14	15-196	Deleted Per 2014 Update
12/31/14	15-197	Deleted Per 2014 Update
12/31/14	15-198	Deleted Per 2014 Update
12/31/14	15-199	Deleted Per 2014 Update
12/31/14	15-200	Deleted Per 2014 Update
12/31/14	15-201	Deleted Per 2014 Update
12/31/14	15-202	Deleted Per 2014 Update
12/31/14	15-203	Deleted Per 2014 Update
12/31/14	15-204	Deleted Per 2014 Update
12/31/14	15-205	Deleted Per 2014 Update

Effective Date	Figure No.	Figure Title
12/31/14	15-206	Deleted Per 2014 Update
12/31/14	15-207	Deleted Per 2014 Update
12/31/14	15-208	Deleted Per 2014 Update
12/31/14	15-209	Deleted Per 2014 Update
12/31/14	15-210	Deleted Per 2014 Update
12/31/14	15-211	Deleted Per 2014 Update
12/31/14	15-212	Deleted Per 2014 Update
12/31/08	15-213	77% of 2568 MWt, Mark-B-HTP Mixed-Core SBLOCA Spectrum Analysis
12/31/08	15-214	0.075 ft^2 CLPD, 77% of 2568 MWt, Mark-B-HTP Mixed-Core SBLOCA Case-Pressure
12/31/08	15-215	0.075 ft^2 CLPD, 77% of 2568 MWt, Mark-B-HTP Mixed-Core SBLOCA Case- Break and ECCS Mass Flow Rates
12/31/08	15-216	0.075 ft^2 CLPD, 77% of 2568 MWt, Mark-B-HTP Mixed-Core SBLOCA Case- Hot Channel Levels
12/31/08	15-217	0.075 ft^2 CLPD, 77% of 2568 MWt, Mark-B-HTP Mixed-Core SBLOCA Case- Peak Cladding Temperature
12/31/08	15-218	0.075 ft^2 CLPD, 77% of 2568 MWt, Mark-B-HTP Mixed-Core SBLOCA Case- HC Vapor Temperature at Core Exit
12/31/11	15-219	Mark-B-HTP Full-Core BOL LBLOCA - Reactor Vessel Upper Plenum Pressure
12/31/11	15-220	Mark-B-HTP Full-Core BOL LBLOCA - Break Mass Flow Rates
12/31/11	15-221	Mark-B-HTP Full-Core BOL LBLOCA - Hot Channel Mass Flow Rates
12/31/11	15-222	Mark-B-HTP Full-Core BOL LBLOCA - Core Flooding Rates
12/31/11	15-223	Mark-B-HTP Full-Core BOL LBLOCA - Hot Pin Fuel & Clad Temperatures at Ruptured Location
12/31/11	15-224	Mark-B-HTP Full-Core BOL LBLOCA - Hot Pin Fuel & Clad Temperatures at Unruptured Location
12/31/11	15-225	Mark-B-HTP Full-Core BOL LBLOCA - Quench Front Advancement
12/31/11	15-226	Mark-B-HTP Full-Core BOL LBLOCA - Hot Pin Heat Transfer Coefficients
12/31/11	15-227	102% of 2568 MWt, Full Core Mark-B-HTP SBLOCA Break Spectrum Analysis

Effective Date	Figure No.	Figure Title
12/31/11	15-228	$0.15ft^2$ CLPD, 102% of 2568 MWt, Full Core Mark-B-HTP SBLOCA - Pressure
12/31/11	15-229	$0.15ft^2$ CLPD, 102% of 2568 MWt, Full Core Mark-B-HTP SBLOCA - Break and ECCS Mass Flow Rates
12/31/11	15-230	$0.15ft^2$ CLPD, 102% of 2568 MWt, Full Core Mark-B-HTP SBLOCA - RV Collapsed Liquid Level & Hot Channel Mixture Level
12/31/11	15-231	$0.15ft^2$ CLPD, 102% of 2568 MWt, Full Core Mark-B-HTP SBLOCA - Hot Pin Peak Clad Temperature
12/31/11	15-232	$0.15ft^2$ CLPD, 102% of 2568 MWt, Full Core Mark-B-HTP SBLOCA - Hot Channel Vapor Temperature at Core Exit

Attachment 3
Update Insertion Instructions
(for hardcopy distribution only)
(3 pages)

Update Insertion Instructions
(for hardcopy distribution only)

1. Replace List of Effective Pages (LOEP) for Tables and Figures with the 2014 LOEP Update.
2. Replace entire text portions for each chapter with the updated text portion (including the Table of Contents, List of Figures, and List of Tables).
3. Update Tables and Figures according to the instructions below.

   NOTE: Tables and Figures from prior year were re-issued in order to remove revision bars.

	Remove	Insert
Chapter 3	Table 3-2 (5 pages)	Table 3-2 (6 pages)
	Table 3-5	Table 3-5
	Table 3-68 (8 pages)	Table 3-68 (8 pages)
Chapter 4	Table 4-22	Table 4-22
	Table 4-24	Table 4-24
Chapter 5	Table 5-5 (2 pages)	Table 5-5 (2 pages)
Chapter 6	Table 6-7 (7 pages)	Table 6-7 (7 pages)
	Table 6-22	Table 6-22
	Table 6-23	Table 6-23
	Table 6-24	Table 6-24
	Table 6-25	Table 6-25
	Table 6-26 (2 pages)	Table 6-26 (2 pages)
	Table 6-31(2 pages)	Table 6-31 (2 pages)
	Figure 6-36	Figure 6-36
	Figure 6-37	Figure 6-37
Chapter 7	Table 7-4	Table 7-4
	Figure 7-1 (16 pages)	Figure 7-1 (16 pages)
	Figure 7-5 (8 pages)	Figure 7-5 (8 pages)
	Figure 7-8	Figure 7-8
	Figure 7-10	Figure 7-10
Chapter 8	Table 8-3 (1 page)	Table 8-3 (2 pages)
	Table 8-4	Table 8-4
	Figure 8-1 (11 x 17)	Figure 8-1 (11 x 17)
	Figure 8-3 (11x17) (pg 1 of 2)	Figure 8-3 (11 x 17) (pg 1 of 2)
	Figure 8-4 (11 x 17) (3 pages)	Figure 8-4 (11 x 17) (3 pages)
	Figure 8-5 (11 x 17)	Figure 8-5 (11 x 17)

	Remove	Insert
Chapter 9	Figure 9-12	Figure 9-12
	Figure 9-17	Figure 9-17
	Figure 9-40	Figure 9-40
	new	Figure 9-44
	new	Figure 9-45
	new	Figure 9-46
Chapter 10	Figure 10-8	Figure 10-8
Chapter 13	Figure 13-1	Figure 13-1
	Figure 13-3	Figure 13-3
	Figure 13-4	Figure 13-4
	Figure 13-8	Figure 13-8
Chapter 15	Table 15-2	Table 15-2
	Table 15-32 (2 pages)	Table 15-32 (2 pages)
	Table 15-33	Table 15-33
	Table 15-34 (8 pages)	Table 15-34 (8 pages)
	Table 15-36	Table 15-36
	Table 15-46	Table 15-46
	Table 15-56	Table 15-56 (Deleted)
	Table 15-57	Table 15-57 (Deleted)
	Table 15-60	Table 15-60 (Deleted)
	Table 15-62 (2 pages)	Table 15-62 (2 pages)
	Table 15-63 (2 pages)	Table 15-63 (2 pages)
	new	Table 15-67
	new	Table 15-68 (2 pages)
	Figure 15-1	Figure 15-1
	Figure 15-2	Figure 15-2
	Figure 15-3	Figure 15-3
	Figure 15-4	Figure 15-4
	Figure 15-5	Figure 15-5
	Figure 15-6	Figure 15-6
	Figure 15-24	Figure 15-24
	Figure 15-29	Figure 15-29
	Figure 15-30	Figure 15-30
	Figure 15-31	Figure 15-31
	Figure 15-32	Figure 15-32
	Figure 15-33	Figure 15-33
	Figure 15-34	Figure 15-34
	Figure 15-35	Figure 15-35
	Figure 15-36	Figure 15-36

