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ONS-2013-015

November 14, 2013

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Station, Units 1, 2, and 3
Docket Nos. 50-269,-270,-287
Emergency Plan, Volume A, Revision 2013-01

Please find attached for your use and review copies of the revision to the Oconee Nuclear Station Emergency Plan.

This revision is being submitted in accordance with 10 CFR 50.54(q) and does not reduce the effectiveness of the Emergency Plan or the Emergency Plan Implementing Procedures. By copy of this letter, two copies of this revision are being provided to the NRC, Region II, Atlanta, Georgia.

If there are any questions or concerns pertaining to this submittal please call
Pat Street, Emergency Planning Manager at 864-873-3124.

Sincerely,

Scott L. Batson

Scott L. Batson
Vice President
Oconee Nuclear Station

Attachments:
Revision Instructions
Emergency Plan, Volume A, Revision 2013-01
50.54(q) Evaluation(s)

AX45
NRR

xc: (w/2 copies of attachments)

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Oconee Nuclear Station

OCONEE NUCLEAR STATION
EMERGENCY PLAN

APPROVED:

Scott L. Batson

Scott L. Batson
VP, Oconee Nuclear Station

November 15, 2013
Date Approved

November 15, 2013
Effective Date

VOLUME A
REVISION 2013-01
NOVEMBER 2013

November 14, 2013

OCONEE NUCLEAR STATION

SUBJECT: Oconee Nuclear Site Emergency Plan
Volume A, Revision 2013-01

Please make changes to the Emergency Plan, Volume A by following the below directions.

Each Section listed has been revised; replace entire Sections.

Cover Sheet

List of Effective Pages

List of Figures

Record of Changes

Appendix 4 Evacuation Time Estimates

Section D

Section I

Section J

Section P

Pat Street
Emergency Planning Manager
Oconee Nuclear Station

Oconee Emergency Plan
List Of Effective Pages

SECTION	PAGE NUMBER	REVISION NO.	DATE
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Coversheet		Rev. 2013-01	October 2013
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	Page 1	Rev. 2012-05	December 2012
I. Introduction			
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A. Assignment of Responsibility			
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B. Onsite Emergency Organization			
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	Page C-1 & C-2	Rev. 2012-05	December 2012
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	Page D-1 - D-100	Rev. 2013-01	October 2013
E. Notification			
	Page E-1 & E-2	Rev. 2008-02	December 2008
F. Emergency Communications			
	Page F-1 - F-8	Rev. 2012-05	December 2012
G. Public Information and Education			
	Page G-1 - G-5	Rev. 2012-05	December 2012
	2013 Emergency Planning Calendar	Rev. 2012-05	December 2012
H. Emergency Facilities And Equipment			
	Page H-1 - H-39	Rev. 2012-02	June 2012
I. Accident Assessment			
	Page I-1 - I-37	Rev. 2013-01	October 2013

Oconee Emergency Plan
List Of Effective Pages

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APPENDIX 5
Letters of Agreement

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Agreement Letter 1	Rev. 2012-05	May 17, 2011
Agreement Letter 2	Rev. 2012-05	April 15, 2011
Agreement Letter 3	Rev. 2012-05	April 18, 2011
Agreement Letter 4	Rev. 2012-05	April 18, 2011
Agreement Letter 5	Rev. 2012-05	April 14, 2011
Agreement Letter 6	Rev. 2010-01	February 23, 2010
Agreement Letter 7	Rev. 2012-05	October 21, 2011
Agreement Letter 8	Rev. 2012-05	April 30, 2012
Agreement Letter 9	Rev. 2012-05	April 14, 2011
Agreement Letter 10	Rev. 2012-05	April 25, 2011
Agreement Letter 11	Rev. 2012-05	April 15, 2011
Agreement Letter 12	Rev. 2012-05	December 13, 2011
Agreement Letter 13	Rev. 2012-05	September 21, 2009
Agreement Letter 14	Rev. 2012-05	July 9, 2012
Agreement Letter 15	Rev. 2012-05	July 6, 2012
Agreement Letter 16	Rev. 2012-05	September 30, 2011
Agreement Letter 17 DELETED	Rev. 2004-01	DELETED - 08/26/03
Agreement Letter 18	Rev. 2012-05	September 05, 2012

	Agreement Letter 19	Rev. 2012-05	March 22, 2011
	Agreement Letter 20	Rev. 2010-01	January 05, 2010
	Agreement Letter 21	Rev. 2012-05	June 7, 2011
	Agreement Letter 22	Rev. 2012-05	May 29, 2012
	Agreement Letter 23	Rev. 2012-05	April 24, 2012
	Agreement Letter 24	Rev. 2012-05	December 24, 2011
	Agreement Letter 25	Rev. 2012-03	May 23, 2012
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APPENDIX 6 **Distribution List**			
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Spill Prevention Control And Countermeasure Plan	Page 1 - 40	Rev. 2012-05	October 2009
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	Page 1 - 12	Rev. 2010-01	February 2010

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RECORD OF CHANGES

REVISION NUMBER	EFFECTIVE DATE	REASON FOR REVISIONS
Revision 1	April 1, 1981	Meteorological Update
Revision 2	December 31, 1981	Rewrite Emergency Plan in Nureg 0654 Format
Revision 3	March, 1982	Update Emergency Plan
Revision 4	April, 1982	Revisions & Changes to update Emergency Plan
Revision 5	September 1, 1982	Revision to coincide with Crisis Management Plan
Revision 6	November 1, 1982	Revision update
Revision 7	December 14, 1982	Review and update
83-1	June 10, 1983	Changes required by action items due to annual exercise and review and general update
83-2	November 17, 1983	Changes required by review and general update
84-1	March 26, 1984	Revisions as determined by QA audit and minor editing
84-2	November 15, 1984	Revisions as determined by annual review
85-1	June 7, 1985	Revisions/changes/editing
85-2	-------------	Revisions/changes/editing-annual review
86-1	March 8, 1986	New Oconee Brochure
86-2	November 13, 1986	Revisions/changes/editing-annual review
86-3	December 9, 1986	Correct changes identified as deficiencies by the NRC in Rev. 85-2.
87-1	February 4, 1987	Revision update, minor editing changes, included failed fuel accident assessment information.
87-2	-------------	Revision update, minor editing changes Review Section D. Agreement letters updated.
87-4	December 10, 1987	Incorporate alternate TSC and OSC into Emergency Plan
88-1	June 7, 1988	Revised EALS in Section D.
88-2	October 14, 1988	Annual review. Minor editorial revisions.
89-1	February 28, 1989	Major revision to Section D. Added Appendix 7. Minor editorial changes.
89-2	August 14, 1989	Change to Section D. Minor editorial revisions.
89-3	January 5, 1990	Annual Review.

RECORD OF CHANGES (Continued)

REVISION NUMBER	EFFECTIVE DATE	REASON FOR REVISIONS
90-1	March 1, 1990	Changes to Section D as required by NRC commitment.
90-2	June 1, 1990	Changes reflect upgrade of radiation monitor system and minor editing.
90-3	July 2, 1990	Change to Section D, Emergency Classification.
90-4	October 31, 1990	Annual Review
91-1	January 21, 1991	Section D revision. (RIA upgrade)
91-2	February 20, 1991	Section D revision. (TS to SLC)
91-3	March 22, 1991	Section D revision. (RIA upgrade); Section D revision. (SLB revision)
91-5	September 19, 1991	Section D revision. (RIA upgrade)
91-6	December 16, 1991	Annual review.
92-1	March 1, 1992	Section D (RIA upgrade). Minor editorial changes.
92-2	June 30, 1992	Major Revision
92-3	October 29, 1992	Annual review
92-4	12/31/92	Section B, D, H, J, Appendix 4, 5 & 6 changes.
93-1	03/01/93	Sections D, G, H, N, P, and Appendix 6
93-2	05/07/93	Sections A, B, D, Appendix 5 and 6
93-3	07/23/93	Sections A, B, G, H, I, J, L, M, N, & Appendix 6
93-4	08/11/93	Sections B, D, and Appendix 5
93-05	01/01/94	Annual Review, Incorporation of EPA-400 guidelines.
94-01	03/15/94	Additions of Appendix 8 and 9. - (Minor revisions)
94-02	05/09/94	Changes to Appendix 5, Pages 1 and 2; Changes to Appendix 6, Pages 2 and 4; State of South Carolina Agreement Letter
94-03	05/25/94	Changes to Appendix 5, Page 2; Changes to Appendix 6, Pages 4 and 5; INPO Agreement Letter
94-04	06/06/94	Changes to Appendix 5, Page 2; Change Teledyne Isotopes Badge Service agreement letter to Northeast Utilities Service Company
94-05	08/08/94	Changes to Section D
94-06	12/29/94	Annual review. Editorial changes, minor revisions.

RECORD OF CHANGES (Continued)

REVISION NUMBER	EFFECTIVE DATE	REASON FOR REVISIONS
95-01	02/23/95	Changes to Sections B, G, Appendix 5.
95-02	10/23/95	Annual review and changes
95-03	11/01/95	Section D. Change, Incorporated new EAL'S.
95-04	12/31/95	Calendar 1996, HAZMAT Changes, RP/14 deleted
96-01	02/13/96	Changes to Sections B, D, and N.
96-02	06/25/96	Changes to Section D
96-03	07/96	Changes to Section D
96-04	12/96	Annual review, editorial changes, minor changes with major change to Appendix 10.
97-01	07-97	Section B, I, Appendix 5 & 7, with editorial/minor changes to Section H & P
97-02	12-97	Annual review and editorial/minor changes
98-01	02-98	Section D, page 35. Correction of title on Enclosure 4.3
98-02	03-98	Section N, page 1 & 2, Added part a (General) to Section N.2 to ensure drills conducted between NRC evaluated exercises are performed in accordance with 10CFR50, Appendix E, Section IV.F.2.b
98-03	04-98	List of Figures page number corrections, Added Emergency Operation Facility to Figure H-15, Figure H-20 reformatted. Added Agreement Letter with Keowee-Key Volunteer Fire Department, Appendix 5, #24. Appendix 10 - Hazardous Materials Response Plan, corrections on Table of Contents with minor revisions. Headings on Appendix 10, Figure 2 with minor revisions.
98-04	12-98	Annual review and editorial/minor changes.
99-01	03-99	The ONS Technical Specifications have been converted to a set of Technical Specifications based on NUREG 1430. "Standard Technical Specifications Babcock and Wilcox Plants." Replaced the description phrases (titles) in Section D for Operating Modes with the Mode number from Improved Technical Specifications. In Section I the portion describing leak rate volume percent per day was changed to percent of the containment air weight per day. The reference to Tech Spec 4.4.1.1 was changed to reference Improved Technical Specification 5.5.2. NOTE: The implementation date of Improved Tech Specs was moved from March 4, 1999 to March 27, 1999, therefore the revision date for revision 99-01 will depict February when the actual administrative changes were completed.

RECORD OF CHANGES (Continued)

REVISION NUMBER	EFFECTIVE DATE	REASON FOR REVISIONS
99-02	12-99	Annual review and editorial/minor changes
2000-01	04-2000	Addition of List of Effective Pages
2000-02	05/2000	Editorial /minor changes
2000-03	12/2000	Annual review and editorial/minor changes
2001-01	02/07/2001	Additions and corrections as result of 50.54(t) audit. Additional information added to Basis Document and additional EAL's resulting from EP drill critiques.
2001-02	08/2001	Changes in areas of responsibility. Added note concerning RVLS to Fission Product Barrier Matrix; 2001 calendar; information added to EP Functional Area Manual; added/updated information on annual average meteorology; Appendix 5; Appendix 6; editorial/minor changes.
2001-03	12/2001	Added information in Basis Document concerning a reactor building containment break. Replaced the 2001 calendar with the 2002 calendar. Editorial/minor changes.
2002-01	01/02	The present Oconee Nuclear Station Emergency Operating Procedure is written in a different format and with some different terms than the earlier version. The term PTS (Pressurized Thermal Shock) has replaced TSOR (Thermal Shock Operating Range). This is only a change in terminology. The additional EAL is to ensure a site specific credible threat results in a declaration of a notification of Unusual Event (NOUE). This change is also intended to achieve an appropriate level and consistent response Nationwide.
2002-02	06/02	Section B - minor changes; Section D - Added information requested by Emergency Coordinators to Enclosure 4.1; Section G - Rewrite of entire section; Section H - Updated information on Figure H-4 relating to Met Data; Appendix 5 - Updated Letters of Agreement; and miscellaneous spelling/grammar errors.
2002-03	09/02	Section A - Compliance with the NRC Security Interim Compensatory Measure (ICM) issued 02/25/02; Section P - Audit frequencies per revised 10 CFR 50.54 (t) as stated in Federal Register Vol 64, 03/29/99. Appendix 1 - Added definition of monthly and Semi-Annual; Appendix 5, Agreement Letters, updated #17, Appendix 6 - Changed name on 78A. Miscellaneous corrections.
2003-01	02/03	Section D - RIA setpoints change, Section G - 2003 Calendar, Appendix 3 - Siren upgrade, new map (i-5) ; Appendix 5 - Agreement Letters, Appendix 6 - Issued To change, Section B, E, F editorial/minor changes
2003-02	08/03	Section D - incorporates additional guidance for the Emergency Coordinator/EOF Director related to classification of a high energy line break, such as a Main Steam Line Break. In addition, Section D has been retyped using a consistent font style - no changes in content resulted from the retype.

RECORD OF CHANGES (Continued)

REVISION NUMBER	EFFECTIVE DATE	REASON FOR REVISIONS
2004-01	02/04	Incorporates a retype of the majority of the sections as an editorial change to adopt a consistent format: Section G - Added information concerning One Mile Exclusion Area Signs; Section H - Strip Chart Recorders were removed under an NSM; Section J - Incorporated guidance on the use of KI as a protective action recommendation; Section K - changed KI dose to 5 REM CDE from 25 REM; Appendix 4 - Incorporate results of Evacuation Time Estimate; Appendix 5 - Revised Agreement Letters
2004-02	12/21/04	Editorial changes to correct typos, drawings, and title/organizational names. This revision also incorporates clarifying information from the latest Evacuation Time Estimate (ETE); clarification of offsite agency responsibilities for protective actions for impediments and special populations; revised EAL #2 for Enclosure 4.3, Unusual Event IC #2; clarification of ERO activation after normal working hours; and revisions to the site's SPCC Plan included in Appendix 8. In addition to these changes, applicable references have replaced generic references in Figure P-1. This revision also incorporates the 2005 Calendar distributed to the 10 mile EPZ population.
2005-01	02/01/05	Section D, Enclosure 4.7, Page 66 - Duke Power Hydro-Electric Group has revised the Lake Keowee water level from 807 to 815.5 feet for initiating a Condition B. This elevation is used in Enclosure 4.7 for classifying the event as an Unusual Event. The Hydro -Electric Group notifies the Control Room when Condition B has been declared. No protective actions by the plant are changed.
2005-02	05/17/05	Section I & Letters of Agreement - Incorporates an editorial revision that describes the makeup of Field Monitoring Teams and updated Agreement Letters. I.7&8 replaced "....personnel from Radiation Protection and Chemistry." with "..a RP Technician and a Driver." Editorial Change - Chemistry personnel no longer perform the function of FMT Driver. FMT Drivers are now provided by other groups.
2005-03	08/24/05	Revision 2005-03 incorporates an addendum for the Fire Department/Volunteer Fire Department Agreement Letters. This addendum was added as a result of NRC guidance provided to utilities. The addendum to these letters provides guidance on the use of the Incident Command System at ONS and identifies the ONS Fire Brigade Leader as the on-scene commander and site-interface for responding offsite fire departments.
2005-04	09/15/05	Revision 2005-04 is a change to Page 66, Enclosure 4.7, Emergency Action Levels #1 - Reservoir elevation greater than or equal to 807.0 feet with all spillway gates open and the lake elevation continues to rise. This change undoes Revision 2005-01 which changed Keowee Lake level from 807 feet elevation to 815.5 feet elevation. This revision was determined to be a non conservative change in that it delayed the Unusual Event emergency classification. Appendix 5, Agreement Letter #21 has been updated.
2005-05	01/09/06	Revision 2005-05 incorporates editorial changes that clarify organizational charts/responsibilities, revise procedure references, replaces public information calendar, and replaces obsolete survey instruments. Agreement Letters #16 and #19 were updated.

RECORD OF CHANGES (Continued)

REVISION NUMBER	EFFECTIVE DATE	REASON FOR REVISIONS
2006-03	06/8/06	Section D - Change #1 Revised initiating condition #2 for the Alert classification for Enclosure 4.6 (Fire/Explosions and Security Events). This change is based on a correction to the NEI White Paper, Enhancements to Emergency Preparedness Programs For Hostile Actions which was endorsed in a letter from the NRC on December 8, 2005. Change #2 - Renumbered Emergency Action Levels through out Section D to match the numbering scheme found in RP/0/B/1000/001 (Emergency Classification) procedure - Renumbering makes it easier for procedure users to locate the correct emergency action level in the Basis Document. Appendix 5 - Agreement Letters #8, 14,15 & 23 were updated.
2006-04	11/06	Reference changes to the deletion of the Clemson EOF and incorporates reference to the Charlotte EOF. In addition, miscellaneous editorial changes are included in this revision.
2007-01	03/07	Appendix 5 Agreement Letters that have been updated/revised.
2007-02	12/07	Editorial changes including a revised 50 mile radius map (Figure B), a revision to the Emergency Classification Basis Section D , the 2008 Emergency Planning Calendar, a revised layout drawing for the JIC, a revised listing of portable survey instruments, the latest renewal of existing agreement letters and a revised Ground Water Monitoring Plan
2008-01	09/08	The original order of the EALs created a human performance trap. The first fission barrier column that the procedure user reviews is the RCS Barrier column which is on the left side of the page. The second fission barrier column that is reviewed is the Fuel Clad Barrier which is in the center of the page. This order gives the procedure user the mind set that the EALs are listed in the same order: RCS EAL followed by the Fuel Clad EAL. Changing the order of the EALs is not a deviation from the approved EAL scheme but is a difference. This change does not constitute a decrease in the effectiveness of the EPLAN since the EALs are exactly the same.
2008-02	10/08	As of this change 2008-02, the Emergency Plan is now available on NEDL/SCRIBE and has been completely re-issued. All changes in the future to the Emergency Plan will be completed thru NEDL/SCRIBE. The following Agreement Letters were also updated: 1, 2, 3, 4, 5, 6, 7, 9, 10, 11, 19 and 21.
2009-01	02/09	Revised existing information relating to organization names that have changed, removed specific names and replaced with a title to mitigate the need for future revisions due to personnel changes, and changed staging location names based on changes made to area designation names; however staging will still occur in same area. Changes made only reflect actual organization names, functional position names, and current location names being used to make the E-Plan more accurately reflect current information. No changes are being made to the process or conduct of the how the E-Plan is to be implemented.

RECORD OF CHANGES (Continued)

REVISION NUMBER	EFFECTIVE DATE	REASON FOR REVISIONS
2010-10	02/10	Revised existing information relating to changes made to the callback system, who performs the dose assessments, the basis information for the Containment Barrier EAL based on NEI 99-01 Rev 5 FAQ lessons learned. Made name change for Oconee Medical Center, corrected information relating to testing frequency for major elements referenced in the E-Plan, the new neutron instrument used by radiation protection, and street name change for figure H-3A. Changes made are the result of the Annual Review process and no changes are being made to the process or conduct of how the E-Plan is to be implemented. The following Agreement Letters were also updated: Number - 6, 8, 13, 14, 15, 16, 18, 20, 22, & 23.
2011-01	05/11	Figure B-10 - Redistribution of support for Field Monitoring Teams from Chemistry to Business Management and Work Control. Section D - Basis corrected to delete reference to USFAR Table 15-114 which has been deleted, revised ICs 4.3.A.3 and 4.4.A.3, EAL A to align with RP/0/B/1000/001, revised ICs and EALs to add levels of operating modes that represent the operating levels of hot shutdown, cold shutdown and hot standby were listed, added "AC" back to IC 4.5.A.1 where it had been inadvertently deleted, add SSF to IC 4.6.U.1, correct IC 4.5.G.1, EAL 1 to reflect SSF maintaining Mode 3 (hot standby) rather than hot shutdown, add new ICs for Jocassee Dam condition A and B declarations, correct misprint in IC 4.7.A.2, EAL B, correct formatting errors, and add Security EALs. Section F - deleted onsite areas requiring phone notifications for site assembly due to new wireless system being installed in those areas. Section G - replace 2010 calendar with 2011 calendar. Figure H-1 - revised room layout to reflect current arrangement. Section N - Revised the testing cycle for the EPLAN from a 5 year cycle to a 6 year cycle. Appendix 5 - update letters of agreement.
2011-02	10/11	This evaluation supports a request to revise the Oconee (ONS), McGuire (MNS), and Catawba (CNS) Emergency Plans to allow for an alternate approach for compliance with 10 CFR 50.47(b)(2) relative to meeting the minimum staffing requirement during emergencies for site Radiation Protection (RP) personnel and the Emergency Operations Facility (EOF) position staffing to that in Table B-1 in NUREG-0654, endorsed by Regulatory Guide 1.101.
2012-01	6/12	Section F - A change to the process for answering the 4911 emergency phone calls. The new process will have both Operations and Security(SAS) answering the phone. Appendix 7 -Will clarify the ERDS related system description verbiage from the modem based data transfer system to the new VPN System.
2012-02	06/12	The NRC published Federal Register notice [RIN 3150-AI10], "Enhancements to Emergency Preparedness Regulations" on November 23, 2011. The amendments contained in the rule are summarized as twelve (XII) topics with varying implementation due dates. Emergency Plan changes to the following sections (C, D, H, I, J, P, and Appendix 1) are made in accordance with the rule and the appropriate guidance documents pertaining to Topic V – Emergency Action Level for Hostile Action, Topic VI – Emergency Declaration Timeliness, Topic VIII – Emergency Operation Facility (Performance Based), Topic IX – Emergency Response Organization Augmentation at Alternate Facility, and Topic XI – Protective Actions for On-site Personnel.

RECORD OF CHANGES (Continued)

REVISION NUMBER	EFFECTIVE DATE	REASON FOR REVISIONS
2012-03	06/13	Added Agreement Letter 25 - G&G Metal Fabrication to provide Hale pump technical support and Agreement Letter 26 Operating Agreement between Duke Energy's Lincoln Combustion Turbine Facility & MNS, CNS and ONS Nuclear Supply Chain concerning an Emergency Supply of Diesel Fuel.
2012-04	12/12	Section B - This change is to incorporate the new staffing analysis for the new EP rule and editorial changes.
2012-05	12/12	Revised Section D, Enclosure 4.3 to add threshold values for unit vent sampling as a compensatory measure. Unit vent sampling is performed on the 6th floor auxiliary building at sampling equipment where manual grab samples are retrieved per HP/0/B/1000/060-D. Additionally, the use of RIA 56 was added as a compensatory measure for Site Area Emergency and General Emergency Classifications. This change allows for classification of gaseous radiological releases in the event of a loss of either RIA-45 or 46. This change only clarifies the values to be used in the event normal monitoring is not available. The plan is also being revised based on annual review requirements, changes are mainly editorial or formatting. Additional changes are being made to reflect current name changes, update Agreement letters, Spill Prevention and Control, and Groundwater monitoring programs.
2013-01	10/13	Section D - Added clarification in the basis for Loss of Shutdown function. Section I - Revised to reference procedures versus RPSM 11.7 which has been deleted. Section J - Revised to incorporate latest revision to ETE. Deleted climate data tables which were duplicative to information contained within the ETE (Appendix 4). Section P - Updated appropriate references. Appendix 4 - Added latest ETE as reference.

Duke Energy Oconee Nuclear Station **EMERGENCY PLAN A - APPENDIX 4 EVACUATION TIME ESTIMATES**	Procedure No. **EPA APPENDIX 04**
	Revision No. **001**
	Electronic Reference No. **OAP000HF**

This Procedure was printed on 10/15/13 at 09:59:18 from the electronic library as:

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APPENDIX 4

DUKE ENERGY COMPANY
OCONEE NUCLEAR SITE

Evacuation Time Estimates

The Evacuation Time Estimates (ETEs) for the Oconee Nuclear Station, dated November 2012, KLD Engineering, P.C. Report TR-494, Oconee Nuclear Station, Development of Evacuation Time Estimates, Revision 1, November 2012 was submitted under separate cover and is considered to be incorporated as part of this document by reference.

See ONS-ETE-12142012, Rev. 000: ONS EVACUATION TIME ESTIMATES (ETE) DATED 12/14/2012.

D. EMERGENCY CLASSIFICATION SYSTEM

NUREG 1.101, Rev. 3, August, 1992, approved the guidance provided by NUMARC/NESP-007, Revision 2, as an Alternative Methodology for the Development of Emergency Action Levels. Oconee Nuclear Site used the NUMARC guidance for the development of initiating conditions and emergency action levels. The emergency action levels provided in this section have been modified to implement the guidance provided in NRC Bulletin 2005-02, NEI guidance as endorsed in Regulatory Issue Summary 2006-12 and to support the implementation of NEI 03-12.

The emergency classification system utilizes four categories for classification of emergency events.

D.1.a. UNUSUAL EVENT

Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

The purpose of an Unusual Event classification is to provide notification of the emergency to the station staff, State and Local Government representatives, and the NRC.

Specific initiating conditions and their corresponding emergency action levels are provided in the Basis Document beginning on page D-4.

D.1.b ALERT

Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

The purpose of the Alert classification is to assure that emergency personnel are readily available to:

1. Activate the onsite response centers

2. Respond if the situation becomes more serious or to perform confirmatory radiation monitoring if required

3. Provide offsite authorities current status information

Specific initiating conditions and their corresponding emergency action levels are provided in the Basis Document beginning on page D-4.

D.1.c. SITE AREA EMERGENCY

Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

The purpose of the Site Area Emergency classification is to:

1. Activate the offsite response centers

2. Assure that monitoring teams are mobilized

3. Assure that personnel required for taking protective actions of near site areas are at duty stations should the situation become more serious

4. Provide current information to the public and be available for consultation with offsite authorities

Specific initiating conditions and their corresponding emergency action levels are provided in the Basis Document beginning on page D-4.

D.1.d. GENERAL EMERGENCY

Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

The purpose of the General Emergency classification is to:

1. Initiate predetermined protective actions for the public

2. Provide continuous assessment of information from onsite and offsite measurements

3. Initiate additional measures as indicated by event releases or potential releases

4. Provide current information to the public and be available for consultation with offsite authorities

Specific initiating conditions and their corresponding emergency action levels are provided in the Basis Document beginning on page D-4.

D.2 Initiating Conditions

Initiating conditions and their corresponding emergency actions levels are contained in the BASIS document beginning on page D-4. Classification procedure (RP/0/B/1000/001) provides the guidance necessary to classify events and promptly declare the appropriate emergency condition within 15 minutes after the availability of indications to cognizant facility staff that an emergency action level threshold has been exceeded. Specific response procedures are in place for the Control Room, Technical Support Center and the Emergency Operations Facility which delineate the required response during the appropriate classification.

