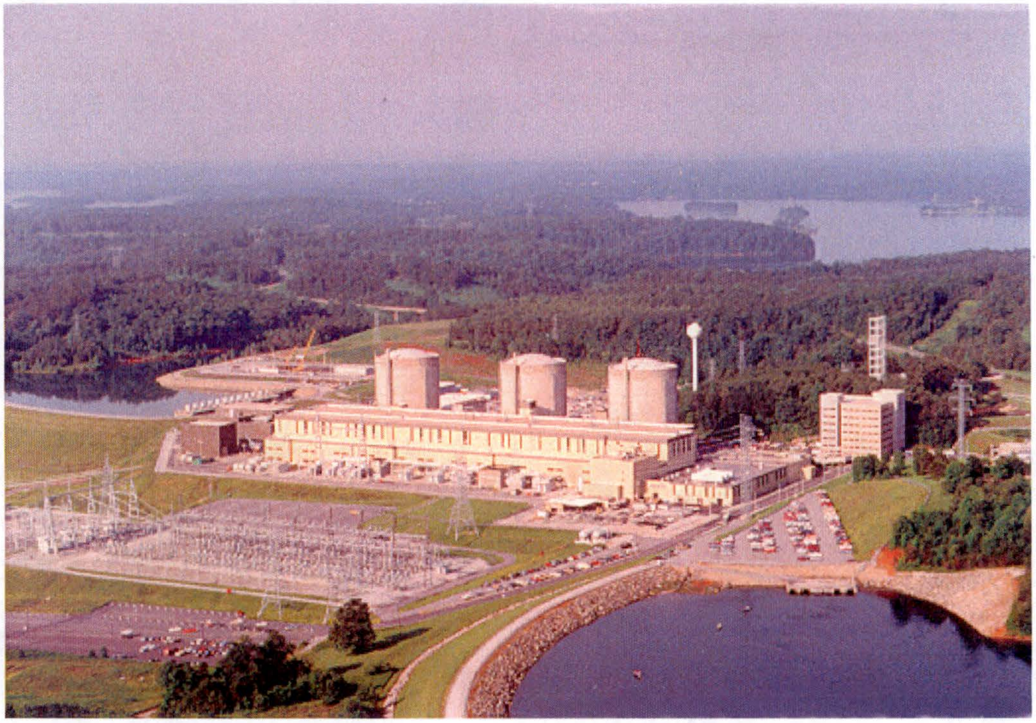
DUKE ENERGY®

OCONEE NUCLEAR STATION
EMERGENCY PLAN



APPROVED:

Thomas D. Ray

Thomas D. Ray

4/20/17

Date Approved

REVISION 2017-002
March, 2017

INSTRUCTION

March 22, 2017

OCONEE NUCLEAR STATION

SUBJECT: Emergency Plan Revision 2017-002

Please make the following changes to the Emergency Plan:

REMOVE	**INSERT**
Cover Sheet Rev 2017-001, March 2017	Cover Sheet Rev 2017-002, March 2017
Rec of Changes Rev 2017-001, March 2017	Rec of Changes Rev 2017-002, March 2017
LOEP Rev 2017-001, March 2017	LOEP Rev 2017-002, March 2017
EPLAN Sec. B, Rev 2016-003, Dec 2016	EPLAN Sec. B, Rev 2017-002, March 2017
EPLAN Sec. D, Rev 2017-001, March 2017	EPLAN Sec. D, Rev 2017-002, March 2017
EPLAN Sec. H, Rev 2016-002, Sept. 2016	EPLAN Sec. H, Rev 2017-002, March 2017
EPLAN Sec. I, Rev 2015-002, March 2015	EPLAN Sec. I, Rev 2017-002, March 2017
EPLAN Sec. P, Rev 2016-003, Dec. 2016	EPLAN Sec. P, Rev 2017-002, March 2017
EPLAN App. 5, Rev 2016-002, Sept. 2016	EPLAN App. 5, Rev 2017-002, March 2017

Pat Street
ONS Emergency Preparedness Manager

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Screening and Evaluation Number	Applicable Sites	
EREG #: 02089586	BNP	☐
	CNS	☐
	CR3	☐
	HNP	☐
5AD #: 02100608	MNS	☐
	ONS	X
	RNP	☐
	GO	☐

Document and Revision	
ONS Emergency Plan 2017-002	Sections B, D, H, I, P and Appendix 5

Part I. Description of Activity Being Reviewed (event or action, or series of actions that may result in a change to the emergency plan or affect the implementation of the emergency plan):