	Remove	Insert
Chapter 15 (continued)	Figure 15-112	Figure 15-112 (Deleted)
	Figure 15-114	Figure 15-114
	Figure 15-123	Figure 15-123
	Figure 15-129	Figure 15-129
	Figure 15-144	Figure 15-144
	Figure 15-145	Figure 15-145
	Figure 15-146	Figure 15-146
	Figure 15-147	Figure 15-147
	Figure 15-148	Figure 15-148
	Figure 15-149	Figure 15-149
	Figure 15-167	Figure 15-167
	Figure 15-174	Figure 15-174 (Deleted)
	Figure 15-178 thru 15-196 (19 pages)	Figure 15-178 thru 15-196 (Deleted) (1 page)
	Figure 15-197 thru 15-212 (16 pages)	Figure 15-197 thru 15-212 (Deleted) (1 page)
Chapter 18	Table 18-1 (4 pages)	Table 18-1 (4 pages)

Attachment 4
List of Removed Items
(2 pages)

List of Removed Items

1. Section 7.4.3.1.2, System Design - Turbine Driven EFW Pump (TDEFWP)

   The fifth paragraph was a single sentence which stated, "Once automatically started, the TDEFWP will continue to operate until manually secured by the operator or shutdown by the MSLB circuitry." Since this sentence would be corrected similarly to the sentence revised in the fourth paragraph (described below), the fifth paragraph was removed.

   > Section 7.4.3.1.2, fourth paragraph under subheading "Turbine Driven EFW Pump (TDEFWP)", was revised as follows since AFIS had been installed on all three units: "Once automatically started, the TDEFWP will continue to operate until manually secured by the operator or disabled by an AFIS ~~(Unit 1) or MSLB (Units 2 and 3)~~ signal."

   This change is based on the NRC-approved AFIS installation for all three units.

2. Section 9.2.3, Auxiliary Service Water System, was deleted and now refers users to UFSAR Section 9.7, Protected Service Water System. The Auxiliary Service Water System was replaced by the Protected Service Water System as detailed in the Safety Evaluation Report dated August 13, 2014.

3. Following the transition of all Oconee units to full-core Mark-B-HTP loading patterns, the existing UFSAR Chapter 15 description and results of the LBLOCA analyses and full-power SBLOCA analyses for Mk-B11 fuel and Mark-B-HTP fuel in a mixed-core configuration were removed as obsolete information.

   a. Section 15.14.4.1.2, Limiting Linear Heat Rate Analysis (LOCA Limit) for Mk-B11 (M5) Fuel, was deleted since it describes LBLOCA analyses applicable to Mark-B11 fuel. Since all Oconee units have completed the transition to full-core Mark-B-HTP loading patterns, the existing UFSAR Chapter 15 description and results of the LBLOCA analyses for Mark-B11 fuel are no longer applicable. Corresponding to the deletion of Section 15.14.4.1.2, the following UFSAR Figures and Table(s) were also deleted: Figures 15-112, 15-174, 15-178, 15-179, 15-180, 15-181, 15-182, 15-183, 15-184, and 15-185; Table 15-56.

   b. Section 15.14.4.1.3, Mixed Core Mark-B-HTP LOCA LHR Limits, was deleted since it describes LBLOCA analyses applicable to Mark-B11 and Mark-B-HTP fuel in a mixed-core configuration. Since all Oconee units have completed the transition to full-core Mark-B-HTP loading patterns, the existing UFSAR Chapter 15 description and results of the LBLOCA analyses for mixed-core configurations of Mark-B11 and Mark-B-HTP fuel are no longer applicable. Corresponding to the deletion of Section 15.14.4.1.3, the following UFSAR Figures and Table(s) were also deleted: Figures 15-198, 15-199, 15-200, 15-201, 15-202, 15-203, 15-204, 15-205, and 15-206; Table 15-60.

   c. Section 15.14.4.1.4, Mixed Core Mark-B11 LOCA LHR Limits, was deleted since it describes LBLOCA analyses applicable to Mark-B11 and Mark-B-HTP fuel in a mixed-core configuration. Since all Oconee units have completed the transition to full-core Mark-B-HTP loading patterns, the existing UFSAR Chapter 15 description and results of the LBLOCA analyses for mixed-core configurations of Mark-B11 and Mark-B-HTP fuel are no longer applicable.

d. Section 15.14.4.2.1, Mark-B-11 SBLOCA and Break Spectrum Results, was deleted since it describes SBLOCA analyses applicable to Mark-B11 fuel. Since all Oconee units have completed the transition to full-core Mark-B-HTP loading patterns, the existing UFSAR Chapter 15 description and results of the SBLOCA analyses for full cores of Mark-B11 fuel are no longer applicable. Corresponding to the deletion of Section 15.14.4.2.1, the following UFSAR Figures and Table should also be deleted: Figures 15-186, 15-187, 15-188, 15-189, 15-190, 15-191, 15-192, 15-193, 15-194, 15-195, 15-196, and 15-197; Table 15-57.

e. Section 15.14.4.2.2, Mixed Core Mark-B-HTP SBLOCA and Break Spectrum Results, was deleted since it describes SBLOCA analyses applicable to Mark-B11 and Mark-B- HTP fuel in a mixed-core configuration. Since all Oconee units have completed the transition to full core Mark-B-HTP loading patterns, a separate UFSAR section for description and results of the SBLOCA analyses for mixed-core configurations of Mark-B11 and Mark-B-HTP fuel is no longer applicable. Corresponding to the deletion of Section 15.14.4.2.2, the following UFSAR Figures should also be deleted: Figures 15-207, 15-208, 15-209, 15-210, 15-211, and 15-212.

4. Sections 18.3.19.1, Master Integrated Reactor Vessel Surveillance Program, 18.3.19.2, Cavity Dosimetry Program, 18.3.19.3, Fluence and Uncertainty Calculations, and 18.3.19.4, Pressure Temperature Limit Curves, were combined into a single section, 18.3.19, Reactor Vessel Integrity Program. The change updates the nomenclature and better defines the different components of the Program. The time limited aging analysis (TLAA) and surveillance and monitoring activities are defined with clear separation. Finally, the need to periodically exchange cavity dosimetry, update fluence transport calculations and predict end of life fluence is emphasized.

   Section 18.3.19.5, Monitoring Effective Full Power Years, was deleted. The implication that effective full power years (EFPY) are used to determine when updates are needed to TLAAs instead of fluence values is being corrected by deleting this section. A review of the Safety Evaluation Report (NUREG-1723) determined that EFPY was not part of the NRC's understanding of the need for TLAA fluence updates, so the correction is supported by the wording in the SER. The changes do not affect License Renewal commitments or change any of the regulatory requirements.

Attachment 5
10 CFR 50.59 Report
(18 pages)

Title: EC 100744 (Non-LAR Scope Only) – Making the Reverse Osmosis System Operable (Reverse Osmosis to Clean Silica from the SFPs and BWSTs) (10 CFR 50.59 Revision 0) (AR 00437862 / 01337862)[(1)]

Summary:
Engineering Change (EC) 100744 addresses making a Reverse Osmosis (RO) System operational that is to be used for treating the contents of the Units 1 & 2 Spent Fuel Pool (SFP), the Unit 3 SFP, the Unit 1 Borated Water Storage Tank (BWST), the Unit 2 BWST, and the Unit 3 BWST by removing silica. This EC is only addressing making the flowpath functional for processing the SFPs and the BWSTs. The RO Unit, piping, and electrical supplies to the RO System components were or will be installed by other ECs. The scope of two other ECs that install air operated valves in the BWST flowpath are also credited for RO System operation on the BWST. The RO Unit has been installed by another EC, but it was installed as an inert object, with no justification for its being operational. EC 100744 will allow the RO skid to be powered, and allow the opening of isolation valves associated with the flowpath for cleaning silica from the SFPs and BWSTs. These collective ECs have not made the RO Unit operational because it was determined that NRC approval was required for the use of the RO System. NRC approval has now been received.