D.3 LOCAL AND STATE EMERGENCY ACTION LEVELS

Pickens County FNF Plans
Oconee County FNF Plans
State of South Carolina FNF Plans (Site Specific)

D.4 LOCAL AND STATE EMERGENCY PROCEDURES

Pickens County FNF Plans
Oconee County FNF Plans
State of South Carolina FNF Plans (Site Specific)

ENCLOSURE 4.1
FISSION PRODUCT BARRIER MATRIX

DETERMINE THE APPROPRIATE CLASSIFICATION USING THE TABLE BELOW: ADD POINTS TO CLASSIFY. SEE NOTE BELOW

RCS BARRIERS (BD 5-7)		FUEL CLAD BARRIERS (BD 8-9)		CONTAINMENT BARRIERS (BD 10-12)	
Potential Loss (4 Points)	**Loss (5 Points)**	**Potential Loss (4 Points)**	**Loss (5 Points)**	**Potential Loss (1 Point)**	**Loss (3 Points)**
RCS Leakrate ≥ 160 gpm	RCS Leak rate that results in a loss of subcooling.	Average of the 5 highest CETC ≥ 700° F	Average of the 5 highest CETC ≥ 1200° F	CETC ≥ 1200° F ≥ 15 minutes **OR** CETC ≥ 700° F ≥ 15 minutes with a *valid* RVLS reading 0″	Rapid unexplained containment pressure decrease after increase **OR** containment pressure or sump level not consistent with LOCA
SGTR ≥ 160 gpm		*Valid* RVLS reading of 0″ NOTE: RVLS is <u>NOT</u> valid if one or more RCPs are running <u>OR</u> if LPI pump(s) are running <u>AND</u> taking suction from the LPI drop line.	Coolant activity ≥ 300 µCi/ml DEI	RB pressure ≥ 59 psig **OR** RB pressure ≥ 10 psig and no RBCU or RBS	Failure of secondary side of SG results in a direct opening to the environment with SG Tube Leak ≥ 10 gpm in the **SAME** SG
Entry into the PTS (Pressurized Thermal Shock) Operation NOTE: PTS is entered under either of the following: • A cooldown below 400°F @ > 100°F/hr. has occurred. • HPI has operated in the injection mode while NO RCPs were operating.	1RIA 57 or 58 reading ≥ 1.0 R/hr 2 RIA 57 reading ≥ 1.6 R/hr 2 RIA 58 reading ≥ 1.0 R/hr 3RIA 57 or 58 reading ≥ 1.0 R/hr		**Hours Since SD / RIA 57 OR RIA 58 R/hr / R/hr** 0 - <0.5 / ≥ 300 / ≥ 150 0.5 - < 2.0 / ≥ 80 / ≥ 40 2.0 - 8.0 / ≥ 32 / ≥ 16	**Hours Since SD / RIA 57 OR RIA 58 R/hr / R/hr** 0 - < 0.5 / ≥ 1800 / ≥ 860 0.5 - < 2.0 / ≥ 400 / ≥ 195 2.0 - 8.0 / ≥ 280 / ≥ 130	SG Tube Leak ≥ 10 gpm exists in one SG. **<u>AND</u>** the other SG has secondary side failure that results in a direct opening to the environment <u>AND</u> is being fed from the affected unit.
HPI Forced Cooling	RCS pressure spike ≥ 2750 psig			Hydrogen concentration ≥ 9%	Containment isolation is incomplete and a release path to the environment exists
Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment	Emergency Coordinator/EOF Director judgment

UNUSUAL EVENT (1-3 Total Points)	ALERT (4-6 Total Points)	SITE AREA EMERGENCY (7-10 Total Points)	GENERAL EMERGENCY (11-13 Total Points)
<u>OPERATING MODE:</u> 1, 2, 3, 4	**<u>OPERATING MODE:</u>** 1, 2, 3, 4	**<u>OPERATING MODE:</u>** 1, 2, 3, 4	**<u>OPERATING MODE:</u>** 1, 2, 3, 4
4.1.U.1 Any potential loss of Containment 4.1.U.2 Any loss of containment	4.1.A.1 Any potential loss or loss of the RCS 4.1.A.2 Any potential loss or loss of the Fuel Clad	4.1.S.1 Loss of any two barriers 4.1.S.2 Loss of one barrier and potential loss of either RCS or Fuel Clad Barriers 4.1.S.3 Potential loss of both the RCS and Fuel Clad Barriers	4.1.G.1 Loss of any two barriers and potential loss of the third barrier 4.1.G.2 Loss of all three barriers

NOTE: An event with multiple events could occur which would result in the conclusion that exceeding the loss or potential loss threshold is <u>IMMINENT</u> (i.e., within 1-3 hours). In this IMMINENT LOSS situation, use judgment and classify as if the thresholds are exceeded.

ENCLOSURE 4.1

BASIS INFORMATION FOR FISSION PRODUCT BARRIER REFERENCE TABLE

RCS BARRIER EALs: (1 or 2 or 3 or 4 or 5)

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

1. RCS Leak Rate

Small leaks may result in the inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the High Pressure Injection System. The capacity of one HPI pump at normal system pressure is approximately 160 gpm. Leakage in excess of this value would call for compensatory action to maintain normal liquid inventory. As such, this is an indication of a degraded RCS barrier and is considered to be a potential loss of the barrier.

The loss of subcooling is the fundamental indication that the inventory loss from the primary system exceeds the capacity of the inventory control systems. If the loss of subcooling is indicated, the RCS barrier is considered lost.

2. SG Tube Rupture

Small Steam Generator tube leaks may result in the inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the High Pressure Injection System. The capacity of one HPI pump at normal system pressure is approximately 160 gpm. Leakage in excess of this value would call for compensatory action to maintain normal liquid inventory. As such, this is an indication of a degraded RCS barrier and is considered to be a potential loss of the barrier.

A tube rupture (> than 160 gpm) with an unisolable secondary line rupture is generally indicated by a reduction in primary coolant inventory, increased secondary radiation levels, and an uncontrolled or complete depressurization of the ruptured SG. This set of conditions represents a potential loss of the RCS and loss of containment fission product barrier and will result in the declaration of a Site Area Emergency. Escalation to a General Emergency would be indicated by at least a potential loss of the fuel clad barrier.

ENCLOSURE 4.1

2. SG Tube Rupture

Secondary radiation increases should be observed via radiation monitoring of Condenser Air Ejector Discharge, Main Steam, and/or SG Sampling System. Determination of the "uncontrolled" depressurization of the ruptured SG should be based on indication that the pressure decrease in the ruptured steam generator is not a function of operator action. This should prevent declaration based on a depressurization that results from an EOP induced cooldown of the RCS that does not involve the prolonged release of contaminated secondary coolant from the affected SG to the environment. This EAL should encompass steam breaks, feed breaks, and stuck open safety or relief valves.

A steam generator tube leak less than 160 gpm would be classified under Enclosure 4.2, Systems Malfunctions, RCS leakage as an Unusual Event. If a release also occurs such as steam through a steam relief valve failed open, feedwater line break, steam line break on the affected steam generator then a loss of the Containment Barrier has also occurred. Upgrade to a higher classification would be by Enclosure 4.3, Abnormal Rad Levels/Radiological Effluent or further degradation of RCS or Fuel Clad Barriers.

3. Entry Into PTS

Entry into Pressurized Thermal Shock Operation could cause damage to the reactor vessel severe enough to cause a loss of coolant accident. Therefore, this situation represents a potential loss of the RCS. This EAL is satisfied if Rule 8 (Pressurized Thermal Shock) is implemented.

4. Reactor Coolant System Integrity

HPI Forced cooling represents the failure of the steam generators to remove heat from the core. To use this mode of cooling indicates that all feedwater (both main and emergency) are not available for use and the pressure in the reactor coolant system is greater than or equal to 2300 psig. The power-operated relief valve must be opened to initiate the cooling through the high pressure injection system. In effect, a self-imposed loss of coolant is established. The condition is classified as a potential loss of the reactor coolant system.

A reactor coolant system pressure spike of greater than or equal to design pressure of 2750 psig represents a loss of the RCS barrier.

ENCLOSURE 4.1

5. Containment Radiation Monitoring

A containment radiation monitor reading of > 1 R/hr on radiation monitors 1RIA-57 or 58 (Unit 1), 2RIA-58 (Unit 2), and 3RIA-57 or 58 (Unit 3) indicates the release of reactor coolant to the containment. A containment radiation monitor reading of >1.6 R/hr on radiation monitor 2RIA-57 (Unit 2) also indicates the release of reactor coolant to the containment. The difference in these values is due to the relative strength of the detector check source which affects the background readings for the detector (the source for 2RIA-57 is stronger than that for the remaining detectors). This reading is less than that specified for Fuel Clad Barrier EAL#3. Thus, this EAL would be indicative of a RCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad Barrier EAL #3, fuel damage would also be indicated.

There is no "Potential Loss" EAL associated with this item.

6. Emergency Coordinator/EOF Director Judgment

This EAL is intended to address unanticipated conditions not addressed explicitly but warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator/EOF Director to fall under either the loss or potential loss of the RCS Barrier.

ENCLOSURE 4.1

FUEL CLAD BARRIER EALs: (1 or 2 or 3 or 4)

The Fuel Clad Barrier is the zircalloy tubes that contain the fuel pellets.

1. Core Exit Thermocouple Readings

The "Potential Loss" EAL reading corresponds to loss of subcooling. The value of 700 °F is indicative of superheated steam and is a value referenced in the Emergency Operating procedure. The loss of subcooling may lead to clad damage and, therefore, this is a potential loss of the fuel clad barrier.

The "Loss" EAL reading (1200 °F) indicates significant superheating of the coolant and core uncovery. Clad damage under these conditions is likely; therefore, this is indication of loss of the Fuel Clad Barrier.

2. Primary Coolant Activity Level

The value of 300 μCi/ml DEI coolant activity is well above that expected for iodine spikes and corresponds to about 4% fuel clad damage. This amount of clad damage indicates significant clad damage and thus the Fuel Clad Barrier is considered lost. Basis for determination is Engineering Calculation OSC-5283.
There is no equivalent "Potential Loss" EAL for this item.

3. Reactor Vessel Water Level

A valid reading of 0" on the RVLS (Reactor Vessel Level System) is an indicator that the fuel **could be** uncovered and would signify a potential loss of the fuel clad barrier. RVLS is invalid if LPI pumps are running and taking suction from the LPI drop line.

4. Containment Radiation Monitoring

Containment monitor readings on RIA 57/58 in the below listed table is higher than can be attributed to normal reactor coolant activity alone. These levels indicate that approximately 4% of the fuel cladding has failed which is consistent with the release of 300 uC/ml DEI to the containment atmosphere. Release of this amount of activity into containment corresponds to a loss of both the fuel clad and RCS barriers. Basis for the calculation which determined the activity levels can be found in Engineering calculation OSC-5283.

Hours Since SD	RIA 57	RIA 58
0 - <0.5	≥ 300	≥ 150
0.5 - < 2.0	≥ 80	≥ 40
2.0 - 8.0	≥ 32	≥ 16

There is no "Potential Loss" EAL associated with this item.

5. Emergency Coordinator/EOF Director Judgment

This EAL is intended to address unanticipated conditions not addressed explicitly but warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator/EOF Director to fall under either the loss or potential loss of the Fuel Clad Barrier.

ENCLOSURE 4.1

CONTAINMENT BARRIER EALs: (1 or 2 or 3 or 4 or 5 or 6)

The Containment Barrier includes the containment building, its connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve.

1. Containment Pressure

- Containment pressure above 59 psig (the design pressure) indicates that the containment or its heat removal systems are not functioning as intended. This degradation of containment pressure control represents a potential loss of containment integrity.

- Containment pressure of 10 psig with no reactor building cooling units or reactor building spray available represents a degradation in the control of the containment conditions. Therefore, this situation represents a potential loss of containment integrity.

- A containment hydrogen concentration greater than 9 percent volume is sufficient to expect that any ignition would result in complete combustion of the hydrogen in containment and a significant pressure rise. At hydrogen concentrations near 9 percent volume no challenge to containment integrity would be expected. At levels somewhat higher the possibility of a deflagration to detonation transition raises the uncertainty as to the actual response of the containment. Therefore, it is prudent that this level of hydrogen in the containment be considered a potential loss of containment integrity.

- Rapid unexplained loss of pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure increase indicates a loss of containment integrity.

  Containment pressure and sump levels should increase as a result of the mass and energy release into containment from a LOCA. Thus, sump level or pressure not increasing indicates an interfacing systems LOCA which is a containment bypass and a loss of containment integrity.

ENCLOSURE 4.1

2. Containment Isolation Valve Status After Containment Isolation

Failure to isolate those containment pathways which would allow containment atmosphere to be released to the environment is a loss of the containment barrier.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in-line charcoal filter does not make a release path indirect since the filter is not effective at removing fission product noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

There is no Potential Loss threshold associated with this item.

The decision of whether this EAL is satisfied should be based on present and readily available information. This includes physical data seen and heard. It is not the intent of this EAL to use relatively long term calculations to make the determination. If there is a pathway which would allow containment atmosphere to be released to the environment, this EAL is satisfied.

There is no "Potential Loss" EAL associated with this item.

3. SG Secondary Side Release With Primary To Secondary Leakage

Secondary side releases directly to the atmosphere include atmospheric dump valves and stuck open main steam safety valves. If the main condenser is available, there may be releases via air ejector, gland seal exhauster, and other similar controlled, and often monitored, pathways. These pathways do not meet the intent of a direct opening to the environment. These minor releases are assessed using Abnormal Rad Levels/Radiological Effluent Initiating Conditions. A failure of the secondary side which results in a direct opening to the environment, in combination with Primary to Secondary leakage $\geq$ 10 gpm in the same steam generator, constitutes a bypass of the containment, and therefore, a loss of the containment barrier.

Likewise, a failure of the secondary side which results in a direct opening to the environment, in combination with Primary to Secondary leakage $\geq$ 10 gpm in the other steam generator, constitutes a bypass of the containment, **IF** the SG with the secondary side failure is being fed feedwater from the affected unit. Therefore, this condition also constitutes a loss of the containment barrier.

In combination with the SG Tube Rupture EAL under the RCS barrier section, the appropriate classification can be determined.

There is no "Potential Loss" EAL associated with this item.

4. Significant Radioactive Inventory in Containment

Containment radiation readings shown in the table below are values which indicate significant fuel damage well in excess of the EALs associated with both loss of Fuel Clad and loss of RCS Barriers. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%. This amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment.

By treating the radioactive inventory in containment as a potential loss, a General Emergency will be declared when the conditions of the fuel clad and RCS barriers are included in the evaluation. This will allow the appropriate protective actions to be recommended.

Hours Since SD	RIA 57	RIA 58
0 - < 0.5	≥ 1800	≥ 860
0.5 - < 2.0	≥ 400	≥ 195
2.0 - 8.0	≥ 280	≥ 130

There is no "Loss" EAL associated with this item.

5. Core Exit Thermocouple

Core Exit Thermocouple temperatures ≥ 1200 °F or ≥ 700 °F with a valid RVLS reading for greater than 15 minutes, in this potential loss EAL represent imminent core damage that, if not terminated, could lead to vessel failure and an increased potential for containment failure. The potential for containment challenge as a result of events at reactor vessel failure makes it prudent to consider an unmitigated core damage condition as a potential loss of the containment barrier.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the reactor vessel in a significant fraction of the core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function

5. Core Exit Thermocouple

restoration procedures to arrest the core melt sequence. Whether or not the procedures will be effective should be apparent within 15 minutes. The Emergency Coordinator should make the declaration as soon as it is determined that the procedures have been, or will be ineffective.

There is no "Loss" EAL associated with this item.

6. Emergency Coordinator/EOF Director Judgement

This EAL is intended to address unanticipated conditions not addressed explicitly but warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator/EOF Director to fall under either the loss or potential loss of the Containment Barrier.

Reference

NUMARC/NESP-007, Rev 2, 01/92, Table 5-F-3

ENCLOSURE 4.2

SYSTEM MALFUNCTION

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
RCS Leakage	Unplanned Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a Significant Transient in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable	Inability to Monitor a Significant Transient in Progress	
Unplanned Loss of Most or All Safety System Annunciation or Indication in the Control Room for Greater than 15 minutes			
Inability to Reach Required Shutdown Within Technical Specification Limits			
Unplanned Loss of All Onsite or Offsite Communications			
Fuel Clad Degradation			

ENCLOSURE 4.2

SYSTEM MALFUNCTION

UNUSUAL EVENT

1. RCS Leakage

OPERATING MODE APPLICABILITY: **1,2,3,4**

EMERGENCY ACTION LEVELS:

A. Unidentified leakage ≥ 10 gpm

B. Pressure boundary leakage ≥ 10 gpm

C. Identified leakage ≥ 25 gpm
- Includes SG tube leakage

BASIS:
Reactor Coolant system (RCS) Leakage is defined in RCS Operational Leakage in the Technical Specifications Basis B 3.4.13.

This IC is included as an Unusual Event because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is observable with normal control room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. In either case, escalation of this IC to the Alert level is via Fission Product Barrier Degradation ICs or IC, Enclosure 4.4, Loss of Shutdown Function, "Inability to Maintain Plant in Cold Shutdown".

Reference
NUMARC/NESP-007, Rev. 2, 01/92, SU5

ENCLOSURE 4.2

SYSTEM MALFUNCTION

UNUSUAL EVENT

2. Unplanned Loss of Most or All Safety System Annunciation or Indication in The Control Room for Greater Than 15 Minutes.

OPERATING MODE APPLICABILITY: 1,2,3,4

EMERGENCY ACTION LEVEL:

The following conditions exist:

A. Unplanned loss of >50% of the following annunciators for greater than 15 minutes

Units 1&3	1SA 1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, 16 and 18 3SA 1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, 16 and 18
Unit 2	2SA 1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15 and 16

AND

In the opinion of the Operations Shift Manager, the loss of the annunciators or indicators requires additional personnel (beyond normal shift compliment) to safely operate the unit.

BASIS:

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

"Unplanned" loss of annunciators or indicator excludes scheduled maintenance and testing activities. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. Equipment monitored by referenced annunciator panel is shown on page 20.

This Unusual Event will be escalated to an Alert if a transient is in progress during the loss of annunciation or indication.

Due to the limited number of safety systems in operation during cold shutdown, refueling, and defueled modes, no IC is indicated during these modes of operation.

Reference NUMARC/NESP-007, Rev. 2, 01/92, SU3

ENCLOSURE 4.2

SYSTEM MALFUNCTION

UNUSUAL EVENT

3. Inability to Reach Required Shutdown Within Technical Specification Limits

OPERATING MODE APPLICABILITY: 1, 2, 3, 4

EMERGENCY ACTION LEVELS:

A. Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time.

BASIS:

Technical Specification Actions Statements require the plant to be brought to a required shutdown mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the site Technical Specifications requires a one hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate Notification of an Unusual Event is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. **Declaration of an Unusual Event is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed.** Other required Technical Specification shutdowns that involve precursors to more serious events are addressed by other System Malfunction, Hazards, or Fission Product Barrier Degradation ICs.

<u>Reference</u>

NUMARC/NESP-007, Rev. 2, 01/92, SU2

ENCLOSURE 4.2

SYSTEM MALFUNCTION

UNUSUAL EVENT

4. Unplanned Loss of All Onsite or Offsite Communications

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. Loss of all onsite communications capability(internal phone system, PA system, pager system, onsite radio system) affecting the ability to perform routine operations.

B. Loss of all offsite communications capability (Selective Signaling, ETS lines, offsite radio system, commercial phone system) affecting the ability to communicate with offsite authorities.

BASIS:

The purpose of this IC and its associated EALs is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities. The loss of offsite communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

This EAL is intended to be used only when extraordinary means are being utilized to make communications possible (relaying of information from radio transmissions, individuals being sent to offsite locations, etc.).

Reference
NUMARC/NESP-007, Rev. 2, 01/92, SU6

ENCLOSURE 4.2

SYSTEM MALFUNCTION

UNUSUAL EVENT

5. Fuel Clad Degradation.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVEL:

A. DEI > 5 uCi/ml

BASIS:

Chemistry analysis which indicates the presence of > 5 uci/ml dose equivalent iodine in the reactor coolant system clearly denotes a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The basis for the 5 uCi/ml is based upon the Oconee FSAR, Chapter 15, Table 15-14 of RCS Coolant Activity for 1% failed fuel. Escalation of this IC to the Alert level is via the Fission Product Barrier Degradation Monitoring ICs, Enclosure 4.1 of this document.

<u>Reference</u>
NUMARC/NESP-007, Rev. 2, 01/92, SU4

ENCLOSURE 4.2

SYSTEM MALFUNCTION

ALERT

1. Unplanned Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a Significant Transient in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable.

OPERATING MODE APPLICABILITY: 1, 2, 3, 4

EMERGENCY ACTION LEVEL:

The following conditions exist:

A. Unplanned loss of > 50% of the following annunciators for greater than 15 minutes.

Units 1&3	1SA 1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, 16, and 18 3SA 1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, 16, and 18
Unit 2	2SA 1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15 and 16

AND

In the opinion of the Operations Shift Manager, the loss of the annunciators or indicators requires additional personnel (beyond normal shift compliment) to safely operate the unit.

AND

Either of the following:

A significant plant transient is in progress.

OR

Loss of the OAC and PAM indications.

ENCLOSURE 4.2

SYSTEM MALFUNCTION

BASIS:

SA 1-9 : ES, RPS, CRD breakers, basic information concerning primary system, fire alarms, seismic trigger, condenser cooling, HPSW and LPSW system status.

SA 14-16: Electrical load (Keowee emergency start, load shed, emergency power switching logic)

SA-18 : CRD shunt trip relay, ICS, PZR relief valve flow, hydrogen concentration in RB, chlorine gas leakage.

This IC and its associated EAL are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a transient.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

"Significant Transient" includes response to automatic or manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

Significant indication is available from the OAC (operational aid computer) and from post accident monitoring (PAM). Loss of this data in conjunction with the loss of other indications would further impair the ability to monitor plant parameters.

Due to the limited number of safety systems in operation during cold shutdown, refueling and defueled modes, no IC is indicated during these modes of operation.

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress.

Reference
NUMARC/NESP-007, Rev. 2, 01/92/ SA4

ENCLOSURE 4.2

SYSTEM MALFUNCTION

SITE AREA EMERGENCY

1. Inability to Monitor a Significant Transient in Progress

OPERATING MODE APPLICABILITY: 1, 2, 3, 4

EMERGENCY ACTION LEVEL:
The following conditions exist:

A Unplanned loss of > 50% of the following annunciators for greater than 15 minutes.

Units 1&3	1SA 1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, 16, and 18 3SA 1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, 16, and 18
Unit 2	2SA 1, 2, 3, 4, 5, 6, 7, 8, 9, 14, 15, and 16

AND

A significant plant transient is in progress.

AND

Loss of the OAC and the PAM indications.

AND

Inability to directly monitor any one of the following functions:
- Subcriticality
- Inadequate core cooling
- Heat sink
- Containment Integrity
- RCS integrity
- RCS Inventory

BASIS:
This IC and its associated EAL are intended to recognize the inability of the control room staff to monitor the plant response to a transient. The inability to directly monitor indicates that computer data points or SPDS indicators are not available to monitor the critical safety functions.

ENCLOSURE 4.2

SYSTEM MALFUNCTION

SITE AREA EMERGENCY

"Significant Transient" includes response to automatic or manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% of greater.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, SS6

ENCLOSURE 4.3

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the SLC Limits for 60 Minutes or Longer	Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the SLC limits for 15 Minutes or Longer	Boundary Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mRem TEDE or 500 mRem CDE thyroid for the Actual or Projected Duration of the Release	Boundary Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity that Exceeds 1000 mRem TEDE or 5000 mRem CDE thyroid for the Actual or Projected Duration of the Release
Unexpected Increase in Plant Radiation Levels or Airborne Concentration	Major Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel		
	Release of Radioactive Material or Increases in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown		

ENCLOSURE 4.3

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

UNUSUAL EVENT

1. **Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the SLC Limits for 60 Minutes or Longer**

OPERATING MODE APPLICABILITY: ALL
EMERGENCY ACTION LEVELS:

A. A valid indication on radiation monitor RIA 33 of ≥ 4.06E+06 cpm for > 60 minutes. (See Note)

B. Valid indication on radiation monitor RIA-45 of ≥ 9.35E+05 cpm or RP sample reading of ≥ 6.62E-2uCi/ml Xe 133 eq for > 60 minutes. (See Note)

C. Confirmed sample analysis of liquid effluent being released exceeds two times SLC 16.11.1 for > 60 minutes as determined by Chemistry procedures.

D. Confirmed sample analysis of gaseous effluent being released exceeds two times SLC 16.11.2 for > 60 minutes as determined by Radiation Protection procedures.

Note: If monitor reading is sustained for the time period indicated in the EAL <u>AND</u> the required assessments (procedure calculations) cannot be completed within this period, declaration must be made on the valid Radiation monitor reading.

BASIS:
The term "Unplanned", as used in this context, includes any release for which a liquid waste release (LWR) or gaseous waste release (GWR) package was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable package.

Valid means that a radiation monitor reading has been confirmed to be correct.

ENCLOSURE 4.3

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

UNUSUAL EVENT

Chapter 16, Selected Licensee Commitments, of the Oconee Nuclear Station FSAR provides guidance to ensure that the release of liquid or gaseous effluent does not exceed the limits established in 10 CFR 20, Appendix B, Table II and Appendix I, 10 CFR 50. Unplanned releases in excess of two times the selected licensee commitments that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. It is not intended that the release be averaged over 60 minutes. The event should be declared as soon as it is determined that the release duration has or will likely exceed 60 minutes.

1. Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the SLC Limits for 60 Minutes or Longer

Monitor indications are based on the methodology of the site Offsite Dose Calculation Manual (ODCM). Annual average meteorology (semi-elevated 1.672E-06 sec/m3) has been used. Radiation Protection will use HP/0/B/1009/015 to quantify a gaseous release. Chemistry will use CP/0/B/5200/045 and/or CP/0/B/5200/048 to quantify a liquid release.

<u>**Reference**</u>
NUMARC/NESP/-007, Rev. 2, 01/92, AU1

ENCLOSURE 4.3

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

UNUSUAL EVENT

2. Unexpected Increase in Plant Radiation or Airborne Concentration.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. LT 5 reading 14" and decreasing with makeup not keeping up with leakage **WITH** fuel in the core

B. Valid indication of *uncontrolled* water decrease in the SFP or fuel transfer canal with all fuel assemblies remaining covered by water **AND** unplanned *valid* RIA 3, 6 or portable area monitor readings increase.

C. 1 R/hr radiation reading at one foot away from a damaged irradiated spent fuel dry storage module.

D. Valid area or process monitor exceeds limits stated in Enclosure 4.9 of RP/0/B/1000/001.

BASIS:

Valid means that a radiation monitor reading has been confirmed to be correct.

EAL 1 indicates that the water level in the reactor refueling cavity is uncontrolled. **If the area/process monitors reach the HIGH alarm setpoint, classification should be upgraded to an Alert.**

All of the above events tend to have long lead times relative to potential for radiological release outside the site boundary, thus impact to public health and safety is very low.

In light of reactor cavity seal failure incidents, explicit coverage of these types of events via EALs 1 and 2 is appropriate given their potential for increased doses to plant staff. Classification as an Unusual Event is warranted as a precursor to a more serious event.

EAL 3 applies to licensed dry storage of older irradiated spent fuel to address degradation of this spent fuel.

ENCLOSURE 4.3

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

UNUSUAL EVENT

EAL 4 addresses unplanned increases in in-plant radiation levels that represent a degradation in the control of radioactive material, and represent a potential degradation in the level of safety of the plant. The RIA readings for an Unusual Event are 1000 times the normal value. Enclosure 4.9 of RP/0/B/1000/001 will provide the actual readings for the monitors.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, AU2
NEI 99-01, Rev. 4, 08/00, AU2

ENCLOSURE 4.9 (RP/O/B/1000/001)

UNEXPECTED/UNPLANNED INCREASE IN AREA MONITOR READINGS

This initiating condition is not intended to apply to anticipated temporary increases due to planned events (e.g., incore detector movement, radwaste container movement, depleted resin transfers, etc.)

MONITOR NUMBER	UNIT 1, 2, 3	
	UNUSUAL EVENT 1000 x normal levels mRad/hr	ALERT mRad/hr
RIA 7, Hot Machine Shop Elevation 796	150	$\geq$ 5000
RIA 8, Hot Chemistry Lab Elevation 796	4200	$\geq$ 5000
RIA 10, Primary Sample Hood, Elevation 796	830	$\geq$ 5000
RIA 11, Change Room Elevation 796	210	$\geq$ 5000
RIA 12, Chem Mix Tank Elevation 783	800	$\geq$ 5000
RIA 13, Waste Disposal Sink, Elevation 771	650	$\geq$ 5000
RIA 15, HPI Room Elevation 758	**NOTE***	$\geq$ 5000

NOTE*: RIA 15 normal readings are approximately 9 mRad/hr on a daily basis. Applying the 1000 x normal readings would put this monitor greater than 5000 mRad/hr just for an Unusual Event. For this reason, an Unusual Event will not be declared for any reading less than 5000 mRad/hr

ENCLOSURE 4.3

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

ALERT

1. Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times Radiological Technical Specifications for 15 Minutes or Longer.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. Valid indication of RIA-46 of ≥ 2.09E+04 cpm or RP sample reading of ≥ 6.62 uCi/ml Xe 133 eq for > 15 minutes (See Note)

B RIA 33 HIGH alarm AND Liquid effluent being released exceeds 200 times the level of SLC 16.11.1 for > 15 minutes as determined by Chemistry procedure.