The RO System will utilize a Boric Acid Recovery System (BARS) that is supplied by Diversified Technology Services (DTS). The BARS unit is designed to remove silica, while recovering boric acid to the maximum extent possible. This unit is used to remove silica by the reverse osmosis process, and so is typically called the Reverse Osmosis Unit (RO Unit). The reverse osmosis process uses high pressure applied to a solution on one side of a semi-permeable membrane. Some minerals, salts, and colloidal solids are unable to pass through the membrane and are rejected into a waste stream, while the remainder of the solution passes through the membrane and is collected for return to the system. The RO Unit that is to be used for the cleanup of the BWST and SFP has been designed to recover a high percentage of boron.

The elevated concentrations of silica in the SFPs and BWSTs are due to the decomposition of Boraflex, which was used in the spent fuel racks in the SFPs. This decomposition has caused the Reactor Coolant System silica concentration to be above that recommended by the fuel vendor. Reduction of the silica content would also allow the use of "zinc addition" in the Reactor Coolant System (RCS) for dose rate reduction. Zinc addition would be accomplished by another EC.

In the early stages of the Engineering Change process for EC 100744, Duke Energy determined that operating the RO System during Unit operation could not be implemented under the 10 CFR 50.59 process and would require Nuclear Regulatory Commission (NRC) approval before implementing the activity. As such, Duke Energy submitted a License Amendment Request (LAR) to the NRC to review and approve the design features and controls that would be used to ensure that operation of a Reverse Osmosis (RO) system during Unit operation does not significantly impact the BWST or SFP function or other plant equipment. The LAR also included new technical specification changes associated with the RO System. The LAR provided technical justification for periodic limited operation of the RO System during Unit operation. Duke Energy evaluated the effect of potential failures and identified precautionary measures that must be taken before and during RO System operation, and required operator actions to protect affected structures, systems, and components. The NRC has reviewed and approved the LAR.

[(1)] The AR (Activity Record) number is used to track the 10 CFR 50.59 report in Duke Energy's database.

There were aspects of the EC that were not included in the LAR or were changed from the information included in the LAR submittals. The aspect not included was a new automatic shutdown circuit that was added to address ambient temperature effects in Room 349. The new circuit was added by a separate EC, but the effects of the RO Unit increasing the room temperature and the use of the protection circuit as part of the operation of the RO Unit are addressed in the 10 CFR 50.59 evaluation for EC 100744. The aspect that was changed from the information contained in the LAR related to the tables submitted that contain time, boron, and water levels.

A summary of the justification for the responses to the evaluation questions is now provided.

The RO System itself is not required to operate to prevent any design basis events or accidents. The ambient temperature shutdown circuit could inadvertently shutdown the RO Unit, but since the RO System is not used to prevent events/accidents, it does not cause accidents previously evaluated in the UFSAR. So the ambient temperature shutdown circuit does not cause any accidents or events if it causes inadvertent loss of power to the RO Unit. The RO Unit will add heat to Room 349 while the unit is running and a smaller amount after it is shutdown and cools to ambient. The ambient temperature shutdown circuit is being added by another EC that will shutdown the RO Unit before Room 349 reaches temperatures that have been previously found to be acceptable for protection of equipment in the room that has a 10 CFR 50.59 design function. Thus, the activity does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

The new tables that provide requirements for operation of the RO System are supported by calculations that meet the criteria described in the LAR to the NRC. These criteria include that the maximum time period that the RO System can operate without makeup is limited by water inventory rather than boron concentration. In addition, the changes in water level can be easily recognized by plant personnel and low level is alarmed prior to decreasing below the TS low water level limit. Also, that in all cases the boron concentration stays above the TS minimum boron concentration prior to reaching the low water level alarm level or at the end of the 7 days, which will be specified as the maximum RO operating period. The LAR noted that the specific values used in the table that was submitted may be changed if the Oconee calculation was revised to support the change. These criteria were met by the new tables.

The ambient temperature shutdown circuit is designed to ensure that the heat from the RO Unit does not cause the room temperature at any time to go above the previously established room temperature limits. The design of the ambient temperature shutdown circuit is QA-1 and is seismically qualified, which includes all subcomponents that make up the ambient temperature circuits. Non-QA power to the monitoring circuit is used. This is acceptable since the same power source is used for both the RO Unit and the ambient temperature shutdown circuit. The power supply to the ambient temperature circuit comes off the supply to the RO Unit and the circuitry is otherwise redundant so that a single failure could not cause the circuit to be inoperable, since the circuit is redundant/single failure proof. If power is lost to the ambient temperature shutdown circuit, there is no credible way for power to the RO Unit to continue uninterrupted. An additional feature of the circuit is that power is required to keep the main contacts of the contactor supplying power to the RO Unit closed. If power is lost, stored mechanical energy (spring) is used to open the circuit contacts/breaker. This offers additional assurance that the RO Unit cannot continue to receive power for this power interruption event. This last feature is not specifically required because it must trip on high temperature regardless of this feature. Thus, no credible single failure will prevent the circuit from shutting off power to

the RO Unit when the setpoint is reached. The circuitry and equipment of the ambient temperature shutdown equipment is designed to withstand the environmental conditions that would be present up to the time it is called upon to act. The components used for the ambient temperature circuit are not digital and are not tied to components that are digital related. Thus, the activity will not result in a more than minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The RO System itself is not required to operate to mitigate any design basis events or accidents. The ambient temperature shutdown circuit could inadvertently shutdown the RO Unit, but the RO System is not used to mitigate events/accidents and will not increase any consequences of any accidents evaluated in the UFSAR. The new tables are supported by calculations that meet the criteria described in the LAR to the NRC. The ambient temperature shutdown circuit is designed to ensure that the heat from the RO Unit does not cause the room temperature at any time to go above the previously established room temperature limits. The changes to the tables and the addition of the ambient temperature shutdown circuit do not prevent any systems or components from performing their design functions as described in the UFSAR or the NRC's safety evaluation for the operation of the RO System during any accidents previously evaluated in the UFSAR. Thus, the activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

The RO System itself is not required to operate to mitigate any design basis events or accidents, so any malfunctions of the RO System operation itself does not affect dose. The ambient temperature shutdown circuit could inadvertently shutdown the RO Unit, but since the RO System is not used to mitigate events/accidents, it does not cause accidents previously evaluated in the UFSAR. The new tables are supported by calculations that meet the criteria described in the LAR to the NRC. The ambient temperature shutdown circuit is designed to ensure that the heat from the RO Unit does not cause the room temperature at any time to go above the previously established room temperature limits. The changes to the tables and the addition of the ambient temperature shutdown circuit do not prevent any systems or components from performing their design functions as described in the UFSAR or the NRC's safety evaluation for the operation of the RO System. The ambient temperature shutdown circuit is designed such that a single failure will not prevent it from shutting off the RO Unit if the room temperature reaches its setpoint. Thus, the activity does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The new tables are supported by calculations that meet the criteria described in the LAR to the NRC. The ambient temperature shutdown circuit is designed to ensure that the heat from the RO Unit does not cause the room temperature at any time to go above the previously established room temperature limits. The changes to the tables and the addition of the ambient temperature shutdown circuit do not prevent any systems or components from performing their design functions as described in the UFSAR or the NRC's safety evaluation for the operation of the RO System. There are no new accidents that the non-LAR portion of the EC can create. Thus, the activity does not create a possibility for an accident of a different type than previously evaluated in the UFSAR.