C. Gaseous effluent being released exceeds 200 times the level of SLC 16.11.2 for > 15 minutes as determined by RP procedure.

Note: If monitor reading is sustained for the time period indicated in the EAL AND required assessments (procedure calculations) cannot be completed within this period, declaration must be made on the valid Radiation monitor reading.

BASIS:

The term "Unplanned", as used in this context, includes any release for which a liquid waste release (LWR) or gaseous waste release (GWR) package was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable package.

Valid means that a radiation monitor reading has been confirmed to be correct.

This event escalates from the Unusual Event by escalating the magnitude of the release by a factor of 100.

ENCLOSURE 4.3

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

ALERT

It is not intended that the release be averaged over 15 minutes. The event should be declared as soon as it is determined that the release duration has or will likely exceed 15 minutes.

Monitor indications are based on the methodology of the site Offsite Dose Calculation Manual (ODCM). Annual average meteorology (semi-elevated release 1.672 E-06 sec/m3) has been used.

Chapter 16, Selected Licensee Commitments, of the Oconee Nuclear Station FSAR outlines the release limits for gaseous effluent is released by the Control Room. Liquid effluent is discharged by Chemistry from the Radwaste Facility. Effluent monitors have setpoints established to alarm should activity be detected that would exceed limits established by 10 CFR 20, Table B, Appendix II. Radiation Protection and/or Chemistry would calculate the release rate and quantify the amount being released.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, AA1

ENCLOSURE 4.3

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

ALERT

2. Release of Radioactive Material or Increases in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. Valid radiation reading $\geq$ 15 mRad/hr in the Control Room, CAS, or Radwaste Control Room.

B. Unplanned/unexpected valid area radiation monitor readings exceed limits stated in Enclosure 4.9 of RP/0/B/1000/001.

BASIS:
Valid means that a radiation reading has been confirmed by the operators to be correct.

This IC addresses unplanned/unexpected increased radiation levels that impede necessary access to operating stations, or other areas containing equipment that must be operated manually, in order to maintain safe operation or perform a safe shutdown. It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant.

The Control Room, Central Alarm Station (CAS) and the Radwaste Control Room are areas that will need to be continuously occupied. No radiation monitors are in the CAS or the Radwaste Control Room.

Oconee has chosen to use a generic emergency action level of greater than or equal to 5000 mRad/hr for the Alert classification for areas in the plant that would need to be utilized for safe operation or safe shutdown of the unit. Enclosure 4.9 of RP/0/B/1000/001 provides the monitor number and the location of the area monitor.

This IC is not intended to apply to anticipated temporary increases due to planned events (e.g., incore detector movement, radwaste container movement, depleted resin transfers, etc.)

Reference
NUMARC/NESP-007, Rev. 2, 01/92, AA3

ENCLOSURE 4.3

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

ALERT

3. Major Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. Valid RIA 3*, 6, 41 or 49* **HIGH** alarm readings
*Applies to Mode 6 and No Mode Only

B. Valid **HIGH** alarm reading on portable area monitors on the main bridge or spent fuel pool bridge.

C. Report of visual observation of irradiated fuel uncovered.

D. Operators determine water level drop in either the SFP or fuel transfer canal will exceed makeup capacity such that irradiated fuel will be uncovered.

BASIS:

This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

The HIGH alarm for RIA 3 (containment area monitor) and RIA 49 (RB gaseous process monitor) corresponds to the setpoints established to assure that 10 CFR 20 limits are not exceeded.

The HIGH alarm setpoint for RIA 6 (SFP bridge area monitor) is designed to make operators aware of increased readings above 10 CFR 20 limits. The HIGH alarm setpoint for RIA 41 (Spent Fuel Pool gaseous atmosphere) is set to alarm if 4 times the limits of 10 CFR 20 are exceeded based upon Xe-133. RIA 49 monitors the reactor building gas. Portable monitors are established during refueling outages and are located on the main bridge, and the spent fuel pool bridge.

ENCLOSURE 4.3

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

ALERT

There is time available to take corrective actions, and there is little potential for substantial fuel damage. Thus, an Alert Classification for this event is appropriate. Escalation, if appropriate, would occur via Abnormal Rad Level/Radiological Effluent or Emergency Coordinator Judgement.

Reference NUMARC/NESP-007, Rev. 2, 01/92, AA2

ENCLOSURE 4.3

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

SITE AREA EMERGENCY

1. Boundary Dose Resulting from an Actual or Imminent Release of Radioactivity Exceeds 100 mRem TEDE or 500 mRem CDE Adult Thyroid for the Actual or Projected Duration of the Release.

OPERATING MODE APPLICABILITY: ALL
EMERGENCY ACTION LEVELS:

A. Valid reading on RIA-46 of ≥ 2.09E+05 cpm or RIA 56 reading of ≥ 17.5 R/hr or RP sample reading of 6.62E+01 uCi/ml Xe 133 eq for > 15 minutes. (See Note)

B. Valid reading on RIA 57 or 58 as shown on Enclosure 4.8 of RP/0/B/1000/001. (See Note)

C. Dose calculations result in a dose projection at the site boundary of 100 mRem TEDE or 500 mRem CDE Adult Thyroid.

D. Field survey results indicate site boundary dose rates exceeding 100 mRad/hr expected to continue for more than one hour; **OR** analysis of field survey samples indicate adult thyroid dose commitment (CDE) of 500 mRem for one hour of inhalation.

Note: If actual Dose Assessment cannot be completed within 15 minutes, then the valid monitor reading should be used for emergency classification.

BASIS:

Valid means that a radiation monitor reading has been confirmed to be correct.
The calculation for RIA 46 (vent monitor) setpoint is based on whole body dose (100 mRem) using ODCM guidance: average annual meteorology (semi-elevated release 1.672E-6 sec/m3), vent flow rate of 65,000 cfm, and release duration of 15 minutes. No credit is taken for vent filtration.

The calculation for RIA 57/58 (incontainment monitors) setpoints are based on the following: LOCA conditons which provide the more conservative reading, Committed Dose Equivalent (CDE) thyroid (500 mRem), average annual meteorology (7.308 E-6 sec/m3), design basis leakage of 5.6E6 ml/hr, release duration of one hour, and time since unit trip. No credit is taken for filtration.

ENCLOSURE 4.3

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

SITE AREA EMERGENCY

Dose assessment team members use actual meteorology, release duration, and unit vent flow rate or actual leakage rate from containment. Therefore, the predetermined monitor readings would not be used if dose assessment team calculations are available from the TSC or EOF in a timely manner (within approximately 15 minutes).

The 100 mRem Total Effective Dose Equivalent (TEDE) and the 500 mRem Committed Dose Equivalent (CDE) thyroid in this initiating condition is based on 10 CFR 20 annual average population exposure. The dose projection uses a 4-hour default for time of release. If the real time release time is known it will be used in the calculation. One order of magnitude is the gradient factor between the Site Area Emergency and General Emergency classes. These values are 10% of the EPA PAG values given in EPA-400-R-92-001.

The field monitoring survey results are based on actual hand-held instrument readings at the site boundary. It is assumed that the release will continue for more than one hour. Adult thyroid is considered to be the limiting factor.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, AS1

ENCLOSURE 4.8 (RP/0/B/1000/001)
RADIATION MONITOR READINGS FOR EMERGENCY CLASSIFICATION

NOTE: IF ACTUAL DOSE ASSESSMENT CANNOT BE COMPLETED WITHIN 15 MINUTES, THEN THE VALID MONITOR READING SHOULD BE USED FOR EMERGENCY CLASSIFICATION.

ALL RIA VALUES ARE CONSIDERED TO BE GREATER THAN OR EQUAL TO.

HOURS SINCE REACTOR TRIPPED	RIA 57 R/hr		RIA 58 R/hr*	
	Site Area Emergency	General Emergency	Site Area Emergency	General Emergency
0 - < 0.5	5.9E+003	5.9E+004	2.6E+003	2.6E+004
0.5 - < 1.0	2.6E+003	2.6E+004	1.1E+003	1.1E+004
1.0 - < 1.5	1.9E+003	1.9E+004	8.6E+002	8.6E+003
1.5 - < 2.0	1.9E+003	1.9E+004	8.5E+002	8.5E+003
2.0 - < 2.5	1.4E+003	1.4E+004	6.3E+002	6.3E+003
2.5 - < 3.0	1.2E+003	1.2E+004	5.7E+002	5.7E+003
3.0 - < 3.5	1.1E+003	1.1E+004	5.2E+002	5.2E+003
3.5 - < 4.0	1.0E+003	1.0E+004	4.8E+002	4.8E+003
4.0 - < 8.0	1.0E+003	1.0E+004	4.4E+002	4.4E+003

*Note: RIA 58 is partially shielded.

Assumptions used for calculation of high range in-containment monitors RIA 57 and 58:

1. Average annual meteorology (7.308 E-6 sec/m3)
2. Design basis leakage (5.6 E6 ml/hr)
3. One hour release duration
4. General Emergency PAGs are 1 rem TEDE and 5 rem CDE; SAE determination is based on 10% of the General Emergency PAGs.
5. Calculations for monitor readings are based on CDE (adult thyroid - 500 mRem) because thyroid dose is limiting.
6. No credit is taken for filtration.
7. LOCA conditions are limiting and provide the more conservative reading.

Assumptions used for calculation of vent monitor RIA 46:

1. Average annual meteorology (1.672 E-6 sec/m3), semi-elevated
2. Vent flow rate 65,000 cfm (average daily flow rate)
3. No credit is taken for vent filtration
4. Fifteen minute release duration.
5. General Emergency PAGs are 1 rem TEDE and 5 rem CDE; SAE determination is based on 10% of the General Emergency PAGs.
6. Calculations for monitor readings are based on whole body dose (100 mRem).
7. Calculation is based on ODCM methodology and NUMARC guidance

ENCLOSURE 4.3

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

SITE AREA EMERGENCY

2. Loss of Water Level in the Reactor Vessel That Has or Will Uncover Fuel in the Reactor Vessel.

OPERATING MODE APPLICABILITY: 5, 6

EMERGENCY ACTION LEVEL:

Loss of Reactor Vessel Water Level as indicated by:

A. Failure of heat sink causes loss of Mode 5 (Cold Shutdown) conditions **AND** LT-5 indicates 0 inches after initiation of RCS makeup.

B. Failure of heat sink causes loss of Mode 5 (Cold Shutdown) conditions **AND** either train ultrasonic level indication less than 0 inches and decreasing after initiation of RCS makeup.

BASIS:

Under the conditions specified by this IC, severe core damage can occur due to prolonged boiling following loss of decay heat removal. Declaration of a Site Area Emergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via Enclosure 4.3, Abnormal Rad Levels/Radiological Effluent.

Note: Both the LT-5 and the ultrasonic level instrumentation are located in the center line of the hot leg.

Reference
NUMARC/NESP-007, Rev.2, 01/92, SS5

ENCLOSURE 4.3

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

GENERAL EMERGENCY

1. **Boundary Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity that Exceeds 1000 mRem TEDE or 5000 mRem (CDE) Adult Thyroid for the Actual or Projected Duration of the Release Using Actual Meteorology.**

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. Valid reading on RIA 46 of ≥ 2.09E+06 cpm or RIA 56 reading of ≥ 175 R/hr or RP sample reading of 6.62E +02uCi/ml Xe 133 eq for ≥ 15 minutes (See Note)

B. Valid reading on RIA 57 or 58 as shown on Enclosure 4.8 of RP/0/B/1000/001. (See Note)

C. Dose calculations result in a dose projection at the site boundary of ≥ 1000 mRem TEDE **OR** ≥ 5000 mRem CDE (Adult Thyroid).

D. Field survey results indicate site boundary dose rates exceeding 1000 mRad/hr expected to continue for more than one hour; **OR** analyses of field survey samples indicate adult thyroid commitment (CDE) of 5000 mRem for one hour of inhalation.

Note: **If actual Dose Assessment cannot be completed within 15 minutes, then the valid monitor reading should be used for emergency classification.**

BASIS:

Valid means that a radiation monitor reading has been confirmed to be correct.
The calculation for RIA 46 (vent monitor) setpoint is based on the following: whole body dose (100 mRem) using ODCM guidance, average annual meteorology (semi-elevated release 1.672E-6 sec/m3), vent flow rate of 65,000 CFM, and release duration of 15 minutes. No credit is taken for vent filtration.

The calculation for RIA 57/58 (incontainment monitors) setpoints are based on the following: LOCA conditions which provide the more conservative reading, Committed Dose Equivalent (CDE-adult thyroid 500 mRem), average annual meteorology (7.308 E-6, sec/m3), design basis leakage of 5.6E6 ml/hr, release duration of one hour, and time since unit trip. No credit is taken for filtration.

ENCLOSURE 4.3

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT

GENERAL EMERGENCY

Calculations by the dose assessment team use **actual** meteorology, duration, and unit vent flow rate or actual leakage rate from containment. Therefore, the predetermined monitor readings would not be used if dose assessment calculations are available from the TSC or EOF in a timely manner (within approximately 15 minutes).

The 1000 mRem Total Effective Dose Equivalent (TEDE) and the 5000 mRem Committed Dose Equivalent (CDE) adult thyroid in this initiating condition is based on 10 CFR 20 annual average population exposure. These values are EPA PAG guidelines as expressed in EPA-400-R-92-001. The dose calculation procedure utilizes a default of 4 hours for the release time. This default value will be utilized until a corrected release time is determined.

Field monitoring results will utilize a one hour period of time for calculating survey results.

Enclosure 4.8 of RP/0/B/1000/001 is shown on page 34 of this document.

<u>Reference</u>
NUMARC/NESP-007, Rev. 2, 01/92, AG1

ENCLOSURE 4.4

LOSS OF SHUTDOWN FUNCTION

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Unexpected increase in plant radiation levels or airborne concentrations	Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram Was Successful	Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram Was NOT Successful	Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core
	Inability to Maintain Plant in Cold Shutdown	Complete Loss of Function Needed to Achieve or Maintain Hot Shutdown	
	Major damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the reactor vessel	Loss of Water Level in the Reactor Vessel That Has or Will Uncover Fuel in the Reactor Vessel	

ENCLOSURE 4.4

LOSS OF SHUTDOWN FUNCTIONS

UNUSUAL EVENT

1. Unexpected Increase in Plant Radiation or Airborne Concentration.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. LT 5 reading 14" and decreasing with makeup not keeping up with leakage **WITH** fuel in the core

B. Valid indication of *uncontrolled* water decrease in the SFP or fuel transfer canal with all fuel assemblies remaining covered by water **AND** unplanned *valid* RIA 3, 6 or portable area monitor readings increase.

C. 1 R/hr radiation reading at one foot away from a damaged irradiated spent fuel dry storage module.

D. Valid area or process monitor exceeds limits stated in Enclosure 4.9 of RP/0/B/1000/001.

BASIS:

Valid means that a radiation monitor reading has been confirmed to be correct.

EAL 1 indicates that the water level in the reactor refueling cavity is uncontrolled. **If the area/process monitors reach the HIGH alarm setpoint, classification should be upgraded to an Alert.**

All of the above events tend to have long lead times relative to potential for radiological release outside the site boundary, thus impact to public health and safety is very low.

In light of reactor cavity seal failure incidents, explicit coverage of these types of events via EALs 1 and 2 is appropriate given their potential for increased doses to plant staff. Classification as an Unusual Event is warranted as a precursor to a more serious event.

EAL 3 applies to licensed dry storage of older irradiated spent fuel to address degradation of this spent fuel.

ENCLOSURE 4.4

LOSS OF SHUTDOWN FUNCTIONS

UNUSUAL EVENT

EAL 4 addresses unplanned increases in in-plant radiation levels that represent a degradation in the control of radioactive material, and represent a potential degradation in the level of safety of the plant. The RIA readings for an Unusual Event are 1000 times the normal value. Enclosure 4.9 of RP/0/B/1000/001 will provide the actual readings for the monitors.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, AU2
NEI 99-01, Rev. 4, 08/00, AU2

ENCLOSURE 4.4

LOSS OF SHUTDOWN FUNCTIONS

ALERT

1. Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram Was Successful.

OPERATING MODE APPLICABILITY: 1, 2, 3

EMERGENCY ACTION LEVEL:
The following conditions exist:

A. VALID reactor trip signal received or required without automatic scram

AND ONE OF THE FOLLOWING:

DSS has inserted Control Rods

OR

Manual reactor trip from the control room is successful and reactor power is less than 5% and decreasing.

BASIS:

This condition indicates failure of the automatic protection system to scram the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS. Reactor protection system setpoint being exceeded (rather than limiting safety system setpoint being exceeded) is specified here because failure of the automatic protection system is the issue. If the reactor protective system fails, the Diverse Scram Signal system (which was installed at Oconee since 10/7/91 as a result of Generic Letter 83-28) will drop control rod groups 5,6,7 into the core.

A manual scram is any set of actions by the reactor operator(s) at the reactor control console which causes control rods to be RAPIDLY inserted into the core and brings the reactor subcritical.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, SA2

ENCLOSURE 4.4

LOSS OF SHUTDOWN FUNCTIONS

ALERT

Operator action to drive rods does **NOT** constitute a reactor trip, (i.e. does not meet the rapid insertion criterion).

Failure of Diverse Scram Signal and the manual scram would escalate the event to a Site Area Emergency.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, SA2

ENCLOSURE 4.4

LOSS OF SHUTDOWN FUNCTIONS

ALERT

2. Inability to Maintain Plant in Mode 5 (Cold Shutdown).

OPERATING MODE APPLICABILITY: 5, 6

EMERGENCY ACTION LEVELS:

A. Loss of LPI and/or LPSW

AND

Inability to maintain RCS temperature below 200 °F as indicated by either of the following:
RCS temperature at the LPI pump suction

OR

Average of the 5 highest CETCs as indicated by ICCM display.

OR

Visual observation

BASIS:
LPI is the low pressure injection system
LPSW is low pressure service water.

This IC is based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." number of phenomena such as pressurization, vortexing, RCS level differences when operating at a mid-loop condition, decay heat removal system design, and level instrumentation problems can lead to conditions where decay heat removal is lost and core uncovery can occur. NRC analyses show sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

Loss of the LPI system and/or the LPSW system causes an uncontrolled temperature rise in the reactor coolant system. Uncontrolled is understood to be "not as the result of operator action." Rising temperature of the reactor coolant system can be determined at the LPI pump suction, average of the 5 highest CETCs as indicated by ICCM display or through operator visual observation (steam or boiling) in the reactor building.

ENCLOSURE 4.4

LOSS OF SHUTDOWN FUNCTIONS

ALERT

With a loss of LPI pumps there will be no RCS flow at the LPI pump suction and RCS temperature at that point will not represent RCS temperature in the reactor vessel. Also, with the reactor head in place, visual observation may not be possible.

Escalation to the Site Area Emergency is by, "Loss of Water Level in the Reactor Vessel That Has or Will Uncover Fuel in the Reactor Vessel," or by Abnormal Rad Levels/Radiological Effluent ICs.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, SA3

ENCLOSURE 4.4

LOSS OF SHUTDOWN FUNCTIONS

ALERT

3. Major Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. Valid RIA 3*, 6, 41 or 49* **HIGH** alarm readings
Applies to Mode 6 and No Mode Only.

B. Valid **HIGH** alarm reading on portable area monitors on the main bridge or spent fuel pool bridge.

C. Report of visual observation of irradiated fuel uncovered.

D. Operators determine water level drop in either the SFP or fuel transfer canal will exceed makeup capacity such that irradiated fuel will be uncovered.

BASIS:

This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

The HIGH alarm for RIA 3 (containment area monitor) and RIA 49 (RB gaseous process monitor) corresponds to the setpoints established to assure that 10 CFR 20 limits are not exceeded.

The HIGH alarm setpoint for RIA 6 (SFP bridge area monitor) is designed to make operators aware of increased readings above 10 CFR 20 limits. The HIGH alarm setpoint for RIA 41 (Spent Fuel Pool gaseous atmosphere) is set to alarm if 4 times the limits of 10 CFR 20 are exceeded based upon Xe-133. RIA 49 monitors the reactor building gas. Portable monitors are established during refueling outages and are located on the main bridge, and the spent fuel pool bridge.

ENCLOSURE 4.4

LOSS OF SHUTDOWN FUNCTIONS

ALERT

There is time available to take corrective actions, and there is little potential for substantial fuel damage. Thus, an Alert Classification for this event is appropriate. Escalation, if appropriate, would occur via Abnormal Rad Level/Radiological Effluent, Loss of Shutdown Functions or Emergency Coordinator Judgement.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, AA2

ENCLOSURE 4.4

LOSS OF SHUTDOWN FUNCTIONS

SITE AREA EMERGENCY

1. Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram Was NOT Successful.

OPERATING MODE APPLICABILITY: 1, 2

EMERGENCY ACTION LEVEL:

The following conditions exist:

A. VALID reactor trip signal received or required without automatic scram

AND

DSS has **NOT** inserted Control Rods

AND

Manual reactor trip from the control room was not successful in reducing reactor power to less than 5% and decreasing.

BASIS:

Automatic and manual scram are not considered successful if action away from the reactor control console is required to scram the reactor.

This EAL is met if a reactor trip is required and the manual reactor trip function fails. A failure of the manual reactor trip pushbutton to initiate a reactor trip is indication of a failure of the Reactor Protection System.

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency is indicated because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response. Escalation of this event to a General Emergency would be via Fission Product Barrier Degradation or Emergency Coordinator Judgement ICs.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, SS2

ENCLOSURE 4.4

LOSS OF SHUTDOWN FUNCTIONS

SITE AREA EMERGENCY

2. Complete Loss of Function Needed to Achieve or Maintain Mode 4 (Hot Shutdown).

OPERATING MODE APPLICABILITY: 1, 2, 3, 4

EMERGENCY ACTION LEVELS:

Any of the following conditions exist:

A. Average of the 5 highest CETCs ≥ 1200 °F on ICCM.

B. Unable to maintain reactor subcritical

C. EOP directs feeding SG from SSF ASWP or station ASWP

BASIS:

This EAL addresses complete loss of functions, core cooling and heat sink, required for hot shutdown with the reactor at pressure and temperature. Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted. Escalation to General Emergency would be via Abnormal Rad Levels/Radiological Effluent, Emergency Coordinator Judgment, or Fission Product Barrier Degradation ICs.

Core exit thermocouple readings are considered to be the average of the five (5) highest thermocouple readings shown on the Inadequate Core Cooling Monitor.

The SSF can provide the following: (1) makeup to the Reactor Coolant pump seals, (2) low pressure service water to the steam generators (additional method for heat sink), (3) capability to keep the unit in hot shutdown for 72 hours following an Appendix R fire.

<u>**Reference**</u>
NUMARC/NESP-007, Rev. 2, 01/92, SS4

ENCLOSURE 4.4.

LOSS OF SHUTDOWN FUNCTIONS

SITE AREA EMERGENCY

3. Loss of Water Level in the Reactor Vessel That Has or Will Uncover Fuel in the Reactor Vessel.

OPERATING MODE APPLICABILITY: 5, 6

EMERGENCY ACTION LEVEL:

Loss of Reactor Vessel Water Level as indicated by:

A. Failure of heat sink causes loss of Mode 5 (Cold Shutdown) conditions.

AND

LT-5 indicates 0 inches after initiation of RCS makeup.

B. Failure of heat sink causes loss of Mode 5 (Cold Shutdown) conditions.

AND

Either train ultrasonic level indication less than 0 inches and decreasing after initiation of RCS makeup.

BASIS:

Under the conditions specified by this IC, severe core damage can occur due to prolonged boiling following loss of decay heat removal. Declaration of a Site Area Emergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via Enclosure 4.3, Abnormal Rad Levels/Radiological Effluent.

Note: Both the LT-5 and the ultrasonic level instrumentation are located in the center line of the hot leg.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, SS5

ENCLOSURE 4.4

LOSS OF SHUTDOWN FUNCTIONS

GENERAL EMERGENCY

1. **Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core.**

OPERATING MODE APPLICABILITY: 1, 2

EMERGENCY ACTION LEVEL:

The following conditions exist:

A. VALID reactor trip signal received or required **WITHOUT** automatic scram

AND

Manual reactor trip from the control room was not successful in reducing reactor power to less than 5% and decreasing.

AND

Average of five highest CETCs ≥ 1200 °F on the ICCM.

BASIS:
Automatic and manual scram are not considered successful if action away from the reactor control console is required to scram the reactor. Under the conditions of the IC and its associated EALs, the efforts to bring the reactor subcritical have been unsuccessful and, as a result, the reactor is producing more heat than the maximum decay heat load for which the safety systems were designed. The extreme challenge to the ability to cool the core is intended to mean that the core exit temperatures are at or approaching 1200 °F. (Note: CETCs reading ≥ 1200 °F is also a good indicator that the reactor vessel water level is below the top of the active fuel. Oconee does not have an indication for the reactor vessel water level below the top of the active fuel.)

The General Emergency declaration is intended to be anticipatory of the fission product barrier matrix declaration to permit maximum offsite intervention time.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, SG2

ENCLOSURE 4.5

LOSS OF POWER

Error! Bookmark not defined.UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes	Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses During Cold Shutdown Or Refueling Mode	Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses	Prolonged Loss of All (Offsite and Onsite) AC Power
Unplanned Loss of Required DC Power During Cold Shutdown or Refueling Mode for Greater than 15 Minutes	AC power to essential busses reduced to a single power source for greater than 15 minutes such that an additional single failure could result in station blackout	Loss of All Vital DC Power	

ENCLOSURE 4.5

LOSS OF POWER

UNUSUAL EVENT

1. Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes.

OPERATING MODE APPLICABILITY ALL

EMERGENCY ACTION LEVEL:

The following conditions exist:

A. Unit auxiliaries being supplied from Keowee or CT5.

AND

Inability to energize either MFB from an offsite source (either switchyard) within 15 minutes.

BASIS:

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (Station Blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Keowee Hydro station provides the emergency power to the Oconee Nuclear Site. CT5 is powered from the Lee Steam Station and provides back-up power to the site.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, SU1

ENCLOSURE 4.5

LOSS OF POWER

UNUSUAL EVENT

2. Unplanned Loss of Required DC Power During Mode 5 (Cold Shutdown) or Mode 6 (Refueling Mode) for Greater than 15 Minutes.

OPERATING MODE APPLICABILITY: 5, 6

EMERGENCY ACTION LEVEL:

The following conditions exist:

A. Unplanned Loss of Vital DC power to required DC busses as indicated by bus voltage less than 110 VDC.

AND

Failure to restore power to at least one required DC bus within 15 minutes from the time of loss.

BASIS:

The purpose of this IC and its associated EALs is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

"Unplanned" is included in this IC and EAL to preclude the declaration of an emergency as a result of planned maintenance activities.

If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will be per Enclosure 4.4, Loss of Shutdown Functions "Inability to Maintain Plant in Cold Shutdown."

Reference
NUMARC/NESP-007, Rev. 2, 01/92, SU7

ENCLOSURE 4.5

LOSS OF POWER

ALERT

1. Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses During Mode 5 (Cold Shutdown) Or Mode 6 (Refueling Mode).

OPERATING MODE APPLICABILITY: 5, 6, Defueled

EMERGENCY ACTION LEVEL:

The following conditions exist:

A. MFB 1 and 2 de-energized.

AND

Failure to restore power to at least one main feeder bus within 15 minutes from the time of loss of both offsite and onsite AC power.

BASIS:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal and the Ultimate Heat Sink. When in cold shutdown, refueling, or defueled mode the event can be classified as an Alert, because of the significantly reduced decay heat, lower temperature and pressure, increasing the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL. Escalating to Site Area Emergency, if appropriate, is by Enclosure 4.3, Abnormal Rad Levels/Radiological Effluent, or Enclosure 4.7, Natural Disasters, Hazards, and Other Conditions Affecting Plant Safety, Emergency Coordinator Judgement ICs. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

References

NUMARC/NESP-007, Rev. 2, 01/92, SA1

ENCLOSURE 4.5

LOSS OF POWER

ALERT

2. AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that an additional single failure could result in station blackout.

OPERATING MODE APPLICABILITY: 1, 2, 3, 4

EMERGENCY ACTION LEVEL:

The following condition exists:

A. AC power capability has been degraded to a single power source for > 15 min. due to the loss of all but one of the following:

Unit Normal Transformer (backcharged)
Unit Startup transformer
Another Unit Startup Transformer (aligned)
CT4
CT5

BASIS:

This IC and the associated EAL is intended to provide an escalation from IC, "Loss of All Offsite Power To Essential Busses for Greater Than 15 Minutes." The condition indicated by this IC is the degradation of the offsite and onsite power systems such that an additional single failure could result in a station blackout. In this particular situation, a station blackout applies to the unit in question even though the other units may not be affected. This condition could occur due to a loss of offsite power with a concurrent failure of either CT4 or CT5 to supply power to the main feeder busses.

The subsequent loss of this single power source would escalate the event to a Site Area Emergency in accordance with IC, "Loss of All Offsite and Loss of All Onsite AC Power to Essential Busses."

Reference
NUMARC/NESP-007, Rev. 2, 01/92, SA5

ENCLOSURE 4.5

LOSS OF POWER

SITE AREA EMERGENCY

1. Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses

OPERATING MODE APPLICABILITY: 1, 2, 3, 4

EMERGENCY ACTION LEVEL:

Loss of all offsite and onsite AC power as indicated by:

A. MFB 1 and 2 de-energized

AND

Failure to restore power to at least one main feeder bus within 15 minutes from the time of loss of both offsite and onsite AC power.

BASIS:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will cause core uncovering and loss of containment integrity, thus this event can escalate to a General Emergency.

Escalation to General Emergency is via Enclosure 4.1 Fission Product Barrier Degradation or IC, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power."

Loss of offsite power (6900V) eliminates the use of power from Duke Power grid and also eliminates distribution of power from the unit generator. Loss of onsite AC (4160V) which includes both Keowee Hydro units, eliminates the use of HPI pumps, LPI pumps, reactor building spray pumps, low pressure service water pumps, CCW pumps, condensate booster pumps, hotwell pumps, heater drain pumps and motor driven emergency feedwater pumps. Turbine driven emergency feedwater pumps are assumed to be available. It is assumed for this scenario that the Standby Shutdown Facility would be available for RCS and secondary inventory control utilizing the RC makeup pump and the auxiliary service water pump.

References

NUMARC/NESP-007, Rev. 2, 01/92, SS1

ENCLOSURE 4.5

LOSS OF POWER

SITE AREA EMERGENCY

2. Loss of All Vital DC Power.

OPERATING MODE APPLICABILITY: 1, 2, 3, 4

EMERGENCY ACTION LEVEL:

The following conditions exist:

A. Unplanned Loss of Vital DC power to required DC busses as indicated by bus voltage less than 110 VDC.

AND

Failure to restore power to at least one required DC bus within 15 minutes from the time of loss.

BASIS:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. Escalation to a General Emergency would occur by Enclosure 4.3, Abnormal Rad Levels/Radiological Effluent, Enclosure 4.1, Fission Product Barrier Degradation, Enclosure 4.7, Natural Disasters, Hazards and Other Conditions Affecting Plant Safety or Emergency Coordinator Judgement ICs. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

ENCLOSURE 4.5

LOSS OF POWER

SITE AREA EMERGENCY

The purpose of the onsite DC Power system is:

1. Provide a source of reliable, continuous power for instrumentation and controls needed for normal operation and safe shutdown of the unit through the vital DC power distribution system panelboards and essential DC power which feed Inverters for an uninterrupted source of AC power.

2. Supply DC motor operated valves and pumps required during normal operation and a total loss of AC.

Loss of DC power would place the plant in a situation of losing vital instrumentation, valves, and pumps needed to safely operate and shutdown the plant any time the unit is above cold shutdown conditions.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, SS3

<u>ENCLOSURE 4.5</u>

<u>LOSS OF POWER</u>

GENERAL EMERGENCY

1. Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power.

OPERATING MODE APPLICABILITY: 1, 2, 3, 4

EMERGENCY ACTION LEVEL:

Prolonged loss of all offsite and onsite AC power as indicated by:

A. MFB 1 and 2 de-energized

<u>AND</u>

Standby Shutdown Facility (SSF) fails to maintain Mode 3 (Hot Standby).

<u>AND</u>

AT LEAST ONE OF THE FOLLOWING:

Restoration of power to at least one MFB within 4 hours is <u>NOT</u> likely

<u>OR</u>

Indication of continuing degradation of core cooling based on Fission Product Barrier monitoring.

BASIS:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all those functions necessary to maintain hot shutdown will lead to loss of fuel clad, RCS, and containment.

The Standby Shutdown Facility (SSF) is capable of providing the necessary functions to maintain Mode 3 (Hot Standby) condition for up to 72 hours. No fission product barrier degradation would be expected if the SSF is functioning as intended.

<u>Reference</u>
NUMARC/NESP-007, Rev. 2, 01/92, SG1

ENCLOSURE 4.5

LOSS OF POWER

GENERAL EMERGENCY

Analysis in support of the station blackout coping study indicates that the plant can cope with a station blackout for 4 hours without core damage.

The likelihood of restoring at least one emergency bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Coordinator a reasonable idea of how quickly (s)he may need to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is IMMINENT?

2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Coordinator judgment as it relates to IMMINENT Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, SG1

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Fire/Explosion Within the Plant	Fire or Explosion Affecting the operability of plant safety systems required to establish or maintain safe shutdown	HOSTILE ACTION within the Protected Area	
Confirmed Security condition or threat which indicates a potential degradation in the level of safety of the plant	HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY	HOSTILE ACTION resulting in Loss of Physical Control of the Facility
Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NOUE	Other conditions exist which in the judgment of the Emergency Director warrant declaration of an ALERT		Other conditions exist which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

UNUSUAL EVENT

1. ***Explosion or Fire Within the Plant***

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVEL: Note: Within the plant means Turbine Building, Auxiliary Building, Reactor Building, Keowee Hydro, Transformer Yard, B3T, B4T, Service Air Diesel Compressors, Keowee Hydro and associated transformers and SSF.

A. Fire within the plant not extinguished within 15 minutes of control room notification or verification of a control room alarm.

B. Unanticipated explosion within the plant resulting in visible damage to permanent structures/equipment.

- Includes steam line break and FDW line break

BASIS:

The purpose of this IC is to address the magnitude and extent of fires/explosions that may be potentially significant precursors to damage to safety systems. This excludes such items as fires within administration buildings, waste-basket fires, and other small fires of no safety consequence. **This IC applies to buildings and areas contiguous to plant vital areas containing safety equipment or other significant buildings or areas.** Verification of the alarm in this context means those actions taken in the control room to determine that the control room alarm is not spurious. **The intent of the 15-minute duration of extinguishing efforts is to size the fire and to discriminate against small fires that are readily extinguished.**

Only those explosions of sufficient force to damage permanent structures or equipment within the plant and **Keowee Hydro** should be considered. As used here, an explosion is a rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment, that potentially imparts significant energy to near-by structures and materials. A high energy line break (e.g., Main Steam Line or Main Feedwater Line, Heater Drain Line, etc.) would satisfy this EAL **IF** no additional damage is done to ECCS (safety related systems) equipment/components. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the explosion with reports of evidence of damage (e.g., deformation, scorching) is sufficient for declaration. The Emergency Coordinator also needs to consider any security aspects of the explosion, if applicable.

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

UNUSUAL EVENT

Escalation to a higher emergency class is by, "Fire/Explosion Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown".

Reference
NUMARC/NESP-007, Rev. 2, 01/92, HU2

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

UNUSUAL EVENT

2. CONFIRMED SECURITY CONDITION or THREAT which indicates a potential degradation in the level of Safety of the plant.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the security shift supervisor.

B. A credible site-specific security threat notification.

C. A validated notification from NRC providing information of an aircraft threat.

BASIS:

NOTE: Timely and accurate communication between Security Shift Supervisor and the control room is crucial in the implementation of effective Security EALs.

Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under 4.6.A.2, 4.6.S.1, and 4.6.G.1

A higher initial classification could be made based upon the nature and timing of the threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification in accordance with the Safeguards Contingency Plan and Emergency Plans.

EAL A

Reference is made to site specific security shift supervision because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Safeguards Contingency Plan.

This threshold is based on site specific security plans. Site specific Safeguards Contingency Plans are based on guidance provided by NEI 03-12.

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

UNUSUAL EVENT

EAL B

This threshold is included to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the Notification of an Unusual Event.

The determination of "credible" is made through use of information found in the site specific Safeguards Contingency Plan.

EAL C

The intent of this EAL is to ensure that notifications for the aircraft threat are made in a timely manner and that OROs and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

This EAL is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Unusual Event.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Escalation to Alert emergency classification level would be via 4.6.A.2 would be appropriate if the threat involves an airliner within 30 minutes of the plant.

Reference

NEI 99-01, Rev. 5, 02/2008, HU4

Frequently asked questions (FAQs) generated by users and developers during conversion from previous classifications schemes to NEI 99-01, Revision 4

Security EALs with the Hostile Action changes endorsed by the NRC in RIS 2006-12 on July 19, 2006 Enhanced guidance related to Security EALs to ensure consistency with NEI 03-12.

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

UNUSUAL EVENT

3. Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NOUE.

OPERATING MODE APPLICABILITY: ALL

A. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring off-site response or monitoring are expected unless further degradation of safety systems occurs.

BASIS

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the NOUE emergency classification level.

Reference

NEI 99-01, Rev. 5, 02/2008, HU5

Frequently asked questions (FAQs) generated by users and developers during conversion from previous classifications schemes to NEI 99-01, Revision 4

Security EALs with the Hostile Action changes endorsed by the NRC in RIS 2006-12 on July 19, 2006

Enhanced guidance related to Security EALs to ensure consistency with NEI 03-12.

<u>ENCLOSURE 4.6</u>

<u>FIRE/EXPLOSIONS AND SECURITY EVENTS</u>

ALERT

1. Fire or Explosion Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVEL: Note: Only one train of a system needs to be affected or damaged in order to satisfy this condition.

The following conditions exist:

A. Fire or explosion **<u>AND ONE OF THE FOLLOWING:</u>**

Affected safety-related system parameter indications show degraded performance

<u>OR</u>

Plant personnel report visible damage to permanent structures or equipment required for safe shutdown of the unit.

BASIS:

With regard to explosions, only those explosions of sufficient force to damage permanent structures or equipment required for safe operation of the plant should be considered. As used here, an explosion is a rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment, that potentially imparts significant energy to near-by structures and materials. A fire is combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flames is preferred but is NOT required if large quantities of smoke and heat are observed.

The key to classifying fires/explosions as an Alert is the damage as a result of the incident. The fact that safety-related equipment required for safe shutdown of the unit has been affected or damaged as a result of the fire/explosion is the driving force for declaring the Alert. **It is important to note that this EAL addresses a fire/explosion and not just the degradation of a safety system. The reference to damage of the systems is used to identify the magnitude of the fire/explosion and to discrimate against minor fires/explosions.**

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

ALERT

Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels/Radiological Effluent, or Emergency Coordinator Judgement ICs.

Reference

NUMARC/NESP-007, Rev. 2, 01/92

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

ALERT

2. HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVEL: (A or B)

A. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLED AREA as reported by the Security Shift Supervisor.

B. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.

BASIS:

Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

These EALs address the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering).

EAL A

This EAL addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the OCA.

Note that this EAL is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes ISFSI's that may be outside the PROTECTED AREA but still within the OWNER CONTROLLED AREA.

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

ALERT

EAL B

This EAL addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time.

The intent of this EAL is to ensure that notifications for the airliner attack threat are made in a timely manner and that OROs and plant personnel are at a state of heightened awareness regarding the credible threat. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.

This EAL is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

ALERT

Reference

NEI 99-01, Rev. 5, 02/2008, HA4

Frequently asked questions (FAQs) generated by users and developers during conversion from previous classifications schemes to NEI 99-01, Revision 4

Security EALs with the Hostile Action changes endorsed by the NRC in RIS 2006-12 on July 19, 2006

Enhanced guidance related to Security EALs to ensure consistency with NEI 03-12.

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

ALERT

3. Other conditions exist which in the judgment of the Emergency Director warrant declaration of an ALERT.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVEL:

A. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

BASIS:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency classification level.

Reference

NEI 99-01, Rev. 5, 02/2008, HA6

Frequently asked questions (FAQs) generated by users and developers during conversion from previous classifications schemes to NEI 99-01, Revision 4

Security EALs with the Hostile Action changes endorsed by the NRC in RIS 2006-12 on July 19, 2006

Enhanced guidance related to Security EALs to ensure consistency with NEI 03-12.

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

SITE AREA EMERGENCY

1. HOSTILE ACTION within the PROTECTED AREA.

OPERATING MODE APPLICABILITY: **ALL**

EMERGENCY ACTION LEVELS:

A. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (Security Shift Supervision).

BASIS

This condition represents an escalated threat to plant safety above that contained in the Alert in that a HOSTILE FORCE has progressed from the OWNER CONTROLLED AREA to the PROTECTED AREA.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires ORO readiness and preparation for the implementation of protective measures.

This EAL addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the PROTECTED AREA. Those events are adequately addressed by other EALs.

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

SITE AREA EMERGENCY

Reference

NEI 99-01, Rev. 5, 02/2008, HS4

Frequently asked questions (FAQs) generated by users and developers during conversion from previous classifications schemes to NEI 99-01, Revision 4

Security EALs with the Hostile Action changes endorsed by the NRC in RIS 2006-12 on July 19, 2006

Enhanced guidance related to Security EALs to ensure consistency with NEI 03-12.

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

SITE AREA EMERGENCY

2. Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts: (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

BASIS:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for Site Area Emergency.

Reference:

NEI 99-01, Rev. 5, 02/2008, HS3

Frequently asked questions (FAQs) generated by users and developers during conversion from previous classifications schemes to NEI 99-01, Revision 4

Security EALs with the Hostile Action changes endorsed by the NRC in RIS 2006-12 on July 19, 2006

Enhanced guidance related to Security EALs to ensure consistency with NEI 03-12.

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

GENERAL EMERGENCY

1. HOSTILE ACTION resulting in loss of physical control of the facility.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS: (A or B)

A A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.

B. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.

Basis:

EAL A

This EAL encompasses conditions under which a HOSTILE ACTION has resulted in a loss of physical control of VITAL AREAS (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location.

Typically, these safety functions are reactivity control (ability to shut down the reactor and keep it shutdown) RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink).

Loss of physical control of the control room or remote shutdown capability alone may not prevent the ability to maintain safety functions per se. Design of the remote shutdown capability and the location of the transfer switches should be taken into account. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions.

If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the threshold is not met.

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

GENERAL EMERGENCY

EAL B

This EAL addresses failure of spent fuel cooling systems as a result of HOSTILE ACTION if IMMINENT fuel damage is likely such as when a freshly off-loaded reactor core is in the spent fuel pool.

Reference:

NEI 99-01, Rev. 5, 02/2008, HG1

Frequently asked questions (FAQs) generated by users and developers during conversion from previous classifications schemes to NEI 99-01, Revision 4

Security EALs with the Hostile Action changes endorsed by the NRC in RIS 2006-12 on July 19, 2006

Enhanced guidance related to Security EALs to ensure consistency with NEI 03-12.

ENCLOSURE 4.6

FIRE/EXPLOSIONS AND SECURITY EVENTS

GENERAL EMERGENCY

2. Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.

Operating Mode Applicability: All

EMERGENCY ACTION LEVEL:

A. Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels off-site for more than the immediate site area.

BASIS:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for General Emergency.

Reference

NEI 99-01, Rev. 5, 02/2008, HG2

Frequently asked questions (FAQs) generated by users and developers during conversion from previous classifications schemes to NEI 99-01, Revision 4

Security EALs with the Hostile Action changes endorsed by the NRC in RIS 2006-12 on July 19, 2006

Enhanced guidance related to Security EALs to ensure consistency with NEI 03-12.

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Natural and Destructive Phenomena Affecting the Protected Area Natural and Destructive Phenomena Affecting Keowee Hydro Condition B Natural and destructive phenomena affecting Jocassee Hydro Condition B.	Natural and Destructive Phenomena Affecting the Plant Vital Area	Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established	Other Conditions Existing Which in the Judgement of the Emergency Coordinator Warrant Declaration of General Emergency
Release of Toxic or Flammable Gases Deemed Detrimental to Safe Operation of the Plant	Release of Toxic or Flammable Gases Jeopardizes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown Turbine Building Flood	Keowee Hydro Dam Failure Other Conditions Existing Which in the Judgement of the Emergency Coordinator Warrant Declaration of Site Area Emergency	
Other Conditions Existing Which in the Judgement of the Emergency Coordinator Warrant Declaration of an Unusual Event Natural and Destructive Phenomena Affecting Keowee Hydro	Control Room Evacuation Has Been Initiated Other Conditions Existing Which in the Judgement of the Emergency Coordinator Warrant Declaration of an Alert		

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

1. Natural and Destructive Phenomena Affecting the Protected Area.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. Tremor felt and valid alarm on the "strong motion accelerograph".

B. Tornado striking within protected area boundary.

C. Vehicle crash into plant structures or systems within protected area boundary.

D. Turbine failure resulting in casing penetration or damage to turbine or generator seals.

BASIS:

The protected area boundary is typically that part within the security isolation zone and is defined in the site security plan.

EAL 1. Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate. Strong motion accelerograph will begin to record at .01g. As defined in the EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, a "felt earthquake" is:

> ***An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) valid alarm on seismic instrumentation occurs.***

EAL 2. A tornado striking (touching down) within the protected boundary may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. If such damage is confirmed visually or by other in-plant indications, the event may be escalated to Alert.

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

EAL 3 Addresses such items as a car, truck, plane, or helicopter crash, or train crash that may potentially damage plant structures containing functions and systems required for safe shutdown of the plant. If the crash is confirmed to affect a plant area containing equipment required for safe shutdown of the unit, the event may be escalated to Alert.

EAL 4 Addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual fires and flammable gas build up are appropriately classified via other EALs. This EAL is consistent with the definition of an Unusual Event while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of the emergency classification is based on potential damage done by the missiles generated by the failure.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, HU1

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

2. Natural and Destructive Phenomena Affecting Keowee Hydro Condition B.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. Reservoir elevation greater than or equal to 805.0 feet with all spillway gates open and the lake elevation continues to rise.

B. Seepage readings increase or decrease greatly or seepage water is carrying a significant amount of soil particulates.

C. New area of seepage or wetness, with large amounts of seepage water observed on dam, dam toe, or the abutments.

D. A slide or other movements of the dam or abutments which could develop into a failure.

E. Developing failure involving the powerhouse or appurtenant structures and the operator believes the safety of the structure is questionable.

F. Emergency Coordinator judgment

BASIS:

Keowee Hydro is the emergency AC power source for the Oconee Nuclear Station and is covered by the site emergency plan. The conditions cited above are considered to be situations where dam failure may develop. The potentially hazardous situation may allow days or weeks for mitigative actions to prevent failure.

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

3. Natural and Destructive Phenomena Affecting Jocassee Hydro Condition B.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. Condition B has been declared for Jocassee

BASIS:

Jocassee Hydro is located upstream of the Oconee Nuclear Station. The mitigation strategies for a Condition B for the Jocassee Dam includes shutdown of all operating Oconee Nuclear units and relocation and installation of other equipment in anticipation of the Condition B escalating to a Condition A.

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

4. **Release of Toxic or Flammable Gases Deemed Detrimental to Safe Operation of the Plant.**

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. Detection of toxic or flammable gases that could enter within the site area boundary in amounts that can affect normal operation of the plant.

B. Report by Local, County or State Officials for potential evacuation of site personnel based on offsite event.

BASIS:

This IC is based on releases in concentrations within the site boundary that will affect the health of plant personnel or the safe operation of the plant with the plant being within the evacuation area of an offsite event (i.e., tanker truck accident releasing toxic gases, etc.) The evacuation area is as determined from the DOT Evacuation Tables for Selected Hazardous Materials in the DOT Emergency Response Guide for Hazardous Materials.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, HU3

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

UNUSUAL EVENT

5. **Other Conditions Existing Which in the Judgment of the Emergency Coordinator Warrant Declaration of an Unusual Event.**

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVEL:

Other conditions exist which in the judgement of the Emergency Coordinator indicate a potential degradation of the level of safety of the plant.

BASIS:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the Unusual Event emergency class.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, HU5

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

1. Natural and Destructive Phenomena Affecting the Plant Vital Area.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. Tremor felt and seismic trigger actuates (.05g)

Note: Only one train of a safety related system needs to be affected or damaged in order to satisfy these conditions.

B. Tornado, high winds, missiles resulting from turbine failure, vehicle crashes, or other catastrophic events **AND** one of the following:

Plant personnel report visible damage to permanent structures or equipment required for safe shutdown of the unit

OR

Affected safety related system parameter indications show degraded performance

BASIS:

EAL 1 Based on the FSAR design basis. Seismic events of this magnitude can cause damage to safety functions.

EAL 2 is intended to address the threat to safety related structures or equipment from uncontrollable and possibly catastrophic events. Damage to safety-related equipment and or structures housing safety-related equipment caused by natural phenomena after striking the site is the key point of this EAL. Only one train of a safety-related system needs to be affected or damaged in order to satisfy this condition. This EAL is, therefore, consistent with the definition of an ALERT in that if events have damaged areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant.

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

Structures/equipment which provide safety functions are designed to withstand sustained wind force of 95mph. These structures are designed to withstand external wind forces resulting from a tornado having a velocity of 300mph. Because high winds may disable the meteorological instrumentation well before the design basis speed is reached, the meteorological tower should not be used for assessment of tornado winds for emergency classification. For tornados, damage would be the prima facie evidence of winds exceeding design basis.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, HA1

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

2. Release of Toxic or Flammable Gases Jeopardizes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Mode 5 (Cold Shutdown).

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVELS:

A. Report or detection of toxic gases in concentrations that will be life threatening to plant personnel.

B. Report or detection of flammable gases in concentrations that will affect the safe operation of the plant.

Reactor Building
Auxiliary Building
Turbine Building
Control Room

BASIS:

EAL 1 is based on toxic gases that have entered a plant structure that are life-threatening to plant personnel. This EAL applies to structures required to maintain safe operations or to establish or maintain cold shutdown. It is appropriate that increased monitoring be done to ascertain whether consequential damage has occurred. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels/Radioactive Effluent, or Emergency Coordinator Judgement ICs.

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

EAL 2 is based on the detection of flammable gases in areas containing equipment required for safe shutdown of the unit. It is appropriate that increased monitoring be done to ascertain whether consequential damage has occurred. Escalation to a higher emergency class, if appropriate, will be based on System Malfunction, Fission Product Barrier Degradation, Abnormal Rad Levels/Radioactive Effluent, or Emergency Coordinator Judgement ICs.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, HA3

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

3. TURBINE BUILDING FLOOD

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVEL:

A. Turbine building flood requiring use of AP/1,2,3/A/1700/010, Turbine Building Flood.

BASIS:

This initiating condition is discussed in the Oconee Probabilistic Risk Assessment report. A flood caused by the rupture of the Jocassee Dam could flood the turbine building basement which could disable the main feedwater pumps and the turbine and motor driven emergency feedwater pumps. Also, rupture of some portions of the condenser intake piping could result in a flood in the turbine building basement. Water tight doors have been provided to prevent the water from seeping into the auxiliary building. This scenario assumes that the Standby Shutdown Facility (SSF) would be available to provide water to the steam generators. Escalation of the event to a higher category would be based on the ability to maintain core cooling or shutdown functions.

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

4. Control Room Evacuation Has Been Initiated.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVEL:

A. Evacuation of control room **AND** one of the following:

Plant control is established from the Aux SD panel or the SSF

OR

Plant control is being established from the Aux SD panel or the SSF

BASIS:

The auxiliary shutdown panel will allow operators to use turbine bypass valves to maintain RCS temperature, one HPI pump for RCS inventory control, pressurizer heaters to maintain RCS pressure and control of the feedwater startup valves but not control over the feedwater pumps.

The standby shutdown facility can maintain hot shutdown by using auxiliary service water to the steam generators for primary heat removal and also to provide makeup to the reactor coolant system. The SSF is only used under extreme conditions since it may involve pumping lake water into the steam generators for heat removed purposes.

With the control room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other Emergency Operations Facility is necessary. Inability to establish plant control from outside the control room, as evidenced by the inability to maintain RCS or SG inventories, will escalate this event to a Site Area Emergency.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, HA5

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

ALERT

5. Other Conditions Existing Which in the Judgement of the Emergency Coordinator Warrant Declaration of an Alert.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVEL:

A. Other conditions exist which in the Judgement of the Emergency Coordinator indicate that plant safety systems may be degraded **AND** that increased monitoring of plant functions is warranted.

BASIS:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the Alert emergency class.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, HA6

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

SITE AREA EMERGENCY

1. Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVEL:

The following conditions exist:

A. Control room evacuation has been initiated

AND

Control of the plant cannot be established from the Aux SD panel or the SSF within 15 minutes.

BASIS:

The timely transfer of control to alternate control areas has not been accomplished. This failure to transfer control would be evidenced by deteriorating reactor coolant system or steam generator parameters. For most conditions RCP seal LOCAs or steam generator dryout would be indications of failure to accomplish the transfer in the necessary time.

Escalation of this event, if appropriate, would be by Fission Product Barrier Degradation, Abnormal Rad Levels/Radiological Effluent, or Emergency Coordinator Judgement ICs

Reference
NUMARC/NESP-007, Rev. 2, 01/92, HS2

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

SITE AREA EMERGENCY

2. Keowee Hydro Dam Failure

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVEL:

A. Imminent/actual dam failure exists involving any of the following:
Keowee Hydro Dam
Little River Dam
Dikes A,B,C,D
Intake Canal Dike
Jocassee Dam - Condition A

BASIS:

The Keowee Hydro Dam project includes the Keowee Hydro Dam, Little River Dam and Dikes A, B, C, D, and the Intake Canal Dike. Dam failure of any portion of the Keowee Hydro Dam would result in loss of the emergency AC power supply AND the potential to lose the ultimate heat sink source. Some flooding of the site may result. Evaluation of the plant status following failure of the dam would determine the need to escalate to a General Emergency. Failure of the Jocassee Dam has the potential to result in the failure of the Keowee Hydro Project Dams/Dikes.

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

SITE AREA EMERGENCY

3. Other Conditions Existing Which in the Judgement of the Emergency Coordinator Warrant Declaration of Site Area Emergency.

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVEL:

A. Other conditions exist which in the Judgement of the Emergency Coordinator indicate actual or likely major failures of plant functions needed for protection of the public.

BASIS:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator/EOF Director to fall under the emergency class description for Site Area Emergency.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, HS3

ENCLOSURE 4.7

NATURAL DISASTERS, HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

GENERAL EMERGENCY

1. **Other Conditions Existing Which in the Judgement of the Emergency Coordinator Warrant Declaration of General Emergency.**

OPERATING MODE APPLICABILITY: ALL

EMERGENCY ACTION LEVEL:

A. Other conditions exist which in the Judgement of the Emergency Coordinator/ EOF DIRECTOR indicate:

(1) Actual or imminent substantial core degradation with potential for loss of containment

OR

(2) Potential for uncontrolled radionuclide release that would result in a dose projection at the site boundary greater than 1000 mRem TEDE or 5000 mRem CDE Adult Thyroid.

BASIS:

This EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator/EOF Director to fall under the General Emergency class.

Releases (if made) can reasonably be expected to exceed EPA PAG levels outside the site boundary.