The new tables are supported by calculations that meet the criteria described in the LAR to the NRC. The ambient temperature shutdown circuit is designed to ensure that the heat from the RO Unit does not cause the room temperature at any time to go above the previously established room temperature limits. The changes to the tables and the addition of the ambient

temperature shutdown circuit do not prevent any systems or components from performing their design functions as described in the UFSAR or the NRC's safety evaluation for the operation of the RO System. The components used for the ambient temperature circuit are not digital and are not tied to components that are digital related. Thus, this activity does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

The fission product barriers are the fuel pellet, cladding, reactor coolant pressure boundary, and containment. This activity does not modify a fission product barrier, nor does this activity affect a controlling numerical value for a parameter established during the licensing review as presented in the UFSAR for a parameter used to determine the integrity of a fission product barrier. Thus, the modification does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

Specific calculation methods of evaluation regarding the calculation used to determine the values used in the table that contains the boron and level information are not described in the LAR, NRC's safety evaluation, or the UFSAR. This activity does not revise a computer code or calculation that is described in the UFSAR. It also does not change a design or analysis code described in the UFSAR. Thus, the activity does not involve a change in a method of evaluation described in the UFSAR. Thus, the modification does not result in a departure from a method of evaluation in the UFSAR.

The non-LAR scope did not require any changes, additions, or deletions to Technical Specifications (TS).

The responses to the 10 CFR 50.59 questions were all no. There were also no changes, additions, or deletions needed for the TSs. Thus, the activity can be implemented without prior NRC approval.

Title: Revise UFSAR Section 8.3.1.2 to Address 126 Day Keowee Reservoir Water Storage Supply for a Keowee Hydro Unit – Reference PIP O-08-3902 (10 CFR 50.59 Revision 0) (AR 00438727 / 01338727)

Summary:
UFSAR Section 8.3.2.1 contains a paragraph as follows:

The independent Keowee units, along with the alternate circuits, provide the required redundancy to assure reliable emergency power. Storage capacity of the Keowee reservoir and naturally occurring minimum streamflow are such that the generating units can provide continuous emergency power following an accident. The Keowee reservoir, between its normal elevation and maximum planned drawdown, has sufficient storage which, when combined with minimum recorded streamflow on the Keowee River will permit a hydro unit to carry continuously one nuclear unit's emergency auxiliary loads for 126 days.

UFSAR Section 8.3.1.2 is to be revised to remove the statement that provides information that the Keowee reservoir would permit a hydro unit to carry continuously one nuclear unit's emergency auxiliary loads for 126 days. The preceding statement to the one that is being removed is being reworded to remove the word "continuous" to avoid confusion in that continuous can imply going on forever and to remove the word "minimum" since the statement

is now more general. The removed sentence in the paragraph indicated that the Keowee Hydro Units (KHU) were not expected to operate forever.

The new paragraph is revised as follows:

The independent Keowee units, along with the alternate circuits, provide the required redundancy to assure reliable emergency power. The Keowee reservoir and its water supplies (e.g., incoming streamflow) provide the water for the Keowee units so they can provide emergency power following an accident.

The proposed UFSAR change is to remove a statement about the specified number of days that the Keowee reservoir is available for one KHU to carry continuously one nuclear unit's emergency auxiliary loads. Based on research, it was concluded that this information was provided by Duke originally to the Atomic Energy Commission (AEC) as part of a response to a Preliminary Safety Analysis Report (PSAR) question regarding the reliability of the emergency power supplies for Oconee. The Oconee design basis accident is for three units to have a Loss of Offsite Power, not just one. The second sentence in the paragraph is being reworded to remove the word "continuous" to avoid confusion in that continuous can imply going on forever and to remove the word "minimum" since the statement is now more general. The third sentence in the paragraph indicated that the KHUs were not expected to operate forever. The second sentence is also being reworded to clarify the concept that the Keowee reservoir and its water supplies are what provide the water for the KHUs. The third sentence is being deleted since the information is not part of the Keowee design function as discussed below.

The KHUs have a design function to provide emergency power to the three Oconee units. The following information addresses whether the 126 days of reservoir supply is part of the design function for Keowee supplying this emergency power function.

There was a question by the Atomic Energy Commission during the PSAR time period in which the AEC desired information regarding the reliability of the emergency power generation sources. Duke responded to this request with a number of aspects that was to show how reliable these sources were, including the KHUs. It was in this discussion that Duke addressed the Keowee reservoir could provide a storage capacity for a hydro unit carrying one nuclear unit's emergency auxiliary loads for 126 days. The early version of the FSAR also contained similar information to support statements made about the reliability of the KHUs as emergency power sources. The current version of the UFSAR uses this same type information in justification for the reliability of the hydro units as an emergency power source. The AEC's safety evaluations for the original FSARs did not address the lake reservoir from the perspective of supplying the KHUs for emergency power. The AEC review of the emergency power systems was oriented toward the design of the Keowee units themselves and the power distribution system, as well as the use of the Lee Steam Station gas turbines as backup power and the other two units.

The last sentence in the affected UFSAR paragraph is being removed since it is considered to be non-licensing basis information that was initially provided as general information to the AEC with respect to the response to a PSAR question. The statement in the UFSAR relating to the 126 day capacity of the reservoir does not correspond to the 3 unit site's design basis accident, which is a 2 unit LOOP and a 1 unit LOCA/LOOP. The UFSAR statement is the reservoir's water supply for only one nuclear unit's emergency auxiliary loads.

The UFSAR statement being changed also does not relate to a required volume of water in Lake Keowee. The statement refers to a volume of water based on normal elevation and maximum planned drawdown, but the normal lake level elevation is not a requirement. The maximum planned drawdown can be 775 feet based on a Technical Specification 3.8.1 level limit. No normal elevation foot level was specified in the statement. No requirement could be found in early licensing correspondence to maintain any level above 775 feet. At the time that the statement was initially in the UFSAR, there was no lake level requirement except for the TS LCO level limit. There is now a lake level limit in the Selected Licensee Commitments for lake water supply, but this commitment was not part of Oconee's licensing basis at the time that the 126 day value was initially included in the UFSAR.

Based on the subjectivity of the 126 day capacity statement in the UFSAR and the inconsistency with the Oconee design basis accident of 3 units having a LOOP, as well as finding no other pertinent information that utilizes the 126 day time period, the conclusion is made that this 126 day time period was not used in any accident analyses. The original Unit 1 and Units 2 & 3 safety evaluations also do not address the Keowee reservoir's capacity from the perspective of the water used for the KHU's emergency power supplies. The only requirement for lake level at the time of the initial UFSAR information about 126 days was the TS requirement for a lake level of 775 feet elevation, which was not to ensure a volume of lake water for the KHUs. This 126 day time period appears to have been just general information that was included in the original FSAR information.

Based on information in NEI 96-07 Revision 1 regarding what a design function consists of, the 126 days is not credited in safety analyses to meet NRC requirement and is not necessary to comply with regulations, license conditions, orders or technical specifications. In addition, the reservoir water supply with respect to supplying the KHUs is not a function that, if not performed, would initiate a transient or accident that the plant is required to withstand since the KHUs are not initiators of transients or accidents, but are used for mitigation of accidents. Based on the NEI 96-07 guidance and the discussion above, the KHUs' design function does not include the 126 day period included in the UFSAR.

Thus, the activity does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The KHUs would not be used in an emergency power function unless a loss of offsite power (LOOP) occurred, the changes to the UFSAR paragraph would not cause any accident previously evaluated in the UFSAR. Thus, the change does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

The KHU(s) themselves are not changed and will still be able to perform their design basis function of providing adequate emergency power to the Oconee units for a Loss of Coolant Accident (LOCA)/LOOP and other loss of offsite power events. The only change is to revise/remove the statements about a volume of water that may or may not be available in the Keowee reservoir to supply the KHUs. The required minimum lake level is controlled by Technical Specifications and Selected Licensee Commitments. The minimum water level is related to the volume available. If the water supply is not available, either with the existing or revised UFSAR information, the KHUs will not perform their function of providing emergency power supply to Oconee. But this effect will be the same if the water is not available. Therefore there is not a more than minimal increase in the consequences of an accident previously evaluated in the UFSAR.