Reference
NUMARC/NESP-007, Rev. 2, 01/92, HG2

ENCLOSURE 4.8

Radiation Monitor Readings for Emergency Classification

All RIA values are considered GREATER THAN or EQUAL TO

HOURS SINCE REACTOR TRIPPED	RIA 57 R/hr		RIA 58 R/hr*	
	Site Area Emergency	General Emergency	Site Area Emergency	General Emergency
0.0 - < 0.5	5.9E+003	5.9E+004	2.6E+003	2.6E+004
0.5 - < 1.0	2.6E+003	2.6E+004	1.1E+003	1.1E+004
1.0 - < 1.5	1.9E+003	1.9E+004	8.6E+002	8.6E+003
1.5 - < 2.0	1.9E+003	1.9E+004	8.5E+002	8.5E+003
2.0 - < 2.5	1.4E+003	1.4E+004	6.3E+002	6.3E+003
2.5 - < 3.0	1.2E+003	1.2E+004	5.7E+002	5.7E+003
3.0 - < 3.5	1.1E+003	1.1E+004	5.2E+002	5.2E+003
3.5 - < 4.0	1.0E+003	1.0E+004	4.8E+002	4.8E+003
4.0 - < 8.0	1.0E+003	1.0E+004	4.4E+002	4.4E+003

*** RIA 58 is partially shielded**

Assumptions used for calculation of high range in-containment monitors RIA 57 and 58:

1. Average annual meteorology (7.308 E^{-6} sec/m^3)
2. Design basis leakage (5.6 E^6 ml/hr)
3. One hour release duration
4. *General Emergency* PAGs are 1 rem TEDE and 5 rem CDE; *Site Area Emergency* determination is based on 10% of the *General Emergency* PAGs
5. Calculations for monitor readings are based on CDE because thyroid dose is limiting
6. No credit is taken for filtration
7. LOCA conditions are limiting and provide the more conservative reading

I. Accident Assessment

To assure the adequacy of methods, systems and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition.

I.1 Emergency Action Level Procedures

Implementing procedures to the Oconee Nuclear Station Emergency Plan have been developed. These procedures have been developed by many sections of the station. The Oconee Nuclear Station Implementing Procedures make up Volumes B and C of the station emergency plan. The Emergency Classification procedure (RP/0/B/1000/001) identifies plant parameters that can be used to determine emergency situations that require activation of the station emergency plan. NUMARC/NESP-007 (Rev. 2) which was approved by the NRC in Rev. 3 of Regulatory Guide 1.101 and subsequent guidance provided in NRC Bulletin 2005-02, the NEI guidance as endorsed in RIS 2006-12 and to support implementation of NEI 03-12 has been used as guidance. See BASIS document Section D.

I.2 Onsite Capability and Resources to Provide Initial Values and Continuing Assessment

Post Accident Sampling -

The NRC issued Amendments No. 346 (Renewed License No. DPR-38), No. 348 (Renewed License No. DPR-47), and No. 347 (Renewed License No. DPR-55) on 07/12/05. These amendments, effective 01/08/06, delete Technical Specification Section 5.5.4, Post Accident Sampling for Oconee Nuclear Site Units 1, 2, and 3 and thereby eliminate the requirements to have and maintain Post Accident Sampling Systems - PASS (PALS/PAGS). Consistent with the requirements of the NRC safety evaluation, contingency plans for obtaining samples have been developed.

Procedures have been developed for taking and analyzing post accident reactor coolant samples using either the normal sample points or the existing PALS sample panels. Containment atmosphere samples are no longer required; however, procedures are in place for surveying the containment building wall as well as sampling the environment and using these values to develop off site dose projections and provide appropriate protective action recommendations for the public.

Radiation and effluent monitors

Radiation and effluent monitors are indexed in Figure H-5. The chart shows location, range, radiation detected.

Containment High Range Radiation Monitor
Duke Energy has designed a system for monitoring containment high range radiation. 1, 2, 3 RIA-57 and 58 are the post-accident high range containment monitors. RIA-57 is located in a penetration in the East Penetration Room. RIA-58 is located in a penetration in the West Penetration Room. The monitors are coaxial ion chambers with a range of 1 to 10E8 Rad/hr which corresponds to an activity of 1.11E0 µCi/ml to 1.11E8 µCi/ml at the time of trip/incident.

In- Plant Iodine Instrumentation
The Oconee Nuclear Station has developed Procedure HP/0/B/1009/009 for quantifying high level gaseous radioactivity releases during accident conditions. The purpose of the procedure is to determine quantitative release of radioiodines and particulates for dose calculation and assessment.

Failed Fuel Determination

(1) The attached Figures I-1, I-2, I-3, and I-4 provide the technical basis for estimating failed fuel for three conditions: non-overheating, fuel overheating without fuel melt, and overheating with fuel melt, respectively.

(2) The NON-OVERHEATING CONDITION METHODOLOGY for assessing failed fuel is based on steady-state iodine radionuclides in the reactor coolant system. This methodology is judged to provide a significant improvement in accuracy over previous NON-OVERHEATING CONDITION methods employing a single escape coefficient. The reason being the new methods explicitly models the production, decay, and release of radionuclides to the coolant as a function of measured iodine ratio.

The methods CAN ONLY PROVIDE THE best estimate analysis and are not intended for making conservative or licensing related calculations. These methods are benchmarked to long term steady-state iodine behavior, typically reached near mid to end of cycle. Therefore, leaker estimates (percent failed fuel) will vary substantially if based on other than steady-state conditions.

Radioisotope inventories predicted by LOR2 Computer Program are used to compare release isotope quantities to expected core inventories for the fuel overheating without fuel melt and overheating with fuel melt conditions. In order to determine a conservative core inventory for Oconee, three LOR2 computer runs were made. All three runs assumed an enrichment of 3.3%.

Each run represents a different burn-up region of the core. (i.e., one run assumes fuel used for 3 cycles, another run assumes fuel used for 2 cycles and the last run assumes fuel used for 1 cycle.) Each region assumed 59 assemblies. Figure I-5, page 1 of 2, gives activity level for one fuel assembly for each region. Figure I-5, page 2 of 2, gives total activity in the core and compares these values to UFSAR values. Most of the core values are close to UFSAR values except for XE-133 and XE-135. It is possible that this difference is the result of the higher enrichment value used in the LOR2 runs.

(3) Figures I-6 and I-7 provide the technical basis for an estimate of failed fuel from readings from area monitors (without fuel melt and with fuel melt, respectively).

(4) Figure I-8 provides the technical basis for an estimate of failed fuel from readings of containment building hydrogen analyzers.

(5) Figure I-9 provides calculations for decay correction in the event it is not available from analytical instrumentation.

I.3 Method for Determining Release Source Term

I.3.a Source Term of Releases of Radioactive Material within Plant Systems

Operations (Control Room Personnel) will use Enclosure 4.8 & 4.9 of RP/0/B/1000/001 to determine if radiation monitor readings will require classification. This enclosure is a simplified predetermined dose calculation for vent and in-containment radiation monitors. Operations can also get offsite dose projections from on-shift Radiation Protection technicians using procedure HP/0/B/1009/022. HP/0/B/1009/022 uses release paths of unit vents and the main steam relief valves. Assumptions for the calculations are based on the following:

1. Annual average meteorology for ground-level release points (7.308 E-6 sec/m^3) which is used for the reactor building and is in the ODCM. Annual average meteorology for semi-elevated release points 1.672E-6 sec/m^3 is used for the vent and is also in the ODCM.

2. Design basis leakage (5.6 E6 ml/hr) and/or daily average vent flow rate of 65,000 cfm.

3. One hour release duration

4. Calculations for reactor building monitor readings are based on CDE because thyroid dose is more limiting for this pathway. Calculations for vent monitor readings are based on whole body dose because whole body dose is more limiting for this pathway.

5. Offsite Protective Actions Guides are 1 rem Total Effective Dose Equivalent and 5 rem Committed Dose Equivalent (thyroid) for a General Emergency. Site Area Emergency levels are one-tenth the General Emergency PAGs.

6. LOCA conditions are limiting for calculating in-containment high range monitors readings for site area and general emergency conditions.

7. Core melt conditions are limiting for calculating vent monitor radiation monitor readings for site area and general emergency conditions.

I.3.b Magnitude of the Release of Radioactive Materials

Procedure HP/0/B/1009/18, SH/0/B/2005/001 and/or HP/0/B/1009/022 determines the magnitude of the release of radioactive materials based on plant system parameters and effluent monitors (vent release).

I.4 Dose Calculation Methodology

HP/0/B/1009/018 and SH/0/B/2005/001 establish the relationship between effluent monitor readings or reactor building dose rate readings and onsite/offsite doses for various meteorological conditions.

HP/0/B/1009/022 provides guidance for on shift personnel to perform initial dose assessment using a computer based tool.

I.5 Meteorological Information Availability

Meteorological information will be available to the Charlotte Emergency Operations Facility, the Technical Support Center, and the Control Room through the automated plant data system. Meteorological data averaged over a period of 15 minutes, will be available to the NRC through the ETS phone, by direct telephone communications with the individual responsible for making offsite dose assessments at the Emergency Operations Facility or through the NRC Emergency Response Data System.

Meteorological information will also be given to both County Emergency Operations Centers, and the State of South Carolina, during follow-up messages.

I.6 Release Rates/Projected Doses for Offscale Instrumentation Situations

HP/0/B/1009/018 and SH/0/B/2005/001 are procedures that can be used to make offsite dose projections and/or protective action recommendations should instrumentation used for assessment indicate offscale or are inoperable.

I.7 & I.8 Offsite Field Monitoring-Emergency Planning Zone

Field teams have been organized by the Oconee Nuclear Station under the direction of the Field Monitoring Coordinator located in the Emergency Operations Facility. These teams are comprised of a RP Technician and a Driver. Procedures SH/0/B/2005/002 and HP/0/B/1009/026 describe predetermined sampling locations, sampling and monitoring equipment to be used, location of TLD's and air samplers and directions for taking Potassium Iodide Tablets.

I.9 Detect and Measure Radioiodine Concentration in the EPZ

Oconee Nuclear Station shall use appropriate instrumentation to measure radioactivity in counts per minute (CPM) and dose rates in mRad/hr. Air samples (taken with a Portable Air Sampler equipped with appropriate cartridge) shall be measured by a portable iodine analysis system.

Interference from the presence of noble gas and background radiation shall not decrease the minimum detectable activity of 1.0 E-7 uCi/cc (I-131) under field conditions.

Samples taken by the offsite monitoring teams will be evaluated further by one of the available laboratory facilities described in H.6.C of this Plan as necessary.

I.10 Relationship Between Contamination Levels and Integrated Dose/Dose Rates

Duke Energy Company has developed a means for relating the various measured parameters (e.g. contamination levels, air and water) and gross radioactivity levels.

I.11 Plume Tracking

The states of North Carolina, South Carolina and Georgia have arrangements to locate and track an airborne plume of radioactive materials. Duke Energy Company will have monitoring teams in the field, fixed TLD sites, and the capability for airborne monitoring to assist in plume tracking.

See State of North Carolina, FNF Plans
See State of South Carolina, FNF Plans
See State of Georgia, FNF Plans

FIGURE I-1

DUKE ENERGY COMPANY
OCONEE NUCLEAR STATION
ACCIDENT ASSUMPTIONS

DBA assumes draft NUREG 1465 release of fission products to the containment atmosphere:

(1) 100% of all core noble gas activity.

(2) 40% of all core iodine activity.

(3) Various quantities of particulate activity.

Loss of reactor coolant assumes the release of one reactor coolant volume with noble gas and iodine activity associated with operation at 100% power with 1% fuel failure before the release.

Gap activity release assumes that there is cladding failure sufficient to release all fission products in the gas gap of the fuel pins to the containment atmosphere. Assumed is loss of 5% of all core noble gas activity, 5% of all core iodine activity, and 5% of cesium particulate activity to the containment atmosphere.

The maximum allowable containment leakage rate following the accident is expressed in percent of the containment air weight per day.

Regulatory Guide 1.4 requires that we assume the design leak rate (Technical Specifications 5.5.2) the first 24 hours and half the design leak rate for the rest of the accident.

For Oconee these values are:

(a)	0.25%/day	for	0-24 hours
(b)	0.125%/day	for	24 hours - 30 days

The 0.25%/day is the Tech. Spec. leak rate at the peak calculated containment internal pressure, 59 psig, for the design basis LOCA.

Assumptions used in determining the contribution to the total dose from ECCS leakage are:

(a) 7520 cc/hr leakage from the pump seals and valves of the ECCS in the auxiliary building.

(b) An iodine partition factor of 0.1 is used to determine the amount of iodine released to the auxiliary building atmosphere.

(c) All activity released to the auxiliary building is released to the atmosphere with no filtering.

Most Oconee penetrations through the containment are located in the penetration room. This room has its own ventilation system which draws a negative pressure on the room. The air drawn from the penetration room passes through charcoal filters and is exhausted through the unit vent. Bypass leakage is the fraction of the total containment that bypasses the penetration room and escapes to the atmosphere unfiltered. Some examples of potential bypass leakage paths are:

(1) Leakage around the equipment hatch seals.

(2) Leakage through isolation valves that do not seal properly.

(3) Leakage through microscopic holes or cracks in the containment wall.

At Oconee the containment bypass leakage is 50% of the total containment leakage.

Tech. Spec. 5.5.2 requires that during the containment leak rate test, if the containment leakage is greater that 50% of the design leakage rate, local leak rate tests must be performed. These tests must verify that any leakage greater than 50% of the design leakage is going into the penetration room. This only verifies that the maximum leakage bypassing the penetration room is 50% of the containment leakage. It does not give the actual bypass leakage.

Dose contributions are as follows:

(a) Bypass leakage contributes approximately 84% of the total thyroid dose.

(b) ECCS leakage contributes approximately 1% of the total thyroid dose.

(c) Penetration room exhaust contributes approximately 15% of the total thyroid dose.

FIGURE I-2

TECHNICAL BASIS FOR ESTIMATION OF FAILED FUEL
NON-OVERHEATING CONDITION

A. Assumptions

1. All Iodine and Xenon isotopes are at equilibrium.

2. All Iodine isotopes in the RCS pass through a 90% efficient demineralizer at the rate of one coolant volume per day.

3. There is no plate out of Iodine in the RCS.

4. The noble gases are equally mixed throughout the RCS and consideration is not given to noble gases that may be in the letdown storage tank or pressurizer.

5. The reactor is operating at 100% power - 2568 MWT or at any steady-state power level with Steps 1 through 4 applicable.

B. Two Region Model Theory

The two region model assumes a single escape coefficient for the release from the fuel directly into the coolant through the defect site. The model first solves for the dynamic iodine concentrations in the fuel pellet, then through the use of an escape coefficient, solves for the steady-state release into the coolant. Once into the coolant, the methodology also calculates the effects of radioactive decay and coolant purification on the measured iodine concentrations. The following is a delineation of the dynamic solution of the above phenomena, including a simplification for steady-state conditions where appropriate.

I. In-Fuel Concentration

The rate of change of the number of atoms is given by:

$$\frac{dN}{dt}{}^{f}_{t} = \text{(GENERATION RATE) - (DECAY RATE) - (RELEASE RATE)}$$

$$\frac{dN}{dt}{}^{f}_{t} = \dot{F}\,\overline{Y} - N^{f}_{t}\,\lambda - N^{f}_{t}\,\upsilon \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots(1)$$

Where N^{f}_{t} is the dynamic number of atoms of a short-lived isotope in a single fuel rod (atoms/rod)

t is the time (sec)

$\dot{F}$ is the rod volumetric total fission rate, which is a constant for the limits of integration (fiss/sec)

$\overline{y}$ is the effective fission product yield (atoms/fiss)

λ is the decay constant for the isotope (decay probability fraction per atom per second)

υ is the two-region model escape-rate coefficient from the fuel to the coolant for the isotope (escape probability fraction per atom per second).

SOLVING FOR TIME EQUALS TO ZERO YIELDS:

$$N^{f}_{t} = {}^{f}_{o}\,e^{-(\lambda+\upsilon)t} + \frac{\dot{F}\,\overline{y}}{(\lambda+\upsilon)} = (1 - e^{-(\lambda+\upsilon)t})$$

II. In-Coolant Concentration

The rate of change of the number of atoms of the isotope in the reactor coolant system is given by:

$$\frac{dN}{dt}{}^{c}_{t} = N^{f}_{t}\,(\upsilon) - N^{c}_{t}\,(\lambda) - N^{c}_{t}\,K = \frac{dN}{dt}{}^{c}_{t} = \text{(Release Rate) - (Decay Rate) - (Purification Rate)} \qquad (3)$$

where N^{c}_{t} is the dynamic number of atoms of a short-lived isotope in the reactor coolant system (atoms)

K is the purification constant associated with the letdown system and is equal to the system mass flow rate divided by the total non-stagnant coolant mass (purification probability fraction per atom per second)

SOLVING FOR TIME EQUALS TO ZERO YIELDS:

$$N^{c}_{t} = \frac{\dot{F}\,\bar{y}\,\upsilon}{(\lambda+\upsilon)\,(\lambda+K)}\left(1 - \frac{1}{(K-\upsilon)}\left[(\lambda+K)e^{-(\lambda+\upsilon)t} - (\lambda+\upsilon)e^{-(\lambda+K)t}\right]\right)$$

$$+ \frac{N^{f}_{o}\,\upsilon}{(K-\upsilon)}\left[e^{-(\lambda+\upsilon)t} - e^{-(\lambda+K)t}\right]$$

$$+ N^{c}_{o}\left[e^{-(\lambda+K)t}\right]$$

$$\text{or } N^{c}_{t} = N^{c1}_{t} + N^{c2}_{t} + N^{c3}_{t} \qquad (4)$$

In the above:

1) N^{c1}_{t} is the atoms of an isotope remaining in the coolant at time, t, from the inventory of atoms generated by fission events during the current time step (t=0 to t=t)

2) N^{c2}_{t} is the atoms of an isotope remaining in the coolant at time, t, from the inventory of atoms within the rod generated by fission events prior to the current time step; and

3) N_t^{c3} is the atoms of an isotope remaining in the coolant at time, t, from the atoms in the coolant at the beginning of the current time step.

For the event that fissioning begins at time t equals to zero, N_t^{c2} and N_t^{c3} are also equal to zero at all t. Furthermore, assuming steady-state conditons, Equation 4 reduces to:

$$N_{oo}^{c} = \frac{\dot{F}\,\bar{y}\upsilon}{(\gamma+\upsilon)(\lambda+K)} \qquad \text{(steady-state)}$$

The conventional units for measuring the concentration of atoms of a radioisotope are in terms of isotopic activity, with units of µCi/ml. A_t^c is defined as the activity associated with the concentration N_t^c

Since N_t^{c1}, N_t^{c2}, and N_t^{c3} are in units of atoms per rod in Equation 4, the following conversion is required to obtain A_t^c:

$$A_t^c = A_t^{c1} + A_t^{c2} + A_t^{c3}$$

$$A_t^c = [N_t^{c1} + N_t^{c2} + N_t^{c3}]\,\frac{(atoms)}{(rod)} \text{ x } N_r \text{ (rods) x } \frac{1}{V^c(ml)} \text{ x } \lambda \,\frac{(decay\ probability)}{(atom)(\text{sec})}$$

$$\text{x } \frac{1(\mu Ci)}{2.22x\ 10^6} \quad \frac{(decays)}{(\text{min})} \text{ x } 60\ \frac{(\text{sec})}{(\text{min})}\text{ , or}$$

$$A_t^c = [N_t^{c1} + N_t^{c2} + N_t^{c3}]\ [2.703\text{E-}5\ \lambda N_r/V^c],\ (\mu\text{Ci/ml}) \qquad (5)$$

where N_r is the number of perforated rods in the core

V^c is the non-stagnant volume of the reactor primary coolant system

And for steady-state conditions:

$$A_{\infty}^{c} = \frac{\dot{F}\,\bar{y}\,\upsilon}{(\lambda+\upsilon)(\lambda+K)} \times \frac{2.703E-5\,\lambda}{V^{c}}\, N_r \; ((\mu ci/ml) \qquad (6)$$

III. Escape Rate Methodology

This section describes the model and supporting technical basis for an escape rate coefficient model dependent on measured iodine ratio. The need for such a model is illustrated by the following two examples.

At one extreme, assume a leaker with a tight radial through wall capillary type crack, which in effect bottles up the fission products and allows very little leakage to the coolant. At the other extreme, assume a pin with a large open hydride blister, exposing the surrounding fuel directly to the coolant. Obviously, both represent only one defect, however, the latter case would release much more fission products to the coolant than the first case.

Therefore, the need exists to differentiate between various defect conditions. To do this, the concept of holdup time, and its affect on relative radioactive decay is used. The tight defect, due to the long holdup time, would shift the iodine ratio (131/133) towards the high end (>1) due to the faster 133 decay (133 half life - 20.8 hrs, 131 half life - 8.05 days). For little or no holdup times, the existing ratio would be around 0.1. This is consistent with observations during failure generation events in which the observed iodine ratio in the coolant approaches two or greater. Calculations for an intact rod (infinite holdup time) yield ratios in excess of 10 or 15.

This rational forms the basis for an iodine ratio dependent escape coefficient model. It certainly is not perfect in that a combination of defects could easily exist at any one time, but it does give an approximation as to the average condition.

Towards this end, an empirical model was developed based on a Combustion Engineering Data Base. The data consists of several operating cycles in which the coolant activities and specific leaking rods were well characterized. The model is empirical, in that the necessary escape coefficients were back calculated and plotted as a function of corrected iodine ratio. However, the ratio needs to be corrected for the decay and purification effects occurring in the primary coolant, so that a consistent and independent model (independent of letdown flow, resin bed efficiency, etc.) can be developed.

The correct or "normalized" iodine ratio is determined as follows:

The equilibrium coolant activity ratio, as determined by Equation 6, is shown here as follows:

$$AR_{\infty}^{c} = \frac{\bar{y}131\,\upsilon131\,\lambda131\,(\lambda133+\upsilon133)\,(\lambda133+K)}{\bar{y}133\,\upsilon133\,\lambda133\,(\lambda133+\upsilon131)\,(\lambda131+K)} \qquad (7)$$

Assume that:

$R = AR_{\infty}^{c}$

$\underline{\upsilon_{133}} = a$

υ_{133}

Substituting and solving for υ_{131} gives:

$$\upsilon_{131} = \frac{1 - \dfrac{R\ (\lambda 131 + K)\ \bar{y}\,133}{a\ (\lambda\,133 + K\ \bar{y}\,131}}{\dfrac{R\,(\lambda 131 + K)\,y\,\bar{1}33\ \ -1}{a\,(\lambda\,133 + K\ \bar{y}131\lambda 131\,a\lambda 131}} \qquad (8)$$

The normalized iodine ratio (R') is independent of the coolant volume and purification flow rate and is defined as:

$$R' = \frac{(\lambda_{131}\ \ +K)}{(\lambda_{133}\ \ +K)}\ (R) \qquad (9)$$

Equation 8 can be re-written as:

$$\frac{v}{v131} = 1 - \frac{R'}{a}\ \frac{\bar{y}_{133}}{\bar{y}_{131}}$$

$$\frac{R'}{a}\quad \frac{\bar{y}_{133}}{\bar{y}_{131}}\quad \frac{1}{\lambda_{131}}\quad \frac{-1}{a\lambda_{133}} \qquad (10)$$

Equation 10 gives a relationship between the iodine-131 escape rate and the normalized iodine ratio. The constant "a", which describes the relationship between the iodine-131 and iodine-133 escape rates, was derived through an analysis of 3 plant cycles in which the number of leaking rods was determined at the end of each cycle. Parametric cases were run to determine, given the known number of leaking rods, the iodine-131 and iodine-133 escape rates required to predict the equilibrium activity levels of these isotopes for the 3 plant cycles. All data were taken at 100% power and at equilibrium conditions.

The escape rate (ration υ_{131} / υ_{133} = a) was assumed to be function of the normalized iodine ratio. A curve was fit to the data resulting in:

$$\frac{\upsilon_{131}}{\upsilon_{133}} = a = \frac{R'}{.437260 + .021089R' - .013293R'^{2}} \qquad (11)$$

<u>Power Dependence of Escape Rate</u>

The kinetics of fission product migration through the fuel pellet into the rod plenum/gap is governed primarily by temperature. Since temperature is primarily a function of power, a power dependent correction factor was developed based on total rod radial power.

The power dependence of escape rate was determined by evaluating the equilibrium coolant iodine-131 activity as a function of core power level. These data were taken from periods of operation at varying power levels for 8 plant-cycles.

A power function was assumed to represent each individual data set:

$$A_{131} = CP^{n}$$

where:

A_{131} is the equilibrium level of iodine 131 (µCi/ml)

C and n are fitting constants, and

P is the core power level (%)

C varies with the plant conditions (purification flow rate, etc.) and numbers of leaking rods but, in theory, n should be constant if the data are consistent. Least-squares analyses were performed for each of the sets of data. The values of n determined from this analysis ranged from 1.8 to 5.4. A value of n = 3.6 was selected as a reasonable representation of the data.

From Equation 6, the equilibrium coolant activity level is directly proportional to power (fission rate) and the escape rate. Therefore, the escape rate must be proportional to power to the (n-1), or 2.6 power. Escape rates are therefore calculated with the following equation:

$$\upsilon = \upsilon_o (P/P_o)^{2.6} \quad \text{or } \upsilon = \upsilon_o (Pr/P_o)^{2.6} \qquad (12)$$

where:

> υ_o is the escape rate (sec^{-1}) determined at power P_o (%) (As derived from Equation #10); and
>
> P is, optionally, the core power level (%), or Pr, the product of core power level and rod or batch peaking factor (relative to core).

If a specific rod or batch peaking factor is suspected, Pr should be used since it can make an appreciable difference.

FIGURE 1-3

TECHNICAL BASIS FOR ESTIMATION OF FAILED FUEL

Nuclear Engineering uses the following calculations to determine Fuel Overheating without Fuel Melt, Utilization of Area Monitors for Overheat Without Fuel Melt, and Utilization of Area Monitors for Fuel Melt Conditions.

OSC-5283 - ONS Core Damage Assessment Guidelines

OSC-3794 - Failed Fuel Determination using RIA 57-58

Nuclear Engineering uses the following calculations to determine containment volume versus containment level.

OSC-300 - Containment Volume and Heat Sink in Reactor Building

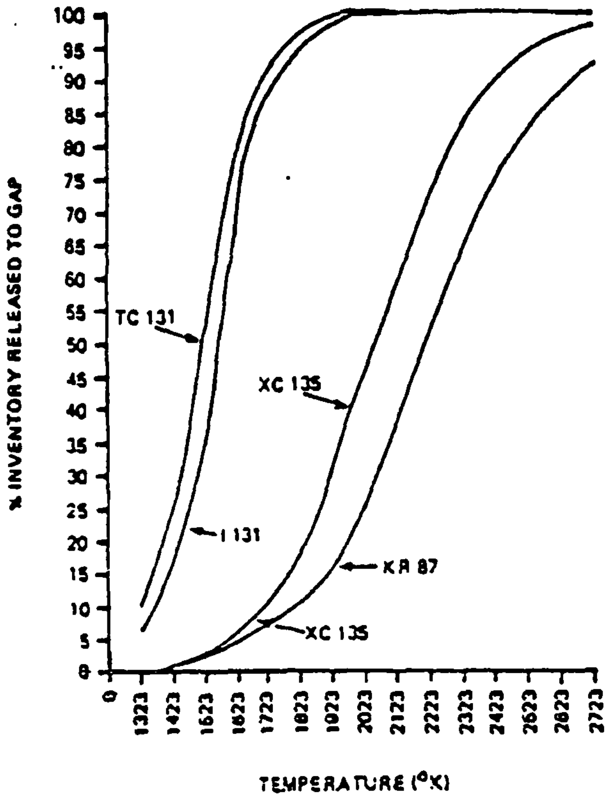
OSC-200 - Water Depth in Reactor Building

Information derived from the above calculations are used in RP/O/B/1000/18 to determine estimated failed fuel.

NOTE: Calculation documentation can be viewed at the Oconee Nuclear Engineering offices.

FIGURE I-3A

GAP INVENTORY VS. TEMPERATURE



BURNUP = 30,000 MWD/MTU
TIME = 420 DAYS

FIGURE I-3B

PERCENT ACTIVITY RELEASE FOR 100 PERCENT OVERTEMPERATURE CONDITIONS

Nuclear	Min.*	Max.*	Nominal**	Min.***	Max.***
Kr-85	40	70			
Xe-133	42	66	52.	40	70
I-131	41	55			
Cs-137	45	60			
Sr-90	0.08****				
Ba-140	0.1	0.2	0.15	0.08	0.2

* Release values based on TMI-2 measurements.

** Normal value is simple average of all Kr, Xe, I, and Cs measurements.

*** Minimum and maximum values of all Kr, Xe, I and Cs measurements.

**** Only value available.

FIGURE I-3C

RELATIONSHIP OF % FUEL OVERTEMPERATURE WITH % CORE INVENTORY RELEASED OF XE, KR, I, OR CS

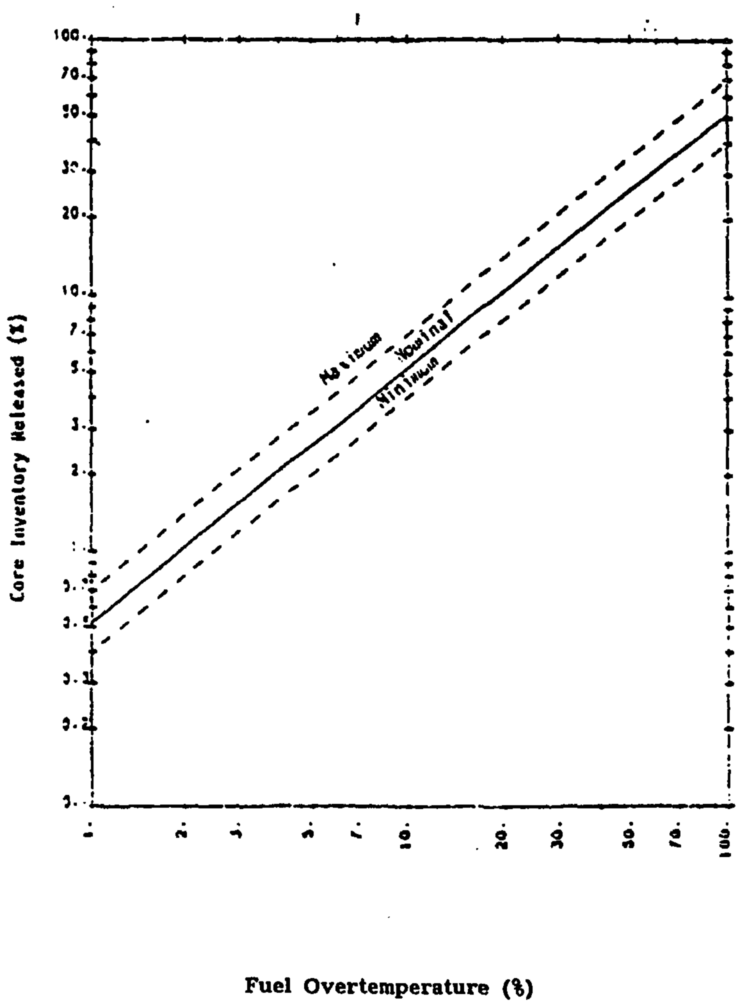


FIGURE I-4

TECHNICAL BASIS FOR ESTIMATION OF FAILED FUEL OVERHEATING WITH FUEL MELT

A. THEORY

In a fuel melt condition, all five release mechanisms discussed in Figure I-3 are involved. As fuel melts, up to 99% of the halogens and noble gases will be released. There will also be a significant release of barium and praseodymium. As in Case II, a linear relationship between failed fuel and isotope activity will be assumed. (See Figures I-4b and I-4c).

The major difference between fuel overheating without fuel melt and overheating with fuel melt is the percent of fission product inventory released from the fuel. The methodology for correcting isotopic decay and reactor power remains the same. The methodology for using hydrogen concentration to estimate core damage remains the same. The main changes will be in the radiochemistry method and area monitor method.

B. General Equations for Iodine and Xenon

1) $$P_i^{low} = \frac{\textit{Total Activity for Isotope} \quad (100)}{(\textit{Power Correction Factor} \; (\textit{Isotope Core Inventory})}$$

2) $$P_i^{high} = \frac{\textit{Total Activity for Isotope} \quad (100)}{(0.7)\,(\textit{Power Correction Factor})\,(\textit{Isotope Core Inventory})}$$

C. General Equations for Barium

1) $$P_i^{low} = \frac{\textit{Total Activity for Isotope} \quad (100)}{(0.44)\,(Y)\,(\textit{Isotope Core Inventory})}$$

2) $$P_1^{high} = \frac{\textit{Total Activity for Isotope} \quad (100)}{(0.10)\,(Y)(\textit{Isotope Core Inventory})}$$

D. General Equations for Praseodymium

1) $$P_i^{low} = \frac{\textit{Total Activity for Isotope} \quad (100)}{(0.024)\,(Y)\,(\textit{Isotope Core Inventory})}$$

2) $$P_i^{high} = \frac{\textit{Total Activity for Isotope} \quad (100)}{(0.008)\,(Y)\;\textit{Isotope Core Inventory})}$$

FIGURE I-4A

PERCENT ACTIVITY RELEASE FOR 100 PERCENT CORE MELT CONDITONS

Species	Large* LOCA	Transient*	Small* LOCA	Nominal** Release	Min.*** Release	Max.*** Release
Xe	88.35	99.45	78.38			
Kr	88.35	99.45	78.38			
				87	70	90
I	88.23	99.44	78.09			
Cs	88.55	99.46	78.84			
Te	78.52	94.88	71.04			
Sr	10.44	28.17	14.80	24	10	44
Ba	19.66	43.87	24.08			
Pr	0.82	2.36	1.02	1.4	0.8	2.4

* Calculated releases for severe accident scenarios without emergency safe-guard features, taken from draft NUREG-0956

** Normal release are averages of Xe, Kr, I, Cs, and Te groups or Sr and Ba groups

*** Maximum and minimum releases represent extremes of the groups.

FIGURE I-4B

RELATIONSHIP OF % FUEL MELT WITH % CORE INVENTORY RELEASED OF BA OR SR

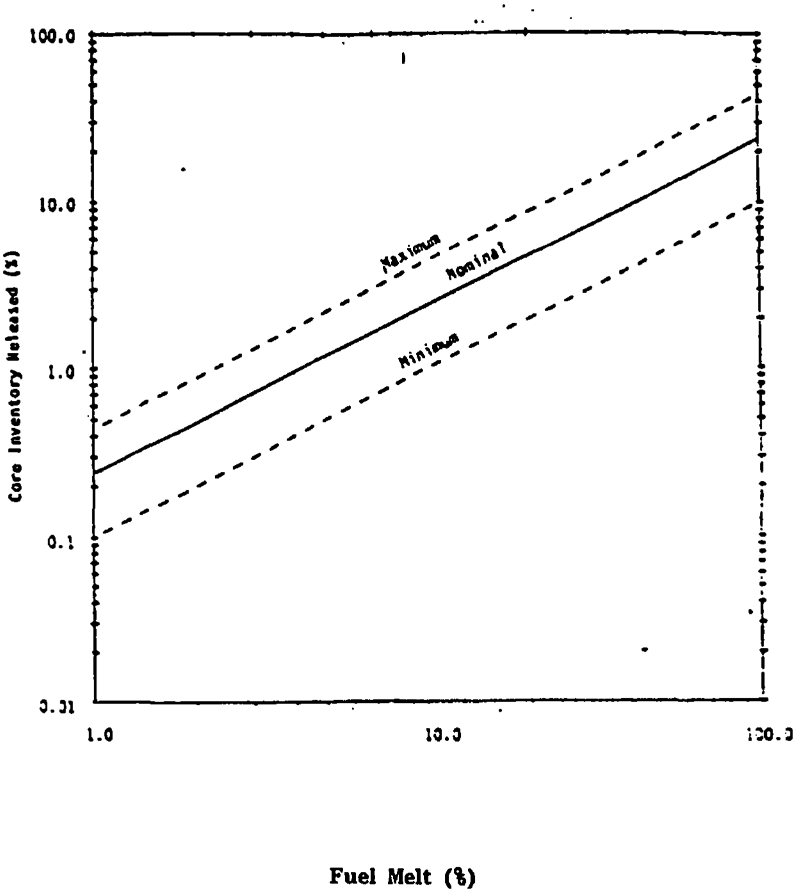


FIGURE I-4C

RELATIONSHIP OF % FUEL MELT WITH % CORE INVENTORY RELEASED OF XE, KR, I, CS, OR TE

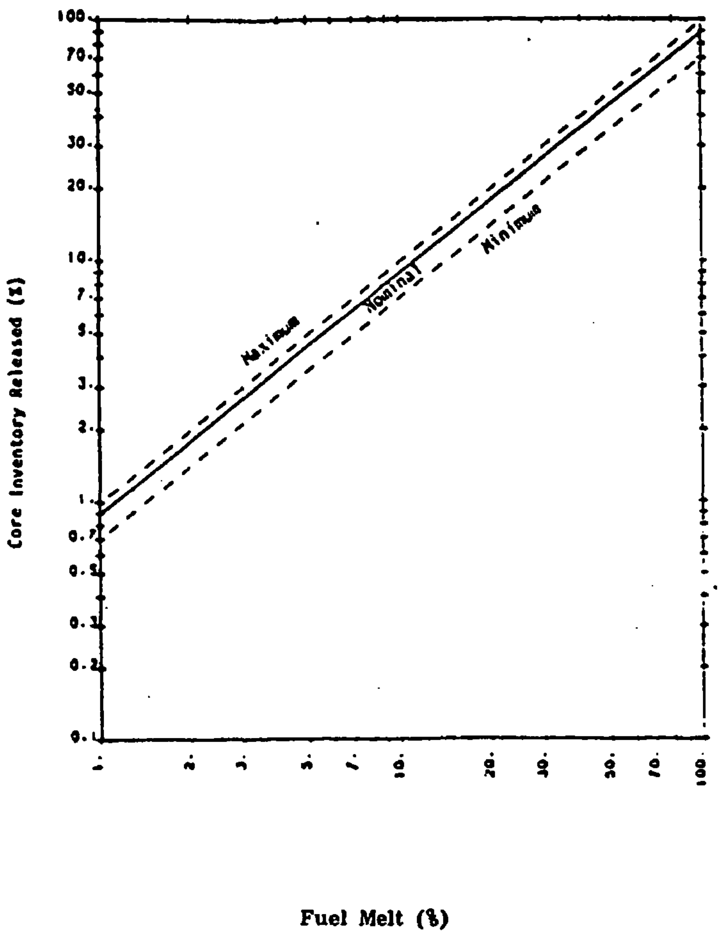


FIGURE I-5

ACTIVITY PER FUEL ASSEMBLY

Isotope	1 Cycle (Curies)	2 Cycles (Curies)	3 Cycles (Curies)
Kr85	2.102(3)*	3.272(3)	4.524(3)
Kr87	2.264(5)	1.433(5)	1.550(5)
Kr88	3.206(5)	2.030(5)	2.194(5)
Xe133	3.483(5)	6.335(5)	3.161(5)
Xe133m	1.610(5)	9.016(4)	1.164(5)
Xe135	4.714(5)	3.973(5)	4.499(5)
Xe135m	1.610(5)	1.255(5)	1.669(5)
I131	3.982(5)	3.075(5)	4.066(5)
I133	8.469(5)	6.317(5)	8.134(5)
I135	7.869(5)	5.879(5)	7.603(5)
Ba139	7.608(5)	5.561(5)	7.051(5)
Ba140	7.429(5)	5.432(5)	6.331(5)
Ba141	6.955(5)	5.073(5)	6.392(5)
Pr145	4.32(5)	3.177(5)	3.950(5)
Pr146	3.437(5)	2.537(5)	3.200(5)

*2.102(3) = 2.102×10^3

TOTAL CORE ACTIVITY

(Isotope)	(Curies)	(Curies)	% Δ*
Kr85	5.8405(5)**	5.84(5)	0.0
Kr87	3.0958(7)	4.00(7)	-29.207
Kr88	4.3837(7)	5.60(7)	-27.775
Xe133	1.355(8)	1.28(8)	7.800
Xe133m	2.1686(7)	3.07(6)	85.84
Xe135	7.7798(7)	2.19(7)	71.85
Xe135m	2.6751(7)	3.31(7)	-23.73
I131	6.5626(7)	7.42(7)	-13.065
I133	1.3523(8)	1.28(8)	5.340
I135	1.2597(8)	1.27(8)	-0.82
Ba139	1.193(8)		
Ba140	1.165(8)		
Ba141	1.087(8)		
Pr145	6.820(7)		
Pr146	5.442(7)		

$$* \frac{LOR2 - FSAR}{LOR2} \times 100$$

LOR2 = 59 (cycle 1 + cycle 2 + cycle 3)
where cycle 1, cycle 2, cycle 3 is on Table 5

**5.8405(5) = 5.8405 x 10^5

NOTE: FSAR values assume 400 EFPD and LOR2 values assume 421 EFPD

FIGURE I-6

TECHNICAL BASIS FOR ESTIMATION OF FAILED FUEL

AREA MONITORS FOR OVERHEAT WITHOUT FUEL MELT

Generally, a radiochemistry sample will give a more accurate indication of core damage than area monitors in the containment building. However, radiochemistry samples take a long time to evaluate, whereas area monitors give results immediately. This section will attempt to make some simplifying assumptions and give a rough estimate of failed fuel versus dose rate in containment. It will be assumed that only noble gases are in the containment atmosphere.* The noble gases are also assumed to be equally distributed throughout the containment building.

$\dot{X} = (2.62 \times 10^5) \times E\gamma \times 3600$ sec/hr

$\dot{X} = (9.432 \times 10^7) \times E\gamma$ R/hr

Where $\dot{X}$ = Ci/cm^3

$E\gamma$ = Average energy of all -γ rays per disintegration

$\dot{X}$ = Dose rate (R/HR)

Figure I-6a lists the average gamma energy level for the most prominent noble gas isotopes. Figure I-6b shows the methodology for calculating total noble gas dose rate. Figure I-6c is a plot of dose rate from Figure I-6b as the noble gases decay.

An approximation of failed fuel can be determined by the equation:

$$Fm = \frac{\dot{X}m}{(Y)\,\dot{X}(t)} \times 100$$

Where: Xm = Area monitor reading in the containment (R/HR)

$\dot{X}$ (t) = Dose rate from Figure I-6c (R/HR) at the appropriate time after shutdown

Y = Power correction factor

$$\text{Where } Y = \frac{\textit{Average Power for Prior 30 days}}{\textit{Rated power level}}$$

Fm = Fuel failure percent according to area monitors

It should be noted that this equation assumes a "PUFF" release of noble gases. If a small break LOCA occurs then the failed fuel estimate of m will be low. One possible method for using this equation during a small break LOCA is to wait until the monitor dose rate peaks and starts to decline. Figure I-9 is used to account for decay if required.

*It is understood that more isotopes than noble gases are released to the containment. However, modeling which isotopes and their activity is difficult. Therefore, only noble gases are considered. This will give a conservative estimate of failed fuel.

FIGURE I-6A

AVERAGE GAMMA ENERGY LEVEL

Isotope	(Mev)	Half-Life
Kr85m	0.151	4.4 Hrs.
Kr85	0.00211	10.76 Yrs.
Kr87	1.37	76.0 Min.
Kr88	1.74	2.79 Hrs.
Xe133m	0.326	2.26 Days
Xe133	0.030	5.27 Days
Xe135m	0.422	15.70 Min.
Xe135	0.246	9.20 Hrs.

FIGURE I-6B

VALUES FOR CALCULATING TOTAL NOBLE GAS DOSE RATE

Isotope	Activity in Containment At Shutdown	Eγ	χ	X
Kr85m	4.8405 (5)*	0.00211	9.3302(-6)	18.569
Kr87	2.0958 (7)	1.370	4.0397(-4)	5.22(5)
Kr88	3.0686 (7)	1.740	5.9148(-4)	9.71(5)
Xe133	9.3558 (7)	0.030	1.8034(-3)	5.10(4)
Xe133m	1.5180 (7)	0.0326	2.9260(-4)	9.00(4)
Xe135	5.4459 (7)	0.246	1.0497(-3)	2.43(5)

Total dose rate at shutdown = 1.88(6) R/HR)

* 4.8405(5) = 4.8405×10^5

FIGURE I-6C

DOSE RATE VS TIME FOR FUEL OVERHEATING WITHOUT FUEL MELT

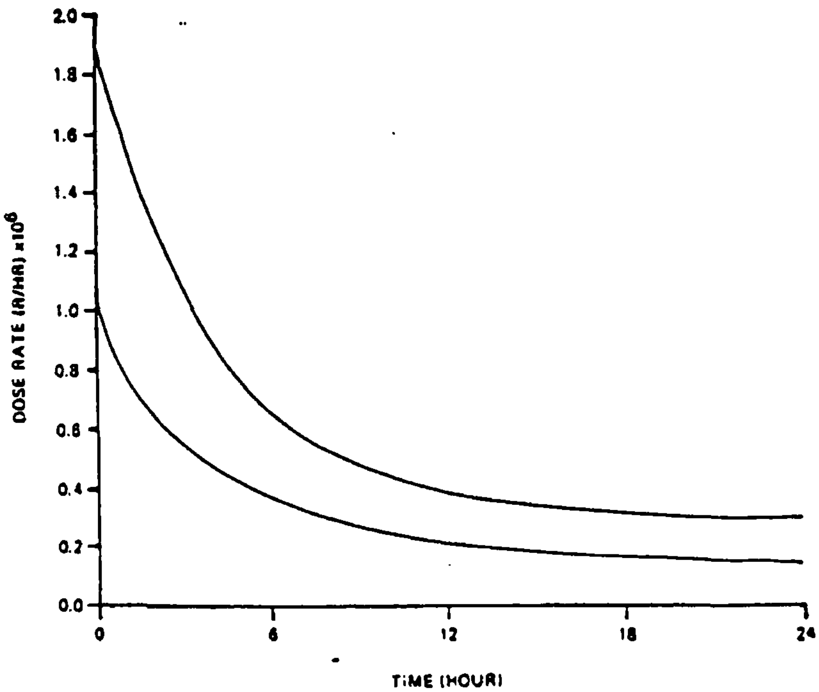


FIGURE I-7

TECHNICAL BASIS FOR ESTIMATION OF FAILED FUEL AREA MONITORS FOR FUEL MELT CONDITION

Dose rate is based on 70 to 100 percent release of noble gases instead of the 40 to 70 percent used in Figure I-6. Figure I-7a shows a plot of dose rate versus time for 100% failed fuel. An approximation of failed fuel can be determined by the equation:

$$Fm = \frac{X\,m}{(Y)\,(X(t))} \cdot 100$$

Where: Xm = Area monitor reading in the containment (R/HR)

X(t) = Dose rate from Figure I-7a

Y = Power correction factor= $\dfrac{Average\ Power\ for\ \mathrm{Pr}\,ior\ 30\,days}{Rated\ power\,level}$

F m = Fuel failure percent

FIGURE I-7A

DOSE RATE VS. TIME FOR FUEL MELT

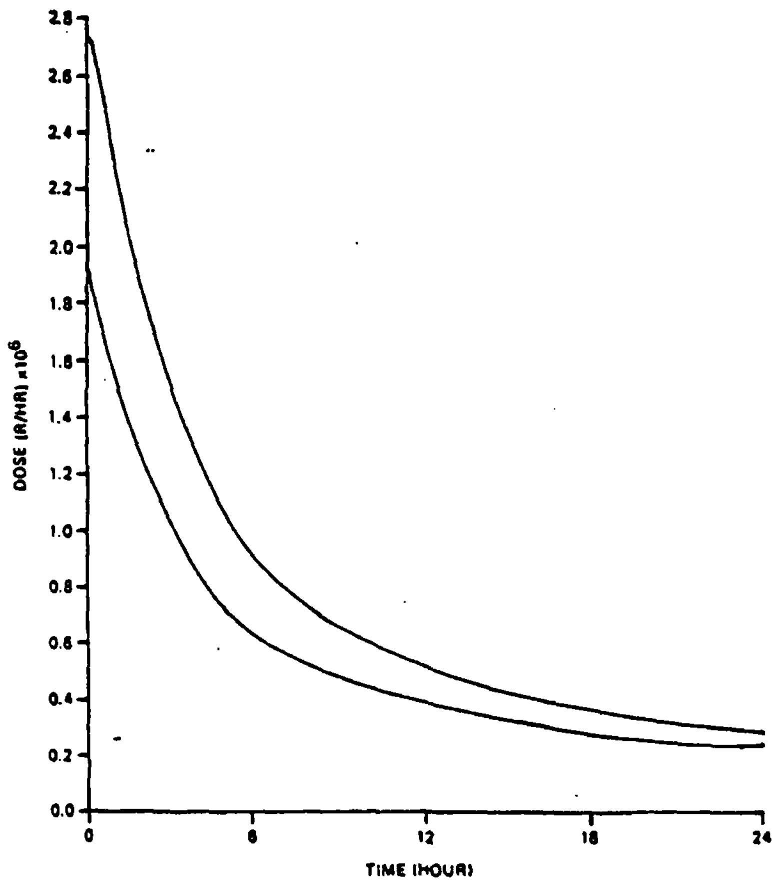


FIGURE I-8

TECHNICAL BASIS FOR ESTIMATION OF FAILED FUEL HYDROGEN CONCENTRATION IN THE CONTAINMENT BUILDING

At approximately 1600°F zirconium reacts with water to produce hydrogen. The greater the temperature the faster the reaction rate. During the zirconium - water reaction heat is also released which raises the cladding temperature which increases the reaction rate. If the hydrogen concentration is constant or increasing slightly without recombiners on, then the cladding temperature is probably around 1600°F or less. If hydrogen concentration is increasing rapidly (with or without recombiners) then the clad temperature is above 1600°F. A rough estimate of core damage can be made, based on hydrogen concentration in the containment if the following assumptions are made.

1. All hydrogen produced in the RCS is released to the containment building.
2. All hydrogen in the containment building comes from the zirconium - water reaction*.
3. The recombiners have not be turned on (i.e., no hydrogen has been burned).

The equation for the zirconium - water reaction is

$Zr + 2H_20 \rightarrow Zr0_2 + 2H_2$

or

Two moles of hydrogen in the containment building are produced by the reaction of one mole of zirconium in the core.

At STP 1 mole of hydrogen has a volume of

22.4. ℓ or 0.79 ft^3

Volume of hydrogen in containment = hydrogen concentration

(volume percent unit) X containment free volume

or

$V_{H_2} = X_{H_2} \times V_{containment}$

where V_{H_2} is the volume of hydrogen in containment as a percent of atmosphere.

$$\frac{P_1 V_1}{T_s} = \frac{P_2 V_2}{T_2}$$

* There are other sources of hydrogen, but assuming all hydrogen is produced by the zirconium - water reaction will give a conservative estimate.

$$V_{STP} = \frac{P_C\, V_{H_2}\, T_{STP}}{P_{STP}\, T_C}$$

$$V_{STP} = \frac{P_C}{T_C} \quad \frac{TSTP}{P_{STP}} \quad XH_2 \quad V_C$$

Where V_{STP}, T_{STP}, P_{STP} = Volume, temperature, and pressure at STP

$T_{STP} = 492°R$

$P_{STP} = 14.7$ PSI

$T_{STP} = 492°R$

$P_{STP} = 14.7$ PSI

V_c = Containment free volume = 1,832,033 ft^3

$$V_{STP} = \frac{P_C}{T_C} \quad \frac{492}{14.7} \quad (1{,}832{,}033) \quad \left(X_{H_2}\right)$$

$$V_{STP} = \frac{P_C}{T_C} \quad X_{H_2} \quad (6.1317 \text{ s } 10^7)$$

The total amount of hydrogen moles in the containment = $\dfrac{V_{STP}}{Volume\ of\ one}\ mole$

$$M_H = \frac{P_C}{T_C} \quad X_{H_2} \quad \frac{6.317\text{x}10^7}{0.79} = \frac{P_C}{T_C} \quad X_{H_2} \quad (7.7616 \text{ x } 10^7)$$

Since it takes 1 mole of Zr to produce 2 moles of H_2 then the number of zirconium moles reacting with hydrogen is 1/2 M_H

or

$$M_{Zr} = 1/2\, M_H = 1/2 \; \frac{P_C}{T_C} \quad X_{H_2} \quad (7.7616 \text{ x } 10^7)$$

The zirconium mass that reacts can be calculated by the equation

$Z_r = M_{Zr} \text{ x } W_m$

Where W_m = gram - Atomic Weight = 91.22 gr/mole

$$Z_r = (M_{Zr})\,(91.22) = (\frac{P_C}{T_C} \quad X_{H_2})\,(3.5401 \text{ x } 10^9)$$

The fraction of zirconium that reacts with water is calculated by

$$F_{Zr} = \frac{Z_r}{Z_{r_{tot}}}$$

Where Zr_{tot} = total amount of zirconium in the core = 8.1204×10^7 gm

$$F_{Zr} = \frac{P_C}{T_C} \quad X_{H_2} \frac{3.5401 \times 10^9}{8.1204 \times 10^7} = \frac{P_C}{T_C} \quad X_{H_2} \quad (43.594)$$

$$F_{Zr} = \frac{P_C}{T_C} \quad X_{H_2} \frac{43.6}{100} = \frac{P_C}{T_C} \quad P_{H_2} \quad (.436)$$

Where: F_{Zr} = Fraction of core damage

P_{Zr} = Percent of core damage

P_c = Containment pressure (PSIA)

T_c = Containment temperature (°F + 460)

P_{H_2} = Percent of hydrogen in containment atmosphere

$$X_{H_2} = \frac{P_{H_2}}{100}$$

It should be noted that when estimates of core damage are made using radio-chemistry samples, area monitors and hydrogen concentration that the results can be greatly different. Whenever possible, all three methods should be used and their combined results used as an indication of core damage.

FIGURE I-9

TECHNICAL BASIS FOR ESTIMATION OF FAILED FUEL

ISOTOPE DECAY CORRECTION

The specific activity of a sample is decay adjusted to time of reactor shutdown using the following equation.

$$\text{Specific activity at shutdown} = \frac{\textit{Specific activity}(\textit{meaured})}{e^{-\lambda_i t}}$$

Where:

λ_i = Radioactive decay constant, 1/sec

t = Time period from reactor shutdown to time of sample analysis, sec.

Since this correction may also be performed by some analytical equipment, care must be taken to avoid duplicate correction. Also, considerations must be given to account for precursor effect during the decay of the nuclide. For this methodology, only the parent-daughter relationship associated with the methodology. The decay scheme of the parent-daughter relationship (Figure I-9a) is described by the following equation.

$$Q_B = \frac{\lambda_B}{\lambda_B - \lambda_A} Q_A^{o} \left(e^{-\lambda_A t} - e^{-\lambda_B t} \right) Q_B{}^{o} e^{-\lambda_B t}$$

Where:

Q_A^{o} = Activity (Ci) or specific activity (µCi/gm or µCi/cc) of the parent at shutdown

Q_B^{o} = Activity (Ci) or specific activity (µCi/gm or µCi/cc) of the daughter at shutdown

Q_B = Activity (Ci) or specific activity (µCi/gm or µCi/cc) of the daughter at time of sample

λ_A = Decay constant of the parent, sec^{-1}

λ_B = Decay constant of the daughter, sec^{-1}

t = Time period from reactor shutdown to time of sample analysis, sec.

Since the activity of the daughter at sample time is due to the decay of the parent and the decay of the daughter initially released at shutdown, an estimation of the fraction of the measured activity at sample time due to only the decay of daughter is required.