The UFSAR change does not affect the single failure design of the Oconee emergency power system. The change is removing non-design function information about a volume of water that may or may not be available to the KHUs. The required minimum lake level is controlled by TSs and SLCs. The minimum water level is related to the volume available. The effect to Oconee would be the same if the KHUs were not available due to loss of the lake source. If no emergency power from a KHU is available, the consequences will be the same for any condition when emergency power is needed. Thus, the activity does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The UFSAR change does not cause any accidents of a different type. The lake provides water for cooling functions at Oconee, but this water is returned to the lake. The function of the lake and the KHUs are not changed. The revision/removal of information regarding a potentially available (since no required volumes are specified) volume of reservoir water does not cause the KHUs or the lake to operate in a manner different than currently evaluated in the UFSAR. The station blackout (SBO) event already assumes the loss of the KHUs as an emergency power source. Thus, the activity does not create a possibility for an accident of a different type than previously evaluated in the UFSAR.

Lake Keowee provides the source of water for the KHUs. If the reservoir water supply is lost, the effect of running out of water for the KHU emergency power supply will be the same. The KHUs emergency power supply will be lost. The change does not create any new potential for the loss of multiple emergency power sources. Thus, there are no new malfunctions that can be created with a different result. Thus, the activity does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

The fission product barriers are the fuel pellet, cladding, reactor coolant pressure boundary, and containment. This activity does not modify a fission product barrier, nor does this activity affect a controlling numerical value for a parameter established during the licensing review as presented in the UFSAR for a parameter used to determine the integrity of a fission product barrier. Thus, the activity does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

This activity does not revise a computer code or calculation that is described in the UFSAR. It also does not change a design or analysis code described in the UFSAR. No method of evaluation regarding the 126 days of availability was found to be described in the UFSAR. Thus, the activity does not involve a change in a method of evaluation described in the UFSAR. Thus, the UFSAR change does not result in a departure from a method of evaluation in the UFSAR.

Title: Oconee Unit 3 Cycle 28 Reload 10 CFR 50.59 Evaluation (AR 00443458 / 01343458)

Summary:

This activity installs the core designed for Oconee Nuclear Station Unit 3 Cycle 28 (O3C28). The O3C28 Reload Design Safety Analysis Review (REDSAR), performed in accordance with Engineering Directives Manual EDM-501, "Engineering Change Program for Nuclear Fuel", and the O3C28 Reload Safety Evaluation confirms the UFSAR accident analyses remain bounding with respect to predicted O3C28 safety analysis physics parameters (SAPP), and fuel thermal

and mechanical performance limits. The SAPP method is described in topical report DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology".

The O3C28 core reload is similar to past cycle core designs, with a design generated using NRC approved methods. Applicable Technical Specifications have been reviewed and no changes are required for the operation of O3C28. This 10 CFR 50.59 evaluation concluded that no prior NRC approval is necessary for O3C28 operation.

Title: EC 91875, Rev. 75 (OD500927) – Keowee AC Power Supply Tie-ins to PSW, EC 91873, Rev. 16 (OD500922) - PSW Power Feed Installation, 10 CFR 50.59 Evaluation Rev. 1 (AR 00444590 / 01344590)

Summary:
This 10CFR50.59 Evaluation addresses engineering change packages EC 91875, Rev. 75 and EC 91873, Rev. 16, which have been developed to provide an alternate QA-1 power supply to the PSW system. EC 91875 installs, at Keowee, the following:

- PSW switchgear cabinets (KPF-1 and KPF-2).
- Incoming PSW switchgear breakers KPF-9 and KPF-10.
- Outgoing PSW switchgear breakers KPF-11 and KPF-12.
- New tap connections to the existing KHU segregated buses.
- Power cable between the new tap connections on the KHU segregated buses and breakers KPF-9 and KPF-10.
- Cable tray and cable tray supports.
- Control switches and instrumentation in the Keowee control room.
- New protective relaying and modifications to existing protective relaying.
- Control power from the existing Keowee 125VDC Power System.

EC 91873 installs the following:

- New and existing (spare) underground power cable from outgoing PSW switchgear breakers KPF-11 and KPF-12 to PSW switchgear B6T and B7T in the PSW Building. The cable will be routed in cable trenches and duct banks.
- Three new PSW MCCs, 1XPSW, 2XPSW, and 3XPSW, in the Unit 1/2/3 Auxiliary Building. The MCCs will be powered from a load center in the PSW Building via underground power cables. (NOTE: EC 91873, Rev. 16 only includes a partial turnover of new PSW MCCs 1XPSW, 2XPSW, and 3XPSW, which permits the MCCs to be energized and to power prescribed loads.
- Underground power cable from the PSW Building load center (PX13) to the Auxiliary Building MCCs 1XPSW, 2XPSW, and 3XPSW, via underground duct bank.
- New instrument cable from new MCCs 1XPSW, 2XPSW, and 3XPSW to existing terminal cabinets.

AR 444477 is a single 10CFR50.59 Screening that was prepared for the above ECs. The screening concluded that certain aspects of these ECs resulted in adverse effects to design functions of the KHUs as described in the UFSAR, and in controlling the electrical distribution system in a manner that is outside the reference bounds of its design, as described in the UFSAR. Therefore, the scope of this 10CFR50.59 Evaluation will address these specific aspects of EC 91875 and EC 91873.

10CFR50.59 Screening AR 444477 determined that EC 91875 and EC 91873 resulted in the following adverse effects to design functions as described in the UFSAR (Screening Question 1):

a) The existing differential protective relay for each KHU (3Ø 87GB-1 and 3Ø 87GB-2) is being replaced with three (3) single phase differential relays for each KHU (87GB1-ZØ, -YØ, -XØ and 87GB2-ZØ, -YØ, -XØ) for differential protection of each Keowee electric generator. The existing differential protective relays provide an alarming function, while the replacement differential protective relays will provide an emergency lockout (trip) of the respective KHU in the event of a phase-to-phase fault in either the emergency electric power system or in the new PSW electric distribution system. Installation of the replacement relays is considered to have an adverse effect on the design function of the KHUs because a failure of any one of the new 87GB relays may initiate a trip of the respective KHU and prevent the KHU from performing its credited design function.

b) A current transformer, a 50B overcurrent relay and a 62-timer relay are being added upstream of PSW electrical distribution system feeder breakers KPF-9 and KPF-10 to provide breaker fault detection and protection for these circuits. If breakers KPF-9 or KPF-10 are tripped and subsequently fail to open, the breaker failure scheme becomes active. If the 50B overcurrent relay detects continued fault current for the time setting of the 62-timer relay, the 62-timer relay will initiate an emergency lockout (trip) of the respective KHU. Installation of the 62-timer relay is considered to have an adverse effect on the design function of the KHUs because a failure of the relay may initiate a trip of the respective KHU and prevent the KHU from performing its credited design function.

c) The existing KHU generator buses, which are segregated and metal-enclosed, are being modified to provide a means of routing new power cable from the KHU buses to new PSW feeder breakers KPF-9 and KPF-10. A new bolted connection will be made to each of the generator buses. Flexible electrical connections will be installed on each of the segregated phase buses and connected to newly installed transition junction boxes. Cable terminations will be made inside the transition junction boxes and the cables will be routed to breakers KPF-9 and KPF-10. These new components (bolted connections, flexible electrical connectors, transition junction boxes, cabling and termination points) create new potential fault locations for the KHU generator buses. A fault on any of the new components may result in a lockout (trip) of the associated KHU and prevent the KHU from performing its credited design function.

10CFR50.59 Screening AR 444477 also determined that EC 91875 and EC 91873 involves an SSC that will be controlled in a manner that is outside the reference bounds of its design as described in the UFSAR (Screening Question 4):

The existing Keowee frequency and voltage out-of-tolerance (OOT) protection logic ensures that KHU generator frequency and voltage are within analyzed limits before loads are placed onto the KHU buses. This protection is accomplished either by blocking the closure or by tripping existing emergency power system breakers. Since this protection is performed automatically, no operator action is required to verify adequate frequency and voltage during either the initial automatic addition of emergency loads onto the KHU buses or during the subsequent manual addition of non-emergency loads (supplied from the main feeder buses) onto the KHU buses. However, new PSW breakers KPF-9 and KPF-10 have not been provided with automatic frequency and voltage OOT protection, and therefore will require that Operations personnel verify that the voltage and frequency of a running KHU is acceptable before placing PSW system loads onto the KHU buses. The PSW system loads will be the only loads at Oconee that can be added onto the KHU buses without automatic frequency and voltage OOT protection.

Normal operation and failures of the new 87GB relays and the new 62-timer relays, and faults within the new components installed to route power from the existing KHUs to the new PSW feeder breakers KPF-9 and KPF-10, have been evaluated and shown to result in an increase in the probability of a Station Blackout accident that is less than 10%. Therefore, installation of the new 87GB relays, the new 62-timer relays, and the new components installed to route power from the existing KHUs to the new PSW feeder breakers KPF-9 and KPF-10 does not result in more than a minimal increase in the frequency of occurrence of an accident.

The manual addition of PSW system electrical loads to a KHU generator will not result in an overload condition where the combination of the current design basis electrical loads and the PSW system electrical loads exceed the rated capacity of the Keowee generators. It will also not result in a KHU generator loading transient that causes a frequency out-of-tolerance (OOT) trip or a voltage OOT trip of emergency power path breakers. Manual verification of proper KHU frequency and voltage prior adding PSW system loads will also not increase the probability of a KHU failure. Therefore, the manual addition of PSW system electrical loads onto a KHU generator will not initiate a Station Blackout and does not result in any increase in the frequency of occurrence of an accident.

The new protective relaying components, the components that make up the new power path between the KHU generators and the new PSW switchgear, and the potential Keowee generator trips that they may introduce, have been evaluated and determined to be within the existing single failure design criteria of the KHU generator and the emergency electric power system. Also, these components will be furnished as QA-1 (as required), will be seismically qualified, and will be seismically installed. Hence, EC 91875 and EC 91873 will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety, and will have no increase in the consequences of a either an accident or a malfunction of an SSC important to safety.

The new protective relaying components and the components that make up the new power path between the KHU generators and the new PSW switchgear will not introduce any equipment failures or malfunctions that are new or that have a different result than what is considered credible for the existing protective relaying and power path components in these systems. Therefore, the new protective relaying and power path components will not create a new type of accident or a malfunction with a different result.

The manual verification of proper KHU frequency and voltage prior to the addition of PSW system electrical loads onto the Keowee generator buses, and the addition of the PSW system

electrical loads themselves have been evaluated and shown not to prevent the KHU generators from performing their credited design functions. Hence, the manual addition of PSW system electrical loads onto the Keowee generator buses will not increase either the probability or consequences of accidents or malfunctions, and will not create a new type of accident or a malfunction with a different result.

Accident analyses assume that electrical power is provided to the essential equipment that is required to protect fuel cladding, the reactor coolant pressure boundary, and the containment. For accidents which include the consideration of the loss of offsite power (LOCA and main steam line break), the KHUs are credited for providing the required electrical power to these essential systems within 23 seconds. The new 87GB relays, the new 62-timer relays, the new components installed to route power from the existing KHU generator buses to new breakers KPF 9 and KPF-10, and the manual loading of the PSW system electrical loads onto the KHU generators will have no impact on the ability of at least one of the two redundant KHUs to automatically start and align to the emergency buses within 23 seconds. Therefore, the new relays, new power path components, and new manual loading of the KHU generators do not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

A revision to Technical Specifications will not be required.

Title: EC 91834, REV. 29, (OD100950) – U1 HPI ALTERNATE POWER FEED FROM PSW, 10 CFR 50.59 EVALUATION, REV. 0 (AR 00445172 / 01345172)

Summary:
This 10 CFR 50.59 Evaluation addresses engineering change package EC 91834, Rev. 29, which has been developed to provide an alternate power source from the PSW electrical distribution system to selected components in the Unit 1 High Pressure Injection (HPI) system and the Unit 1 Reactor Coolant System (RCS) to support core reactivity control, reactor coolant inventory control, and reactor coolant pressure control functions following events (fire and HELB) that damage the essential 4160VAC power system on the affected unit. The components that will be furnished with an alternate power supply from the PSW electrical distribution system are:

- <u>1A and 1B HPI Pumps</u> – Two motor-operated power transfer switches (1HPISXTRN001 and 1HPISXTRN002) will be installed to allow remote switching of the power source for the 1A and 1B HPI pumps. Power will be available from either the pumps normal power supply or from the PSW electrical distribution system. The motor operated transfer switches will be operated via new control switches that were installed in the main control room (MCR) by EC 91830. Since only one switchgear in the PSW electrical distribution system has been provided to support Unit 1 HPI pump operation, a separate manual alignment switch (1HPISXALGN001) will be installed for the purpose of selecting which HPI pump (1A or 1B) will be powered and operated from the designated PSW switchgear.
- <u>1A HPI BWST Suction Valve (1HP-24)</u> – A new power transfer switch (1HPISXTRN003) will be installed to allow remote switching of the source of motive and control power for valve 1HP-24. Power will be available from either the valves normal power supply or

from the PSW electrical distribution system. The power transfer switch will be operated by a new control switch that was installed in the MCR by EC 91830.

- 1A HPI to RCS Loop A Injection Valve (1HP-26) – A new power transfer switch (1HPISXTRN004) will be installed to allow remote switching of the source of motive and control power for valve 1HP-26. Power will be available from either the valves normal power supply or from the PSW electrical distribution system. The power transfer switch will be operated by a new control switch that was installed in the MCR by EC 91830.
- RCS Loop High Point Vent Valves (1RC-155, -156, -157 and -158) and Reactor Vessel Head Vent Valves (1RC-159 and -160) – New control switches installed within the control circuits of valves 1RC-155, -156, -157, -158, -159 and 160 by EC 91830 will be wired into the PSW electrical distribution system. These control switches will permit the switching of the source of motive and control power for valves 1RC-155, -156, -157, -158, -159 and 160. Power will be available from either the valves normal power supply or from the PSW electrical distribution system.

In addition, EC 91834 will replace existing manually operated RCP Seal Flow Isolation Valve (1HP-139) and RCP Seal Flow Control Bypass Valve (1HP-140) with electric motor operated (EMO) valves and will power these valves from the PSW electrical distribution system. These new EMO valves will provide the ability to remotely control flow to the RCP seals after a postulated failure of the existing RCP Seal Flow Control Valve (1HP-31). Finally, EC 91834 will add a PSW piping tie-in to the existing low pressure service water (LPSW) system line to provide the capability to provide cooling water to the HPI pump motor coolers from the PSW system.

AR 444959 is the 10 CFR 50.59 Screening that was prepared for EC 91834, Rev. 29. The screening concluded that only one aspect of EC 91834 resulted in an adverse effect to the design function of an SSC, as described in the UFSAR. Therefore, the scope of this 10 CFR 50.59 Evaluation will address the components installed by EC 91834 that created this adverse effect.

10 CFR 50.59 Screening AR 444959 concluded that the installation of new power transfer switch 1HPISXTRN003 will have an adverse effect on the design function of valve 1HP-24. A motor contactor within power transfer switch 1HPISXTRN003 must be energized and its contacts must be in the "closed" position for the existing normal power supply to be available to valve 1HP-24. Failure of the contactor will result in the normal power supply to valve 1HP-24 being unavailable. Without its normal power supply, valve 1HP-24 will not be able to actuate in response to an engineered safeguards actuation signal.

A failure of new power transfer switch 1HPISXTRN003 will render the normal power supply to valve 1HP-24 unavailable and thereby prevent valve 1HP-24 from operating when required. However, neither the operation of valve 1HP-24 nor its failure to operate will initiate any of the accidents or transients that have been evaluated in the UFSAR. Therefore, the installation of power transfer switch 1HPISXTRN003 will not result in an increase in the frequency of an accident previously evaluated in the UFSAR.