To use the above equation to determine the fraction, an assumption is made that the fraction of source inventory released of the parent and the daughter at time of shutdown are equal (for the nuclides used here within a factor of 2). The following steps should be followed to calculate the fraction of the measured activity due to the decay of the daughter that was released and then to calculate the activity of the daughter released at shutdown.

1. Calculate the hypothetical daughter concentration (Q_B) at the time of the sample analysis assuming 100 percent release of the parent and daughter source inventory.

$$Q_B = \frac{\lambda_B}{\lambda_B - \lambda_A} Q_A^o \left(e^{-\lambda_A t} - e^{-\lambda_B t} \right) Q_B^o e^{-\lambda_B t}$$

Where:

Q_A^o	=	100% source inventory (Ci) of parent, Table 6
Q_B^o	=	100% source inventory (ci) of daughter, Table 6
$Q_B(t)$	=	Hypothetical daughter activity (Ci) at sample time
K	=	If parent has 2 daughters, K is the branching factor, Table 6
λ_A	=	Parent decay constant, sec^{-1}
λ_B	=	Daughter decay constant, sec^{-1}
t	=	Time period from reactor shutdown to time of sample analysis, sec.

2. Determine the contribution of only the decay of the initial inventory of the daughter to the hypothetical daughter activity at sample time.

$$Fr = \frac{Q_B^o e^{-\lambda_B t}}{Q_B(t)}$$

3. Calculate the amount of the measured sample specific activity associated with the decay of the daughter that was released.

   M_B = Fr x measure specific activity (µCi/gm or µCi/cc)

4. Decay correct the specific activity (M_B) to reactor shutdown.

$$M_B = \frac{B}{-\lambda_B t}$$

FIGURE I-9A

PARENT-DAUGHTER RELATIONSHIPS

Parent	Parent Half Life[1]*	Daughter	Daughter Half Life*	K^2**
Kr-88	2.8 h	Rb-88	17.8 m	1.00
I-131	8.05 d	Xe-131m	11.8 d	.008
I-133	20.3 h	Xe-133m	2.26 d	0.24
I-133	20.3 h	Xe-133	5.27 d	.976
Xe-133m	2.26 d	Xe-133	5.27 d	1.00
I-135	6.68 h	Xe-135	9.14 h	.70
Xe-135m	15.6 m	Xe-135	9.14 h	1.00
I-135	6.68 h	Xe-135m	15.6m	.30
Te-132	77.7 h	I-132	2.26 h	1.00
Sb-129	4.3 h	Te-129	68.7 m	.827
Te-129m	34.1 d	Te-129	68.7 m	.680
Sb-129	4.3 h	Te-129m	34.1 d	.173
Ba-140	12.8 d	La-140	40.22 h	1.00
Ba-142	11 m	La-142	92.5 m	1.00
Ce-144	284 d	Pr-144	17.27 m	1.00

[1] * Table of Isotopes, Lederer, Hollander, and Perlman, Sixth Edition

[2] ** Branching of decay factor

J. PROTECTIVE RESPONSE

To assure that a range of protective actions is available for the plume exposure pathway for emergency workers and the public. Guidelines for protective actions during an emergency, consistent with Federal guidance, are developed and in place and protective actions for the ingestion exposure pathway appropriate to the locale have been developed.

To protect onsite personnel during hostile action and ensure the continued ability to safely shutdown the reactor and perform the functions of the emergency plan a range of protective actions are in place.

J. 1 Accountability

The Oconee Nuclear Site has a Site Assembly Procedure that gives specific instructions to follow during a site assembly. Also, each division/section has specific directives that provide guidance for their personnel. (Site Assembly locations, Figure J-5)

Methods to notify and alert onsite personnel (essential and non-essential) during hostile action activities are described in AP/0/A/1700/045, "Site Security Threats". RP/0/B/1000/010 "Procedure for Emergency Evacuation/Relocation of Site Personnel". RP/0/B/1000/009, "Procedure for Site Assembly".

J. 2 Relocation Assembly Areas and Evacuation Routes

Should it be determined that non-essential personnel would need to be relocated onsite or evacuated from the site, procedures are in place to handle this process. Agreements have been reached with local authorities for the use of the Oconee and Pickens school facilities for evacuation of personnel. (Appendix 5)

Site directives and procedures establish onsite relocation areas as well as evacuation routes (Figure J-2) to suitable offsite locations.

J. 3 Site Evacuation Procedures - Personnel

The site evacuation procedure establishes guidelines for evacuation from the station site. This procedure outlines the radiological exposure limits. All station personnel inside the protected area will be monitored before being evacuated from the station. Records will be kept of the individual's exposure/contamination level prior to evacuation. All personnel, so designated, will then be evacuated to pre-designated areas for thorough personnel monitoring and decontamination.

Records will be kept for the station and personnel files. All personnel will be required to sign a copy of the monitor readings that will be recorded in personnel files. (Figures J-3, J-4)

During hostile threat conditions relocation of personnel away from the hazard areas are performed in accordance with AP/0/A/1700/045, "Site Security Threats". RP/0/B/1000/010, "Procedure for Emergency Evacuation/Relocation of Site Personnel". RP/0/B/1000/009, "Procedure for Site Assembly".

J. 4 Site Evacuation Procedures-Decontamination/Non Essential/Essential Personnel Criteria

Personnel who have been determined to be non-essential may be evacuated from the plant site in the event of a Site Area Emergency Classification. However, non-essential personnel are always evacuated from the site during a General Emergency Classification. Provisions are made for the decontamination of vehicles and personnel at an offsite location if the situation should warrant that to be necessary.

EPZ - Population Alerting and Notification

See Oconee County FNF Plans.
See Pickens County FNF Plans.
See State of South Carolina FNF Plans, Site Specific.
See Appendix 3.

J.5 & J.6 Site Evacuation Procedures-Personnel Accountability

Within thirty minutes of a Site Assembly, all persons at the Oconee Nuclear Station shall be accounted for and any person(s) determined to be missing from their control station, will be identified by name. To assist in the location of missing person(s), the Emergency Coordinator will appoint a Search and Rescue Team. Search procedures will be coordinated through the Operational Support Center.

After all non-essential personnel have been evacuated from the site, logsheets will be kept by Radiation Protection personnel in the Operational Support Center of all persons onsite together with their Radiation Protection records to include the following:

a. Individual respiratory protection
b. Protective clothing
c. Use of Radioprotective drugs

During hostile threat conditions personnel accountability is performed in accordance with AP/0/A/1700/045, "Site Security Threats" and RP/0/B/1000/009, "Procedure for Site Assembly".

J. 7 Protective Actions Recommendations

The Emergency Coordinator (Operations Shift Manager or Station Manager) or the EOF Director (depending on the facility activation) will be responsible for contacting the State and/or local governments to give prompt notification for implementing protective measures within the plume exposure pathway, and beyond it if necessary. Procedure RP/0/A/1000/024, "Protective Action Recommendations" and SR/0/A/2000/003, "Activation of the Emergency Operations Facility" has been written to provide specific guidance for issuing protective action recommendations under various plant conditions to the Emergency Coordinator in the TSC and the EOF Director in the EOF Figure (J-1) respectively. The decision to use sheltering as an alternative to evacuation for impediments and special populations is one that will be made by offsite officials. If dose projections show that PAGs have been exceeded at 10 miles, the dose assessment code and in-field measurements, when available, shall be used to calculate doses at various distances down wind to determine how far from the site PAG levels are exceeded. The Radiological Assessment Manager shall forward the results to the EOF Director who will communicate this information to the offsite authorities.

Figure J-1A (Protective Action Guides) is adopted from EPA 400 and guidance in state plans on use of KI and considers protective action based on projected avoided dose.

J. 8 Evacuation Time Estimates

A description of the methods and assumptions used in developing the analysis of evacuation time estimates is included in the current Evacuation Time Estimate Study for the Oconee Nuclear Site. (ONS-ETE-12142012, Rev. 000; ONS Evacuation Time Estimates (ETE) Dated 12/14/2012.) The Evacuation Time Estimates will be considered in evaluating protective action recommendations from the Technical Support Center or the Emergency Operations Facility. A copy of the most recent study is available in the Technical Support Center and the Emergency Operations Facility.

An updated ETE analysis will be submitted to the NRC under §50.4 no later than 365 days after ONS determination that the criteria for updating the ETE have been met and at least 180 days before using it to form protective action recommendations and providing it to State and local governmental authorities for use in developing offsite protective action strategies.

The criteria for determination that an updated ETE analysis have been met:

a. The availability of the most recent decennial census data from the U.S. Census Bureau;

OR

b. If at any time during the decennial period, the EPZ permanent resident population increases such that it causes the longest ETE value for the 2-mile zone or 5-mile zone, including all affected Emergency Response Planning Areas, or for the entire 10-mile EPZ to increase by 25 percent or 30 minutes, whichever is less, from the currently NRC approved or updated ETE.

During the years between decennial censuses ONS will estimate EPZ permanent resident population changes once a year, but no later than 365 days from the date of the previous estimate, using the most recent U.S. Census Bureau annual resident population estimate and State/local government population data, if available. ONS will maintain these estimates so that they are available for NRC inspection during the period between decennial censuses and shall submit these estimates to the NRC with any updated ETE analysis.

ONS' ETE analysis, using the 2010 decennial census data from the U.S. Census Bureau, was submitted to the NRC via §50.4 on December 14, 2012.

J.9 <u>Implementing Protective Measures</u>

See Pickens County FNF Plans.
See Oconee County FNF Plan.
See State of South Carolina FNF Plans, Site Specific.

For hostile action events, a range of protective actions for onsite workers including evacuation of essential personnel from potential target buildings, timely evacuation or relocation of non-essential site personnel, dispersal of critical personnel to safe locations, sheltering of personnel away from potential site targets and accountability of personnel after the attack are provided in emergency plan implementing procedures AP/0/A/1700/045, "Site Security Threats", RP/0/B/1000/010, "Procedure for Emergency Evacuation/Relocation of Site Personnel", RP/0/B/1000/009, "Procedure for Site Assembly".

J.10 Implementation of Protective Measures for Plume Exposure Pathway

J.10.a EPZ - Maps of Oconee EPZ.
See Figure A, page i-5.

J.10.b EPZ - Population Distribution Charts
See Appendix 4 Evacuation Time Estimates

J.10.c EPZ - Population Alerting and Notification

See Oconee County FNF Plans.
See Pickens County FNF Plans.
See State of South Carolina FNF Plans, Site Specific.

See Appendix 3.

J.10.d EPZ - Protecting Immobile Persons

See Oconee County FNF Plans.
See Pickens County FNF Plans.
See State of South Carolina FNF Plans, Site Specific.

J.10.e Use of Radioprotective Drugs for Persons in EPZ

See Oconee County FNF Plans.
See Pickens County FNF Plans.
See State of South Carolina Operational Radiological Emergency Response Plan - SCOREP, (FNF Plans, Site Specific).

J.10.f Conditions For Use of Radioprotective Drugs

See Oconee County FNF Plans.
See Pickens County FNF Plans.
See State of South Carolina SCOREP, (FNF Plans, Site Specific).

J.10.g Means of Relocation and
J.10.h State/County Relocation Center Plans

See Oconee County FNF Plans.
See Pickens County FNF Plans.
See State of South Carolina FNF Plans, Site Specific.

J.10.i Evacuation Route - Traffic Conditions

See Oconee County FNF Plans.
See Pickens County FNF Plans.
See State of South Carolina FNF Plans, Site Specific.

J.10.j Evacuated Area Access Control

See Oconee County FNF Plans.
See Pickens County FNF Plans.
See State of South Carolina FNF Plans, Site Specific.

J.10.k Planning for Contingencies in Evacuation

See Oconee County FNF Plans.
See Pickens County FNF Plans.
See State of South Carolina FNF Plans, Site Specific.

J.10.l State/County Evacuation Time Estimates

See Oconee County FNF Plans.
See Pickens County FNF Plans.
See State of South Carolina FNF Plans, Site Specific.

J.10.m Bases for Protective Action Recommendations

DUKE ENERGY uses the following considerations in determining protective action recommendations:

1) Protective Action Guides (PAG)
2) Core Condition

See State of South Carolina FNF Plan, Site Specific

J.11 Ingestion Pathway Planning:

See State of South Carolina FNF Plans.
See State of Georgia FNF Plans.
See State of North Carolina FNF Plans.

J. 12 Relocation Center - Registering: & Monitoring

See Oconee County FNF Plans.
See Pickens County FNF Plans.
See State of South Carolina FNF Plans.

FIGURE J-1

DUKE ENERGY COMPANY
OCONEE NUCLEAR SITE

PROTECTIVE ACTION RECOMMENDATION FLOW CHART

CONDITION	FUEL DAMAGE SYMPTOMS	CONTAINMENT STATUS	PROTECTIVE ACTION RECOMENDED
General Emergency Declared	♦Loss of critical functions required for core protection ♦High CETCs ♦RB High rad levels	Not applicable	Evacuate 2- mile radius and 5- miles downwind unless conditions make evacuation dangerous. (See Note 1). Shelter any sector not evacuated.
Additional protective recommendations will be based on the following conditions from either the Technical Support Center or the Emergency Operations Facility. TSC or the EOF shall continue assessment based on all available plant and field monitoring information. Modify protective actions as necessary. Locate and evacuate people from hot spots. Do not relax protective actions until the source of the threat is clearly under control.			
Fuel Damage Detected by Monitors	♦High rad levels as determined by Reactor Building and unit vent monitors	Known containment breach or RB pressure greater than 1 PSIG	Dose calculations required to determine additional evacuation requirements and recommendations on use of stable iodine. Shelter any sector not evacuated.
Condition 2 failed fuel as determined by RP/0/B/1000/018	♦RB high rad levels ♦H-2 increasing ♦Clad >1200° F	No credit is taken for containment.	Evacuate 5-mile radius and 10-miles downwind. Shelter any sector not evacuated.

Note 1. Dangerous travel conditions or immobile infirmed population.

FIGURE J-1A

DUKE ENERGY COMPANY
OCONEE NUCLEAR SITE

PROTECTIVE ACTION GUIDES

Protective Action	Recommended Actions	Comments
Evacuation	1-5 rem TEDE from significant external and internal exposure from gamma radiation from the plume and from deposited material	Although the PAG is expressed as a range, under normal conditions evacuation of the public is usually justified when the projected dose to an individual is one rem.
Evacuation	5-25 rem thyroid CDE from significant inhalation of activity in the plume	Although the PAG is expressed as a range, under normal conditions evacuation of the public is usually justified when the projected dose to an individual is five rem.
Administration of stable iodine (e.g. KI)	5 rem thyroid CDE from radioiodine	Duke Energy will recommend that offsite agencies consider the use of KI at 5 rem thyroid CDE.

Sheltering Concepts:

Duke Energy will make evacuation recommendations to the offsite agencies. However, if hazardous environmental conditions exists, Oconee emergency personnel will provide information (plant status, release magnitude, release duration, consequences) for the offsite agencies to use in making their decisions as to whether or not the public will be evacuated or sheltered.

FIGURE J-2

DUKE ENERGY COMPANY
OCONEE NUCLEAR SITE

<u>EVACUATION ROUTES CHART</u>

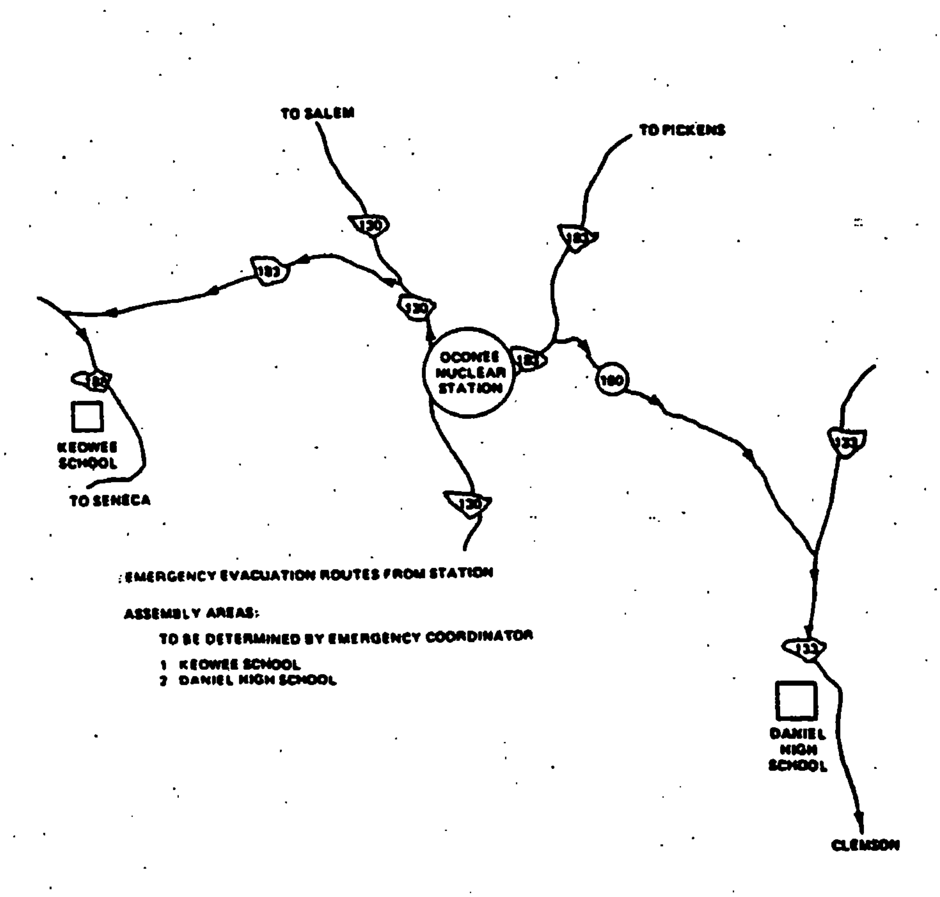


FIGURE J-3

DUKE ENERGY COMPANY
OCONEE NUCLEAR SITE

INDIVIDUAL CONTAMINATION EXPOSURE LEVELS

LICENSEE: DUKE ENERGY COMPANY

IDENTIFICATION INFORMATION

Name:______________________________ Date: ____________________

Social Security Number _________________________ Time: _____________

Employer:______________________ R.P. Badge ______________

CONTAMINATION EXPOSURE LEVELS

Instrument Used:____________ Instrument Reading: ________________________
(RM-14 with thin window detector or equivalent)

____

Date:______________ Employee Signature: _______________________________

Remarks: ____

Address: ______________________________

To the individual named above_____________________________, this report is furnished to you so that you have a prompt record of your radioactive contamination level.

____
Radiation Protection Manager

Date: ______________________

Copies to:
Individual
Individual File
(New Form)

FIGURE J-4

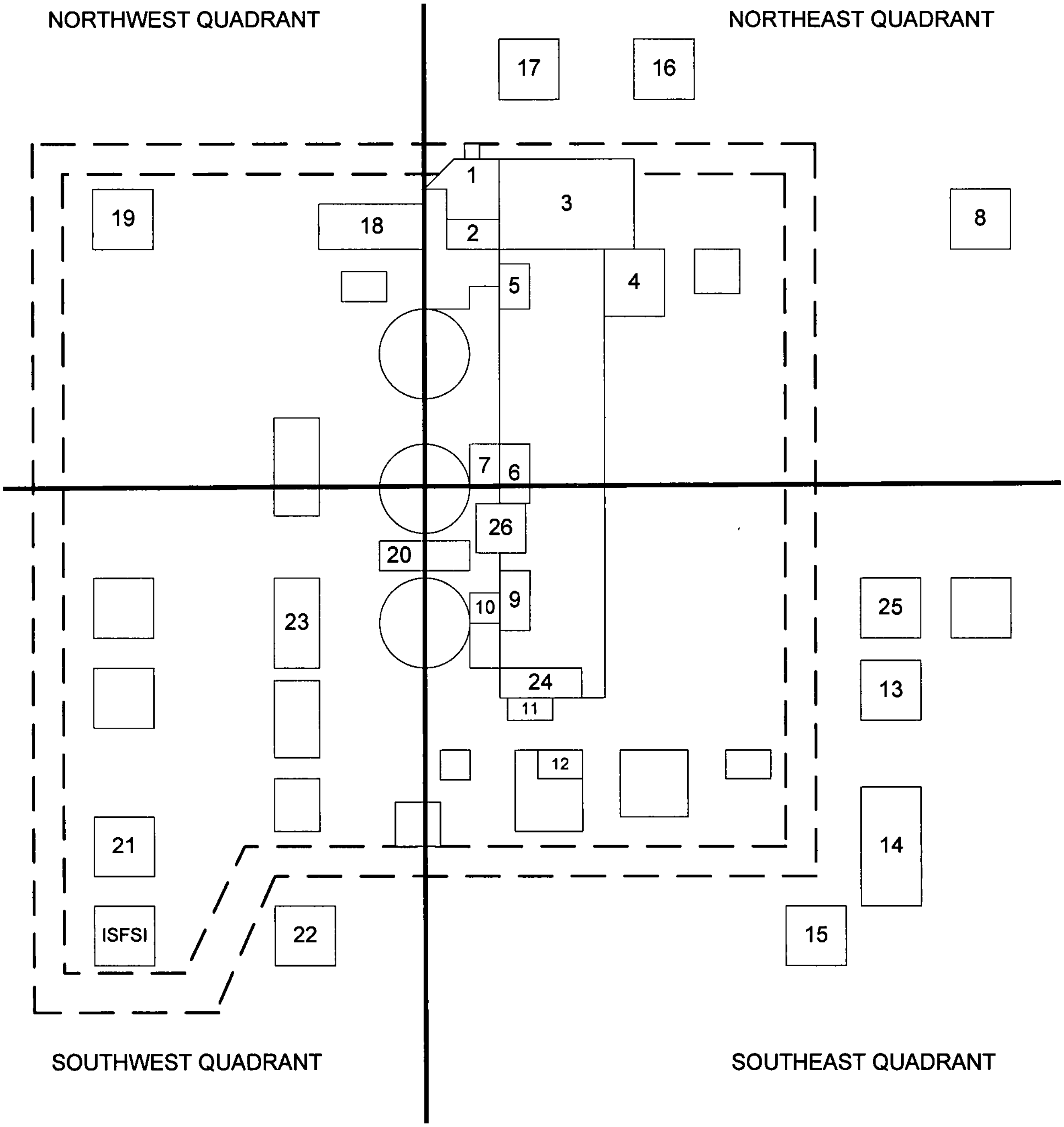
DUKE ENERGY COMPANY
OCONEE NUCLEAR SITE

INITIAL PERSONNEL CONTAMINATION RECORD
(ONSITE)

NAME	RP BADGE NUMBER	INITIAL DOSE RATE (mRad/hr)	DOSE RATE (mRad/hr) After Decon

FIGURE J-5

Oconee Nuclear Site
Building Layout



NORTHWEST QUADRANT		NORTHEAST QUADRANT
18. Administrating Building 19. Oconee Office Building **SOUTHWEST QUADRANT** 20. RP Assembly Building 21. Interim Outage Building 22. Operations Center (Geo-Technical Ctr.) 23. Warehouse Offices **SOUTHEAST QUADRANT** 9. Turbine Bd. 3 Offices 10. Unit 3 CR 11. Technical Support Bd. 12. Radwaste Facility 13. Oconee Garage	 14. Oconee Complex 15. L-1 Storage Yard 24. Turbine Bd. South Offices 25. Maintenance Training Facility 26. SPA, RP Assembly Area	1. Security Building 2. Locker Building 3. Maintenance Service Bd./Clean Machine Shop 4. Maintenance Support Building 5. Turbine Building North Offices 6. Turbine Building 1&2 Offices/WCC 7. Unit 1&2 Control Room 8. Keowee Hydro Station 16. World of Energy 17. Oconee Training Center

P. Responsibility for the Planning Effort: Development, Periodic Review and Distribution of the Emergency Plans

To assure that responsibilities for plan development, review and distribution of emergency plans are established and that planners are properly trained:

P.1 Training for Emergency Planning Personnel

Training for emergency planning personnel shall be provided in the form of workshop/seminar sessions on an annual basis. Courses developed by the Duke Training Center are also available in technically related subjects that will enhance the working knowledge of these people.

P.2 & P.3 Overall Authority

The Site Vice-President has the overall authority and responsibility for all hazards emergency response planning. The planning effort is delegated to the Manager, Emergency Planning.

The Manager of Emergency Planning at the Oconee Nuclear Site shall have the responsibility for the development, review and coordination of the site emergency plans with other response organizations and shall be responsible for conducting the biennial exercise, drills and training sessions to test the Oconee Nuclear Site Emergency Plan. This person is employed in the Safety Assurance Group.

P.4 & P.5 Review and Update of Emergency Plan

The ONS Emergency Plan shall be reviewed and updated annually. An in-depth review of the Emergency Plan will be made to determine if any/all changes have been made as a result of drills, exercises, commitments, audits, new regulatory requirements, and any other identified mechanism used to determine the appropriateness of the Emergency Plan. The Manager of Emergency Planning or designee is responsible for conducting the review and updating/revising the Emergency Plan and/or Implementing Procedures, as required. Once the review has been completed and changes made as determined, the Emergency Plan shall be certified as current.

Approved revisions of the Emergency Plan and Implementing Procedures shall be distributed according to Appendix 6, (Distribution of Emergency Plan and Implementing Procedures). Appendix 6 carries an itemized list of all organizations and individuals receiving copies of the Emergency Plan and Implementing Procedures. Revised pages of the Emergency Plan shall be dated and marked to show where changes have been made.

P.6 Supporting: Plans

Figure P-2 lists plans in support of the ONS Emergency Plan.

P.7 Implementing Procedures

Written procedures will be established, implemented and maintained covering the activities associated with emergency plan implementation. Each procedure and changes thereto, shall be approved by the responsible manager prior to implementation.

Implementing procedures are indexed and cross referenced to the section applicable in NUREG 0654. (Figure P-1)

P.8 Table of Contents

The Oconee Nuclear Site Emergency Plan and Implementing Procedures contain a table of contents and an index tab system.

P.9 Independent Audit

The Nuclear Safety Review Board Chairman will arrange for an independent review of Oconee Nuclear Station's Emergency Preparedness Program as necessary, based on an assessment against performance indicators, and as soon as reasonably practicable after a change occurs in personnel, procedures, equipment, or facilities that potentially could adversely affect emergency preparedness, but no longer than 12 months after the change. In any case, all elements of the emergency preparedness program will be reviewed at least once every 24 months. Guidance for performing the assessment against the performance indicators is provided in the Emergency Planning Functional Area Manual. The independent review will be conducted by the Independent Nuclear Oversight Division, which will include the following plans, procedures, training programs, drills/exercises, equipment, and State/local government interfaces:

1. Oconee Nuclear Station Emergency Plan
2. Oconee Nuclear Station Emergency Plan Implementing Procedures
3. State/Local Support Agency Training Program
4. Site Emergency Response Training Program
5. Public & Media Training/Awareness
6. Equipment: Communications, Monitoring, Meteorological, Public Alerting
7. State/Local Plan Interface

The review findings will be submitted to the appropriate corporate and nuclear site management. The part of the review involving the evaluation of the adequacy of interface with State and local governments will be reported to the appropriate State and local governments. Corporate or nuclear site management, as appropriate, will evaluate the findings affecting their area of responsibility and ensure effective corrective actions are taken. The results of the review, along with recommendations for improvements, will be documented, and retained for a period of five (5) years.

The review findings will be submitted to the appropriate corporate and nuclear site management. Appropriate portions of the review findings will be reported to the involved federal, state, and local organizations. The corporate or nuclear site management, as appropriate, will evaluate the findings affecting their area of responsibility and ensure effective corrective actions are taken. The result of the review, along with recommendations for improvements, will be documented and retained for a period of five years.

P.10 Phone Number Update

The Emergency Telephone Directory is updated quarterly. The Emergency Telephone Directory is a separate document and is not a part of the Oconee Nuclear Site Implementing Procedures.