Failures of new power transfer switch 1HPISXTRN003 have been evaluated and determined to be within the existing single failure design criteria of the HPI system. Also, the new power transfer switch will be furnished as QA-1, will be seismically qualified, and will be seismically installed. Hence, the new power transfer switch will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety, and will have no

increase in the consequences of either an accident or a malfunction of an SSC important to safety.

New power transfer switch 1HPISXTRN003 will not introduce any equipment failures or malfunctions that are new or that have a different result than what is considered credible for the existing HPI system. Therefore, the new power transfer switch will not create a new type of accident or a malfunction with a different result.

The MSLB, REA, small-break LOCA and SGTR accident analyses credit the availability and performance of the HPI system to protect fuel cladding, the reactor coolant pressure boundary, and the containment. Valve 1HP-24 is specifically credited for being one of two redundant flow paths between the BWST and the HPI pumps and for being sufficiently open within 14 seconds of receiving an ES signal to allow the required flow from the HPI system. New power transfer switch 1HPISXTRN003 will have no impact on the ability of at least one of the two redundant flow paths between the BWST and the HPI pumps (via valve 1HP-24 or valve 1HP-25) to automatically open and be sufficiently open within 14 seconds of receiving an ES signal. Therefore, the installation of new power transfer switch 1HPISXTRN003 does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

A revision to the Technical Specifications will not be required.

Title: EC 91857, REV. 19 (OD200953) UNIT 2 HPI ALTERNATE POWER (OUTAGE), EC 91858, REV. 7 (OD200954) UNIT 2 HPI ALTERNATE POWER (PRE-OUTAGE), 10 CFR 50.59 EVALUATION, REV. 0 (AR 00445198 / 01345198)

Summary:
This 10 CFR 50.59 Evaluation addresses engineering change packages EC 91857, Rev. 19, and EC 91858, Rev. 7, which have been developed to provide an alternate power source from the PSW electrical distribution system to selected components in the Unit 2 High Pressure Injection (HPI) system and the Unit 2 Reactor Coolant System (RCS) to support core reactivity control, reactor coolant inventory control, and reactor coolant pressure control functions following events (fire and HELB) that damage the essential 4160VAC power system on the affected unit. The components that will be furnished with an alternate power supply from the PSW electrical distribution system are:

- <u>2A and 2B HPI Pumps</u> – Two motor-operated power transfer switches (2HPISXTRN001 and 2HPISXTRN002) will be installed to allow remote switching of the power source for the 2A and 2B HPI pumps. Power will be available from either the pumps normal power supply or from the PSW electrical distribution system. The motor operated transfer switches will be operated via new control switches that were installed in the main control room (MCR) by EC 91852. Since only one switchgear in the PSW electrical distribution system has been provided to support Unit 2 HPI pump operation, a separate manual alignment switch (2HPISXALGN001) will be installed for the purpose of selecting which HPI pump (2A or 2B) will be powered and operated from the designated PSW switchgear.
- <u>2A HPI BWST Suction Valve (2HP-24)</u> – A new power transfer switch (2HPISXTRN003) will be installed to allow remote switching of the source of motive and control power for valve 2HP-24. Power will be available from either the valves normal

power supply or from the PSW electrical distribution system. The power transfer switch will be operated by a new control switch that was installed in the MCR by EC 91852.

- 2A HPI to RCS Loop A Injection Valve (2HP-26) – A new power transfer switch (2HPISXTRN004) will be installed to allow remote switching of the source of motive and control power for valve 2HP-26. Power will be available from either the valves normal power supply or from the PSW electrical distribution system. The power transfer switch will be operated by a new control switch that was installed in the MCR by EC 91852.
- RCS Loop High Point Vent Va lves (2RC-155, -156, -157 and -158) and Reactor Vessel Head Vent Valves (2RC-159 and -160) – New control switches installed within the control circuits of valves 2RC-155, -156, -157, -158, -159 and 160 by EC 91852 will be wired into the PSW electrical distribution system. These control switches will permit the switching of the source of motive and control power for valves 2RC-155, -156, -157, -158, -159 and 160. Power will be available from either the valves normal power supply or from the PSW electrical distribution system.

In addition, EC 91857 and EC 91858 will replace existing manually operated RCP Seal Flow Isolation Valve (2HP-139) and RCP Seal Flow Control Bypass Valve (2HP-140) with electric motor operated (EMO) valves and will power these valves from the PSW electrical distribution system. These new EMO valves will provide the ability to remotely control flow to the RCP seals after a postulated failure of the existing RCP Seal Flow Control Valve (2HP-31). Finally, EC 91857 and EC 91858 will add a PSW piping tie-in to the existing Low Pressure Service Water (LPSW) system line to provide the capability to provide cooling water to the HPI pump motor coolers from the PSW system.

AR 445081 is the 10 CFR 50.59 screen that was prepared for EC 91857, Rev. 19, and EC 91858, Rev. 7. The screen concluded that two aspects of EC 91857 and EC 91858 resulted in adverse effects to the design function of an SSC, as described in the UFSAR. Therefore, the scope of this 10 CFR 50.59 Evaluation will address the components installed by EC 91857 and EC 91858 the created these adverse effects.

10 CFR 50.59 Screen AR 445081 concluded that the installation of new power transfer switch 2HPISXTRN003 will have an adverse effect on the design function of valve 2HP-24. A motor contactor within power transfer switch 2HPISXTRN003 must be energized and its contacts must be in the "closed" position for the existing normal power supply to be available to valve 2HP-24. Therefore, for the existing normal power supply to be available to valve 2HP-24, a motor contactor within power transfer switch 2HPISXTRN003 must be energized and must change from the "open" to the "closed" position. Failure of the contactor will result in the normal power supply to valve 2HP-24 being unavailable. Without its normal power supply, valve 2HP-24 will not be able to actuate in response to an engineered safeguards actuation signal.

10 CFR 50.59 Screen AR 445081 concluded that the installation of new power transfer switch 2HPISXTRN004 will have an adverse effect on the design function of valve 2HP-26. A motor contactor within power transfer switch 2HPISXTRN004 must be energized and its contacts must be in the "closed" position for the existing normal power supply to be available to valve 2HP-26. Therefore, for the existing normal power supply to be available to valve 2HP-26, a motor contactor within power transfer switch 2HPISXTRN004 must be energized and must change from the "open" to the "closed" position. Failure of the contactor will result in the normal power supply to valve 2HP-26 being unavailable. Without its normal power supply, valve 2HP-26 will not be able to actuate in response to an engineered safeguards actuation signal.

A failure of new power transfer switch 2HPISXTRN004 will render the normal power supply to valve 2HP-26 unavailable and thereby prevent valve 2HP-26 from operating when required. However, neither the operation of valve 2HP-26 nor its failure to operate will initiate any of the accidents or transients that have been evaluated in the UFSAR. Therefore, the installation of power transfer switch 2HPISXTRN004 will not result in an increase in the frequency of an accident previously evaluated in the UFSAR.

Failures of new power transfer switch 2HPISXTRN004 have been evaluated and determined to be within the existing single failure design criteria of the HPI system. Also, the new power transfer switch will be furnished as QA-1, will be seismically qualified, and will be seismically installed. Hence, the new power transfer switch will not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety, and will have no increase in the consequences of either an accident or a malfunction of an SSC important to safety.

New power transfer switch 2HPISXTRN004 will not introduce any equipment failures or malfunctions that are new or that have a different result than what is considered credible for the existing HPI system. Therefore, new power transfer switch 2HPISXTRN004 will not create a new type of accident or a malfunction with a different result.

The MSLB, REA, small-break LOCA and SGTR accident analyses credit the availability and performance HPI system to protect fuel cladding, the reactor coolant pressure boundary, and the containment. Valve 2HP-24 is specifically credited for being one of two redundant flow paths between the BWST and the HPI pumps and for being sufficiently open within 14 seconds of receiving an ES signal to allow the required flow from the HPI system. New power transfer switch 2HPISXTRN003 will have no impact on the ability of at least one of the two redundant flow paths between the BWST and the HPI pumps (via valve 2HP-24 or valve 2HP-25) to automatically open and be sufficiently open within 14 seconds of receiving an ES signal. Therefore, the installation of new power transfer switch 2HPISXTRN003 does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

The MSLB, REA, small-break LOCA and SGTR accident analyses credit the availability and performance HPI system to protect fuel cladding, the reactor coolant pressure boundary, and the containment. Valve 2HP-26 is specifically credited for aligning the HPI discharge header to the reactor vessel and for being sufficiently open within 14 seconds of receiving an ES signal to allow the required flow from the HPI system. New power transfer switch 2HPISXTRN004 will have no impact on the ability of the HPI discharge path to automatically open and be sufficiently open within 14 seconds of receiving an ES signal. Therefore, the installation of new power transfer switch 2HPISXTRN004 does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

A revision to the Technical Specifications will not be required.

Title: EC 109523, Revision 004 – U1&2 BWST RECIRC ISOLATION VALVES, 10 CFR 50.59 Revision 0 (AR 00445282 / 01345282)

Summary:
The Nuclear Regulatory Commission (NRC) issued amendments for the Reverse Osmosis (RO) system by correspondence dated April 30, 2014. The connection from the BWSTs to the RO System includes redundant automatically actuated, safety-related, seismically qualified isolation valves between the RO System supply piping and the Borated Water Storage Tanks (BWSTs). These automatically actuated isolation valves also isolate the BWST recirculation pump from the BWST. The isolation valves actuate to close on declining BWST level before BWST TS level is reached, thereby isolating RO and Spent Fuel Pool Cooling (SFPC) purification systems from BWST prior to entering Reactor Building Emergency Sump (RBES) recirculation phase following drawdown of the BWST. Isolation of RO and SFPC purification systems prevents unanalyzed consequences from leakage from BWST into those systems' piping when in the RBES recirculation phase. The redundant control circuitry of the isolation valves contain digital pressure transmitters which actuate on low BWST level to close the valves. Actuation of any one of 4 digital pressure transmitters will close both redundant valves and secure the BWST recirculation pump. Loss of power to the circuit, or loss of air to the valves will result in the valves closing.

The digital pressure transmitter used in this application has self-diagnostics and is configured to fail LO which closes the redundant valves. The digital pressure transmitter is demonstrated to be a very reliable device and has received a generic qualification from EPRI for use in mild environment nuclear applications.

The evaluation demonstrates that the proposed control circuitry for the redundant isolation valves will preserve the current licensing basis. The activity will not create more than a minimal increase in the frequency or consequences of accidents or malfunctions of SSCs important to safety. The proposed activity will not create the potential for a new type of unanalyzed event, has no impact on the fission product barriers and does not affect evaluation methodology. The answers to all eight evaluation questions is negative, no Tech Spec modification, deletion, or addition is required and no UFSAR revision is required. Therefore under 10 CFR 50.59 it is permissible to implement this modification without prior approval from the NRC.

Title: EC 91877, REV. 30, PSW SYSTEM HEADER; EC 91878, REV. 1, REPLACE ASW SYSTEM WITH PSW SYSTEM; EC 106526, REV. 0, INSTALL MOTOR FEEDER CABLES FOR PSW PUMPS; EC 111881, REV. 12, PSW PUMP ROOM DUCT EXTENSION; 10 CFR 50.59 EVALUATION REV. 0 (AR 00445554 / 01345554)

Summary:
EC 91877, EC 91878, EC 106526 and EC 111881 will replace the existing ASW system with the PSW system. The new PSW system will provide an increase in mechanical capacity, as compared to the ASW system, such that it will be capable of injecting raw water into the secondary side of the steam generators without first depressurizing the steam generators. The new PSW system will also be furnished with the necessary power-operated valves, instrumentation, and controls to permit both initiation and operation of the system from the main control room.

Replacement of the ASW system with the PSW system will require NRC approval based on the following:

- Duration of Decay Heat Removal - After a loss of Lake Keowee event, the stored water inventory in the CCW intake and discharge piping is credited for providing 37 days of decay heat removal capability via the existing ASW system. However, the PSW system will only be credited for providing decay heat removal for a minimum of 30 days, regardless of whether it is placed into operation to mitigate a loss of Lake Keowee event or a tornado event. In addition, the stored water inventory in the CCW intake and discharge piping has only been demonstrated adequate for 30 days of PSW system operation. As such, the replacement of the ASW system with the PSW system results in an increase in the probability of failure of the system that is credited for providing 37 days of decay heat removal after a loss of Lake Koewee event. Therefore, the new PSW system results in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.
- Use of Alternate Criteria for Postulating High Energy Line Breaks (HELBs) - The current ONS criteria for postulating HELBs would require that HELBs be considered in the new PSW system based on its normal operating pressure. However, alternate criteria, which is based on the actual time that the system is in operation, has been applied to the PSW system. Using this alternate criteria, HELBs in the PSW system are not required to be postulated. The application of this alternate criteria constitutes a departure from a method of evaluation described in the UFSAR used in establishing the design basis.

Replacement of the ASW system with the PSW system will not require a revision to the Technical Specifications.

Title: EC 0000110806 - 005 – UNIT 1 & 2 600/208VAC CABLE TRAY SYSTEM FOR PSW SCENARIO –10 CFR 50.59 Evaluation of Selected Unit 1 & 2 Auxiliary Building Penetrations) (AR 00445630 / 01345630)

Summary:

The installation of four (4) penetrations in the walls, floors, and ceilings of the Unit 1 Auxiliary Building were determined to affect from radiation the qualification of equipment located proximate to the penetration installation areas. Using the guidance of NEI 96-07, the appropriate review of such adverse effects is documented by performing the 10 CFR 50.59 Evaluation summarized here. Engineering Change (EC) 110806 provided the evaluations to update the Oconee Equipment Qualification Criteria Manual (EQCM) to demonstrate that with these four (4) penetrations installed, plant equipment proximate to the penetration locations continued to be capable of reliably performing their important to safety functions, and the penetration installations do not create initiators of accidents previously evaluated in the UFSAR. Consequences for accidents previously described in the UFSAR are not increased, nor will these penetration installations increase accident dose consequences through a malfunction of an SSC important to safety previously evaluated in the UFSAR. These activities do not create the possibility for an accident of a different type than any previously evaluated in the UFSAR, nor do they create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR. No design basis for a fission product barrier

as described in the UFSAR is being exceeded or altered, and these activities do not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

Title: Revise SLC 16.9.16, Reactor Building Polar Crane and Auxiliary (Control Rod Drive) Hoist – (RCS System Open) (AR 00445799 / 01345799)

Summary:
This proposed activity is a change to Selected Licensee Commitment 16.9.16, Reactor Building Polar Crane and Auxiliary (Control Rod Drive) Hoist – (RCS System Open). The change is a revision to the Bases of the SLC. This change is needed to allow more flexibility in the use of the polar crane hoist and auxiliary hoist over the fuel transfer canal.

The Bases are to be revised as follows:

Current Version:

Use of either reactor building polar crane hoist and auxiliary (CRD) hoist over the fuel transfer canal when the reactor vessel head is removed is restricted to those operations necessary for the fuel handling and core internals operations.

Proposed Revised Bases:

Use of either reactor building polar crane hoist and auxiliary (CRD) hoist over the fuel transfer canal when the reactor vessel head is removed is restricted to those operations necessary for the fuel handling and core internals operations *<u>or other specific limited activities that have been analyzed appropriately to demonstrate that the activity is incapable of resulting in a radiological release in the event of a load drop onto the reactor core.</u>*

By not allowing activities which could cause a radiological release, this SLC change is bounded by the Fuel Handling accident already evaluated in the UFSAR.