FIGURE P-1

DUKE ENERGY COMPANY
OCONEE NUCLEAR STATION

IMPLEMENTING PLAN CROSS REFERENCE

A.1.a	Appendix 5 - Agreement Letters
A.1.b	RP/0/A/1000/002, RP/0/A/1000/019, SR/0/A/2000/003, RP/0/A/1000/025, SAD 6.1
A.2.a	Appendix 5 - Agreement Letters
B.1	CSM 5.1, MD 9.1, WPG 1.5, OMP 1-7, RPSM 11.1, Business Management Emergency Plan, SSG-102, NSC-110, EM-5.1, ONS HR Emergency Plan, DTG-007
B.4	SAD 6.1, RP/0/B/1000/019, RP/0/A/1000/002, SR/0/B/2000/003
B.5	RP/0/A/1000/019, RP/0/A/1000/025, SR/0/B/2000/003
C.1	RP/0/B/1000/031, Appendix 5 - Agreement Letters
D.1.a	RP/0/A/1000/002, RP/0/A/1000/019, SR/0/A/2000/003
D.1.b	RP/0/A/1000/002, RP/0/A/1000/019, SR/0/A/2000/003, AP/0/A/1700/045
D.1.c	RP/0/A/1000/002, RP/0/A/1000/019, SR/0/A/2000/003, AP/0/A/1700/045
D.1.d	RP/0/A/1000/002, RP/0/A/1000/019, SR/0/A/2000/003, AP/0/A/1700/045
E.1	RP/0/A/1000/002
E.2	Division/Section Directives
E.3	RP/0/A/1000/015 A, RP/0/A/1000/015 B, SR/0/B/2000/004, RP/0/A/1000/001, RP/0/A/1000/002, RP/0/A/1000/019, SR/0/A/2000/003
E.6	EP Functional Area Manual 3.3

FIGURE P-1

DUKE ENERGY COMPANY
OCONEE NUCLEAR STATION

E.7	RP/0/A/1000/024, SR/0/A/2000/003
E.8	RP/0/B/1000/017
F.1.a	RP/0/A/1000/002
F.1.e	Division/Section Directives
F.2	RP/0/A/1000/001
G.3a	SR/0/B/2000/001, RP/0/A/1000/028
H.1	RP/0/A/1000/002
H.4	RP/0/A/1000/002, RP/0/A/1000/019, SR/0/A/2000/003
H.7	HP/0/B/1009/023
H.8	HP/0/B/1009/018, SH/0/B/2005/001, IP/0/B/1601/003
H.12	HP/0/B/1009/023, SH/0/B/2005/002
I.1	RP/0/B/1000/010
I.2	HP/0/B/1009/015, HP/0/B/1009/009, HP/0/B/1009/018, CSM 5.2, CP/1,2,3/A/2002/002, RP/0/B/1000/018, SH/0/B/2005/001, SH/0/B/2005/002, HP/0/B/1009/026
I.3.a	RP/0/A/1000/024, RP/0/A/1000/001, HP/0/B/1009/022
I.3.b	HP/0/B/1009/018, HP/0/B/1009/022, SH/0/B/2005/001
I.4	RP/0/A/1000/001, HP/0/B/1009/018, HP/0/B/1009/022, SH/0/B/2005/001
I.5	RP/0/A/1000/001
I.6	RP/0/A/1000/001, HP/0/B/1009/018, SH/0/B/2005/001
I.7 & 8	SH/0/B/2005/002, HP/0/B/1009/026
I.9	SH/0/B/2005/002, HP/0/B/1009/026

FIGURE P-1

DUKE ENERGY COMPANY
OCONEE NUCLEAR STATION

I.10	HP/0/B/1009/018, SH/0/B/2005/001
J.1	RP/0/A/1000/009
J.2	HP/0/B/1009/016, HP/0/B/1009/018, HP/0/B/1009/009, SH/0/B/2005/001
J.3	HP/0/B/1009/016, HP/0/B/1009/018, RP/0/B/1000/010, Site Directive, SH/0/B/2005/001
J.4	HP/0/B/1009/016, HP/0/B/1009/018, SH/0/B/2005/001
J.5	HP/0/B/1009/009
J.6	Radiation Protection Manual, SH/0/B/2005/003
J.7	RP/0/B/1000/024, SH/0/B/2000/003
J.10.a	Radiation Protection Manual
J.10.e	SH/0/B/2005/003
J.10.m	RP/0/A/1000/024, SR/0/A/2000/003
K.2	RP/0/B/1000/011
K.3.a	Radiation Protection Manual
K.5.a	Radiation Protection Manual
K.5.b	Radiation Protection Manual
K.7	HP/0/B/1009/018, HP/0/B/1009/016, SH/0/B/2005/001
L.2	RP/0/B/1000/016

FIGURE P-1

DUKE ENERGY COMPANY
OCONEE NUCLEAR SITE

L.4	RP/0/B/1000/016
O.1	Oconee Training Division Training Guide ERTG-001
O.2	Oconee Training Division Training Guide ERTG-001
O.3`	Security Training Plan, Oconee Training Division Training Guide ERTG-001
Appendix 4	Evacuation Time Estimates
Appendix 5	Letters of Agreement
Appendix 6	Distribution List
Appendix 7	Data System
Appendix 8	SPCC Plan (Spill Prevention Control And Countermeasure Plan)
Appendix 9	Oconee Nuclear Station Chemical Treatment Ponds 1, 2 and 3, Groundwater Monitoring Sampling And Analysis Plan
Appendix 10	Hazardous Materials Response Plan

FIGURE P-2

DUKE ENERGY COMPANY
OCONEE NUCLEAR SITE

SUPPORTING PLANS

State of South Carolina

Oconee County

Pickens County

DOE-IRAP Plan

INPO-Fixed Facility Agreement

NRC Region II

§50.54(q) Screening Evaluation Form

Activity Description and References: BLOCK 1

See attached sheet for all changes pertaining to this procedure.

Activity Scope: BLOCK 2

☒ The activity is a *change* to the *emergency plan*

☐ The activity is not a *change* to the *emergency plan*

Change Type: BLOCK 3

☐ The change is editorial or typographical

☒ The change is not editorial or typographical

Change Type: BLOCK 4

☐ The change does conform to an activity that has prior approval

☒ The change does not conform to an activity that has prior approval

Planning Standard Impact Determination: BLOCK 5

☐ §50.47(b)(1) – Assignment of Responsibility (Organization Control)

☐ §50.47(b)(2) – Onsite Emergency Organization

☐ §50.47(b)(3) – Emergency Response Support and Resources

☐ **§50.47(b)(4) – Emergency Classification System***

☐ **§50.47(b)(5) – Notification Methods and Procedures***

☐ §50.47(b)(6) – Emergency Communications

☐ §50.47(b)(7) – Public Education and Information

☐ §50.47(b)(8) – Emergency Facility and Equipment

☒ **§50.47(b)(9) – Accident Assessment***

☒ **§50.47(b)(10) – Protective Response***

☐ §50.47(b)(11) – Radiological Exposure Control

☐ §50.47(b)(12) – Medical and Public Health Support

☐ §50.47(b)(13) – Recovery Planning and Post-accident Operations

☐ §50.47(b)(14) – Drills and Exercises

☐ §50.47(b)(15) – Emergency Responder Training

☒ §50.47(b)(16) – Emergency Plan Maintenance

***Risk Significant Planning Standards**

☐ The proposed activity does not impact a Planning Standard

Commitment Impact Determination: BLOCK 6

☐ The activity does involve a site specific EP commitment

Record the commitment or commitment reference: ____________________

☒ The activity does not involve a site specific EP commitment

Results: BLOCK 7

☐ The activity can be implemented without performing a §50.54(q) effectiveness evaluation

☒ The activity cannot be implemented without performing a §50.54(q) effectiveness evaluation

Preparer Name: John Kaminski	Preparer Signature: J Kaminski	Date: 10/7/13
Reviewer Name: Don Crowl	Reviewer Signature: Donald A Crowl / per phone call	Date: 10/14/13

§50.54(q) Effectiveness Evaluation Form

Activity Description and References: ONS Emergency Plan Change 2013-01 BLOCK 1

Activity Type: BLOCK 2

☒ The activity is a *change* to the *emergency plan*
☐ The activity affects implementation of the *emergency plan*, but is not a *change* to the *emergency plan*

Impact and Licensing Basis Determination: BLOCK 3

Licensing Basis:

1. 10CFR50.47 (b) 9 - Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.
2. 10CFR50.47(b)10-A range of protective actions has been developed for the plume exposure pathway EPZ for emergency workers and the public. In developing this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodide (KI), as appropriate. Evacuation time estimates have been developed by applicants and licensees. Licensees shall update the evacuation time estimates on a periodic basis. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.
3. 10CFR50.47(b)16 - Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.
4. 10CFR50 Appendix E.IV.3. Nuclear power reactor licensees shall use NRC approved evacuation time estimates (ETEs) and updates to the ETEs in the formulation of protective action recommendations and shall provide the ETEs and ETE updates to State and local governmental authorities for use in developing offsite protective action strategies.
5. 10CFR50 Appendix E.IV.4. Within 365 days of the later of the date of the availability of the most recent decennial census data from the U.S. Census Bureau or December 23, 2011, nuclear power reactor licensees shall develop an ETE analysis using this decennial data and submit it under § 50.4 to the NRC. These licensees shall submit this ETE analysis to the NRC at least 180 days before using it to form protective action recommendations and providing it to State and local governmental authorities for use in developing offsite protective action strategies.
6. 10CFR50 Appendix E.IV.5. During the years between decennial censuses, nuclear power reactor licensees shall estimate EPZ permanent resident population changes once a year, but no later than 365 days from the date of the previous estimate, using the most recent U.S. Census Bureau annual resident population estimate and State/local government population data, if available. These licensees shall maintain these estimates so that they are available for NRC inspection during the period between decennial censuses and shall submit these estimates to the NRC with any updated ETE analysis.
7. 10CFR50 Appendix E.IV.6. If at any time during the decennial period, the EPZ permanent resident population increases such that it causes the longest ETE value for the 2-mile zone or 5-mile zone, including all affected Emergency Response Planning Areas, or for the entire 10-mile EPZ to increase by 25 percent or 30 minutes, whichever is less, from the nuclear power reactor licensee's currently NRC approved or updated ETE, the licensee shall update the ETE analysis to reflect the impact of that population increase. The licensee shall submit the updated ETE analysis to the NRC under § 50.4 no later than 365 days after the licensee's determination that the criteria for updating the ETE have been met and at least 180 days before using it to form protective action recommendations and providing it to State and local governmental authorities for use in developing offsite protective action strategies.

8. 10CFR50 Appendix E.IV.7. After an applicant for a combined license under part 52 of this chapter receives its license, the licensee shall conduct at least one review of any changes in the population of its EPZ at least 365 days prior to its scheduled fuel load. The licensee shall estimate EPZ permanent resident population changes using the most recent U.S. Census Bureau annual resident population estimate and State/local government population data, if available. If the EPZ permanent resident population increases such that it causes the longest ETE value for the 2-mile zone or 5-mile zone, including all affected Emergency Response Planning Areas, or for the entire 10-mile EPZ, to increase by 25 percent or 30 minutes, whichever is less, from the licensee's currently approved ETE, the licensee shall update the ETE analysis to reflect the impact of that population increase. The licensee shall submit the updated ETE analysis to the NRC for review under § 50.4 of this chapter no later than 365 days before the licensee's scheduled fuel load.
9. NUREG 0654 II.J.8 States, " Each licensee's plan shall contain time estimates for evacuation within the plume exposure EPZ. These shall be in accordance with Appendix 4.
10. NUREG/CR-7002, Criteria for Development of Evacuation Time Estimate Studies.

Compliance Evaluation and Conclusion: BLOCK 4

1. Evaluation:

Sections I and P were updated to ensure the continued accuracy of the plan. These changes are editorial in nature reflecting changes to procedure numbers.

The requirements, criteria and methodology for the development of an ETE is provided within NUREG 7002, Criteria for Development of Evacuation Time Estimate Studies. This new guidance has been endorsed by the NRC. NUREG 7002 was used to develop the ETE being proposed. Some of the key criteria developed in this document include:

- Development of ETEs for the staged evacuation protective action;
- Emphasis on the use of existing emergency preparedness programs when developing the ETE;
- Use of traffic simulation modeling;
- Consideration of shadow evacuations in the analysis;
- Verification of commitment of resources, such as buses and ambulances, etc.;
- Consideration of the evacuation tail; and
- ETE updates.

This guidance document emphasizes the use of existing emergency planning methodology when developing the ETE including:

- Use of existing registration programs for people with disabilities and those with access and functional needs who do not reside in special facilities;
- Modeling of planned or approved evacuation routes;
- Use of approved traffic control plans in the analysis; and
- Use of planned bus routes for analysis of the transit dependent population evacuation.

The 2010 US Census data was obtained and was used throughout the development of the ETE. One of the first steps for developing the ETE, was the development of and conduct of survey of the people within the 10 mile EPZ to gather data pertinent to the ETE. Additionally, more scenarios were considered during the development of the ETE than had been considered in the previous ETE, including one scenario in which a Clemson Football Game was in progress. Newer traffic modeling programming was used, and combined with survey data collected combined to produce a more refined view of the ETE. Additionally, shadow evacuations as described within NUREG 7002 were considered. The ETE was finalized, reviewed, approved and submitted to the NRC in December 2012.

New ETE rules detail the requirement to perform a review annually using changes in population estimated using data from many sources as well as the need to consider any changes in infrastructure. As a result of storms in the spring and summer of 2013, another scenario was added as an addendum due to a failure of the Jones Mill Rd Bridge which was one of the evacuation routes described in the ETE and is considered a change in infrastructure.

All of the above information being input to the analysis resulted in the latest ETE. A comparison of the previous ETE data and the revised ETE can be seen on the attached Table. Thus a conclusion can be easily drawn that the latest ETE, using more scenarios, using the latest census data, using the most up to date traffic modeling program, is therefore in compliance with the newest regulations and rules.

Conclusion:

The proposed activity ☒ does / ☐ does not continue to comply with the requirements.

Reduction in Effectiveness (RIE) Evaluation and Conclusion: BLOCK 5

1. Evaluation:

Sections I and P were updated to ensure the continued accuracy of the plan. These changes are editorial in nature reflecting changes to procedure numbers.

The revised analysis complies with the most recent regulatory requirements. The revised analysis provides a comparison (see attached table) table showing the previous ETE data/information and processes as compared to the current ETE. The current ETE considered many more scenarios than had been previously considered in the previous analysis, those scenarios used updated census data as well as data collected through survey to be more realistic, and finally the data was input to the latest traffic analysis modeling to arrive at the required 90th and 100th percentiles. The revised analysis determined that there is no substantial change in evacuation times. Therefore there is no substantive changes to the ETE, and there is no reduction in the effectiveness of the emergency plan.

Conclusion:

The proposed activity ☐ does / ☒ does not constitute a RIE.

Effectiveness Evaluation Results BLOCK 6

☒ The activity does continue to comply with the requirements of §50.47(b) and §50 Appendix E **and** the activity does not constitute a reduction in effectiveness. Therefore, the activity can be implemented without prior approval.

☐ The activity does not continue to comply with the requirements of §50.47(b) and §50 Appendix E **or** the activity does constitute a reduction in effectiveness. Therefore, the activity cannot be implemented without prior approval.

Preparer Name: John Kaminski	Preparer Signature	Date: 10/7/13
Reviewer Name: Dennis A. Crowl	Reviewer Signature	Date: 10/14/13
Approver Name: Patrick M Street	Approver Signature	Date: 10/15/13

Oconee Emergency Plan revision 2013-01, New Evacuation Time Estimate				
Change #	**Document Number / Page / Section**	**Current Wording**	**Proposed Wording**	**Reason for Change**
1	ONS E Plan / Page J1 / J.1	Site Accountability locations, Figure J-7	Site Accountability locations, Figure J-5	Editorial
2	ONS E Plan / Page J2 / J.3	(Figures J-5, J-6)	(Figures J-3, J-4)	Editorial
3	ONS E Plan / Page J4 / J.8	A description of the methods and assumptions used in developing the evacuation times is included in the 2002 study of the Oconee Nuclear Site prepared by Duke Energy Environmental Health And Safety Services. These estimates will be considered in evaluating protective action recommendations from the Technical Support Center or the Emergency Operations Facility. A copy of the study is available in the Technical Support Center and the Emergency Operations Facility.	A description of the methods and assumptions used in developing the analysis of evacuation time estimates is included in the current Evacuation Time Estimate study for the Oconee Nuclear Site. (ONS-ETE-12142012, Rev. 000; ONS EVACUATION TIME ESTIMATES (ETE) DATED 12/14/2012.) The Evacuation Time Estimates will be considered in evaluating protective action recommendations from the Technical Support Center or the Emergency Operations Facility. A copy of the most recent study is available in the Technical Support Center and the Emergency Operations Facility.	Incorporate newly revised ETE's into the existing Oconee Nuclear Station E Plan.

Oconee Emergency Plan revision 2013-01, New Evacuation Time Estimate				
Change #	**Document Number / Page / Section**	**Current Wording**	**Proposed Wording**	**Reason for Change**
3 Cont.	ONS E Plan / Page J4 / J.8	Figures J-3A through J-4I provide information concerning population (permanent, seasonal and transient). These figures also provide the estimated time for evacuation. See also Appendix 4 for a discussion of the evacuation scenarios covered by the study. Approximately every 10 years after new data becomes available from the U.S. Census Bureau, the data will be reviewed to determine whether the evacuation time estimates need to be updated. The evacuation time estimates will be updated whenever reliable information indicates that significant changes have occurred that would invalidate the current estimates.	An updated ETE analysis will be submitted to the NRC under §50.4 no later than 365 days after ONS determination that the criteria for updating the ETE have been met and at least 180 days before using it to form protective action recommendations and providing it to State and local governmental authorities for use in developing offsite protective action strategies. The criteria for determination that an updated ETE analysis have been met: a. The availability of the most recent decennial census data from the U.S. Census Bureau; OR b. If at any time during the decennial period, the EPZ permanent resident population increases such that it causes the longest ETE value for the 2-mile zone or 5-mile zone, including all affected Emergency Response Planning Areas, or for the entire 10-mile EPZ to increase by 25 percent or 30 minutes, whichever is less, from the currently NRC approved or updated ETE. During the years between decennial censuses ONS will estimate EPZ permanent resident population changes once a year, but no later than 365 days from the date of the previous estimate, using the most recent U.S. Census Bureau annual resident	Eliminated reference to figures which were duplicated from the ETE. Duplication of tables and data from a referenced study is not preferred due to the addition of an error likely situation occurring where one table might be change without changing both. Additionally, new rule requirements were included changing the old requirements.

Oconee Emergency Plan revision 2013-01, New Evacuation Time Estimate				
Change #	**Document Number / Page / Section**	**Current Wording**	**Proposed Wording**	**Reason for Change**
4	ONS E Plan / Page J5 / J.10a	See Figures J-3B through J-4E	See Appendix 4 Evacuation Time Estimates	Eliminated duplicate figures. The actual ETE contains all data table and information and is included by reference. The ET is now considered a part of the ONS E Plan and is available in the TSC and EOF as stated.
5	ONS E Plan / Pages J-11 through J-21	Figures J-3A, J-3B, J-4A, J-4B, J-4C, J-4D, J-4E, J-4F, J-4G, J-4H, J-4I	Deleted	Duplicate to information contained within the ETE
6	ONS E Plan / Pages J-22, 23, 24	Figures J-5, J-6, J-7	Renumbered pages and figures	Editorial
7	ONS E Plan Cover sheet	Rev 2012-05	Rev 2013-01	Revised rev number and date for proposed rev
8	ONS E Plan / List of Effective Pages / 1, 2,	Rev 2012-05	Rev 2013-01	Revised rev number and date for proposed rev
		List of Figures - Rev 2012-05 - December 2012	List of Figures - Rev 2013-01 September 2013	Revised rev number and date for proposed rev
		Record of Changes - Rev 2012-05 - December 2012	Rev 2012-05 - Rev 2013-01 September 2013	Revised rev number and date for proposed rev
		Protective Response- Page J1- J24 Rev 2012-05 - December 2012	Protective Response - Page J1 - J13 Rev 2013-01 September 2013	Revised rev number and date for proposed rev
		Appendix 4 - Evacuation Time Estimates Rev 2012-05 - December 2012	Appendix 4 Evacuation Time Estimates Rev 2013-01 - September 2013	Revised rev number and date for proposed rev
9	ONS E Plan / List of Figures / 4	Figures J-3A, J-3B, J-4A, J-4B, J-4C, J-4D, J-4E, J-4F, J-4G, J-4H, J-4I	Deleted	Revised to be correct for proposed rev
		Figures J-5, J-6, J-7	Renumbered to J-3, J-4, J-5	Revised to be correct for proposed rev
10	Record of Changes / page 4	NA	Rev Number 2013-01 Effective Date 9/13, Section J - Revised to incorporate latest revision to the Evacuation Time Estimate. Eliminated data tables which were duplicate to information contained within the ETE (Appendix 4)	Incorporated latest revision of the ETE and revised rule language
		NA	Appendix 4 - Added latest ETE as a	Incorporated latest revision

Oconee Emergency Plan revision 2013-01, New Evacuation Time Estimate				
Change #	**Document Number / Page / Section**	**Current Wording**	**Proposed Wording**	**Reason for Change**
			reference	of the ETE and revised rule language
11	Appendix 4	Evacuation Time Estimates described in part J of this plan...	The Evacuation Time Estimates (ETEs) for the Oconee Nuclear Station, dated November 2012, KLD Engineering, P.C. Report KLD TR-494, Oconee Nuclear Station, Development of Evacuation Time Estimates, Revision 1, November 2012 was submitted under separate cover and is considered to be incorporated as part of this document by reference. See ONS-ETE-12142012, Rev. 000: ONS EVACUATION TIME ESTIMATES (ETE) DATED 12/14/2012.	Included ETE as a reference.
12	ONS E Plan Section I, Page I-5	Radiation Protections Section Manual 11.7 describes	Procedures HS/0/B/2005/002 and HP/0/B/1009/026 describe	RPSM 11.7 no longer being used. Reference correct procedures.
13	ONS E Plan Section P, pages P-4 thru P-7	• Radiation Protections Section Manual 11.7 • Revised appropriate procedures from B safety classification to A.	• Procedures HS/0/B/2005/002 and HP/0/B/1009/026 • Revised appropriate procedures from B safety classification to A.	RPSM 11.7 no longer being used. Reference correct procedures. Appropriate safety classification of procedures now listed

* *Printed Name and Signature*

§50.54(q) Screening Evaluation Form

Activity Description and References: Oconee Nuclear Station Emergency Plan Volume A revision number 2013-01 — BLOCK 1

Per PIP C-12-6790 the ONS EPLAN, Section D, "Emergency Classification System". The description of the Oconee Emergency Classification System references NRC Bulletin 205-02 and should reference 2005-02.

Per PIP C-12-6790 ONS EPLAN Section D Enclosure 4.6, Page D76 and D-77 - some missing text or extraneous text in the basis discussion. The following information appears to be "orphaned" at the top of page D-77. NORAD, FBI, FAA or NRC. However, the declaration should not be unduly delayed awaiting Federal notification.] (SDP impact - 10 CFR 50.47(b)(16) - no finding). The extraneous wording was deleted

Activity Scope: — BLOCK 2

☒ The activity <u>is</u> a *change* to the *emergency plan*

☐ The activity <u>is not</u> a *change* to the *emergency plan*

Change Type: — BLOCK 3

☒ The change <u>is</u> editorial or typographical

☐ The change <u>is not</u> editorial or typographical

Change Type: — BLOCK 4

☐ The change <u>does</u> conform to an activity that has prior approval

☐ The change <u>does not</u> conform to an activity that has prior approval

Planning Standard Impact Determination: — BLOCK 5

☐ §50.47(b)(1) – Assignment of Responsibility (Organization Control)
☐ §50.47(b)(2) – Onsite Emergency Organization
☐ §50.47(b)(3) – Emergency Response Support and Resources
☐ **§50.47(b)(4) – Emergency Classification System***
☐ **§50.47(b)(5) – Notification Methods and Procedures***
☐ §50.47(b)(6) – Emergency Communications
☐ §50.47(b)(7) – Public Education and Information
☐ §50.47(b)(8) – Emergency Facility and Equipment
☐ **§50.47(b)(9) – Accident Assessment***
☐ **§50.47(b)(10) – Protective Response***
☐ §50.47(b)(11) – Radiological Exposure Control
☐ §50.47(b)(12) – Medical and Public Health Support
☐ §50.47(b)(13) – Recovery Planning and Post-accident Operations
☐ §50.47(b)(14) – Drills and Exercises
☐ §50.47(b)(15) – Emergency Responder Training
☐ §50.47(b)(16) – Emergency Plan Maintenance

***Risk Significant Planning Standards**

☐ The proposed activity does not impact a Planning Standard

Commitment Impact Determination: BLOCK 6

☐ The activity does involve a site specific EP commitment

Record the commitment or commitment reference: ______________________________

☐ The activity does not involve a site specific EP commitment

Results: BLOCK 7

Editorial changes only as a result of audit in support of Compliance Functional Area Manual (CFAM) Directive 3.18, Response to New NRC

☒ The activity can be implemented without performing a §50.54(q) effectiveness evaluation

☐ The activity cannot be implemented without performing a §50.54(q) effectiveness evaluation

Preparer Name:	Preparer Signature	Date:
John Kaminski		October 14, 2013
Reviewer Name:	**Reviewer Signature**	**Date:**
Douglas A. Crowe		10/14/13

§50.54(q) Screening Evaluation Form

BLOCK 1

Activity Description and References: Oconee Nuclear Station Emergency Plan Volume A revision number 2013-01

Per PIP O-13-2872 the ONS EPLAN, Section D, enclosure 4.4 for EAL "Loss of Shutdown Functions" basis required additional clarification. Added the following wording to the basis:

This EAL is met if a reactor trip is required and the manual Reactor Trip function fails. a failure of the manual reactor trip function pushbutton to initiate a reactor trip is indication of a failure of the Reactor Protection System.

BLOCK 2

Activity Scope:

☒ The activity is a *change* to the *emergency plan*

☐ The activity is not a *change* to the *emergency plan*

BLOCK 3

Change Type:

☒ The change is editorial or typographical
☐ The change is not editorial or typographical

BLOCK 4

Change Type:

☐ The change does conform to an activity that has prior approval
☐ The change does not conform to an activity that has prior approval

BLOCK 5

Planning Standard Impact Determination:

☐ §50.47(b)(1) – Assignment of Responsibility (Organization Control)
☐ §50.47(b)(2) – Onsite Emergency Organization
☐ §50.47(b)(3) – Emergency Response Support and Resources
☐ **§50.47(b)(4) – Emergency Classification System***
☐ **§50.47(b)(5) – Notification Methods and Procedures***
☐ §50.47(b)(6) – Emergency Communications
☐ §50.47(b)(7) – Public Education and Information
☐ §50.47(b)(8) – Emergency Facility and Equipment
☐ **§50.47(b)(9) – Accident Assessment***
☐ **§50.47(b)(10) – Protective Response***
☐ §50.47(b)(11) – Radiological Exposure Control
☐ §50.47(b)(12) – Medical and Public Health Support
☐ §50.47(b)(13) – Recovery Planning and Post-accident Operations
☐ §50.47(b)(14) – Drills and Exercises
☐ §50.47(b)(15) – Emergency Responder Training
☐ §50.47(b)(16) – Emergency Plan Maintenance
***Risk Significant Planning Standards**

☐ The proposed activity does not impact a Planning Standard

BLOCK 6

Commitment Impact Determination:

☐ The activity does involve a site specific EP commitment
Record the commitment or commitment reference: ______________________

☐ The activity does not involve a site specific EP commitment

Results: BLOCK 7

This change only adds clarification regarding the intent of the EAL for classification purposes. No changes were made to the NRC endorsed wording of the EAL per NUMARC /NESP 007 "Methodology for Development of Emergency Action Levels" revision 2.

☒ The activity can be implemented without performing a §50.54(q) effectiveness evaluation
☐ The activity cannot be implemented without performing a §50.54(q) effectiveness evaluation

Preparer Name: John Kaminski	Preparer Signature	Date: October 14, 2013
Reviewer Name: Donald A. Crowl	Reviewer Signature	Date: 10/14/13