#	Page /Section	Current	Proposed Change
1.	Section B all pages	Rev. 2016-003 December 2016	Revision No. 2017-002 March 2017
2.	Section B page B-4	Oconee Rural Fire Association	Oconee County Emergency Services Fire/Chemical Spill
3.	Section B page B-6	12. Communications Operations Safety Assurance Training Nuclear Assurance	12. Communications Operations Organizational Effectiveness Training
4.	Section D all pages lower right	Revision No. 2017-001 March 2017	Revision No. 2017-002 March 2017
5.	Section D page D-14	Confinement Barrier	Confinement Boundary
6.	Section D p D-13(4.1.7) page D-14	OP/1,2,3/A/1502/000	OP/1,2,3/A/1502/009
7.	Section D Page D-126	Basis-Related Requirements from Appendix R Appendix R to 10 CFR 50, states in part: Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."	*deleted, upgraded ONS Fire Program to be fully NFPA 805 compliant*

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		When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off. Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents. In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in HU4.2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.	
8.	Section D page D-136	Once the Control Room is evacuated, the objective is to establish control of important plant equipment and maintain knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to shutdown the reactor and maintain it shutdown), RCS inventory , and secondary heat removal .	Once the Control Room is evacuated, the objective is to establish control of important plant equipment and maintain knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to shutdown the reactor and maintain it shutdown), RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink).
9.	Section D / page 170 page 167	If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip)	If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor. Depending upon several factors,

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		using a different switch).	the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.
10.	Section D / page 206	This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ES actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs that a HPI (makeup) pump be placed in service to restore and maintain pressurizer level.	This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ES actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs an action to restore and maintain pressurizer level due to exceeding NORMAL MAKEUP CAPABILITY.
11.	Section D / page 166 page 169 page 172	Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.	Note 8: A manual trip action is any Control Room operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.
12.	Section D / page 231 page 179(SU8.1) page 191	Containment pressure > 10 psig with < one full train of containment heat removal system (1 RBS with > 700 gpm spray flow **OR** 2 RBCUs) operating per design for ≥ 15 min. (Note 1)	Containment pressure > 10 psig with < one full train of containment heat removal system (1 RBS with > 700 gpm spray flow **AND** 2 RBCUs) operating per design for ≥ 15 min. (Note 1)
13.	Section H	Rev. 2016-002 September 2016	updated to Rev 2017-002 March 2017
14.	Section H page H-4	H. EMERGENCY FACILITIES AND EQUIPMENT	*inserted page break to move title to top of page H-5*
15.	Section H page H-24 Instrument Type column	Eberline RM14	Eberline RM14 or Ludlum 177
16.	Section H page H-24 Instrument Type column	Eberline RO20	Eberline RO20 or Ludlum 9-3
17.	Section H page H-24 Instrument	Eberline RO7	Eberline RO7 or Ludlum 9-7

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	Type column		
18.	Section H page H-24 Additional Information column	Has alarm setting. Speaker indication. 50 hr operation on fully charged battery.	Has alarm setting. Speaker indication. 50 hr operation on fully charged battery. Ludlum 177 has additional scale, x 1000 = 0-500 kcpm.
19.	Section I	Rev. 2015-002 March 2015	Rev. 2017-002 March 2017
20.	Section I I-1	<u>Emergency Action Level Procedures</u> 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/A/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.	<u>Emergency Action Level Procedures</u> 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/A/1000/001) identifies plant parameters that can be used to determine emergency situations that require activation of the station emergency plan. Revision 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 (ADAMS Accession Number ML12326A805). ONS conducted an EAL implementation upgrade project that produced the EALs, see BASIS document Section D.
21.	Section I 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/A/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	Source Term of Releases of Radioactive Material within Plant Systems Operations (Control Room Personnel) will use the EAL Wallchart of RP/0/A/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 AD-EP-ALL-0202. AD-EP-ALL-0202 uses release paths of unit vents and the main steam relief valves. Assumptions for the calculations are based on the following:

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		procedure AD-EP-ALL-0202. AD-EP-ALL-0202 uses release paths of unit vents and the main steam relief valves. Assumptions for the calculations are based on the following:	
22.	Section P all pages	Rev. 2016-003 December 2016	Revision No. 2017-002 March 2017
23.	Section P page P-1 P.2 &P.3	The Manager of Emergency Preparedness 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.	The Manager of Emergency Preparedness 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 Organizational Effectiveness Group.
24.	Section P page P-16 O4.10	OCI507-N Appendix R Training	*deleted, ONS fully compliant with NFPA 805*
25.	Section P B7a-d C1a,b,c G3a,b G4a,b,c H4 M3	RP/0/A/1000/031 Joint Information Center Emergency Response Plan	AD-EP-ALL-0108 Activation and Operation of the Near-Site Joint Information Center
26.	Appendix 5 page 1	• DELETED - Dominion Nuclear Connecticut, Inc. (DNC) Superseded by letter from GO RP	*deleted*
27.	Appendix 5 page 2	• Safe Industries	*deleted*

Part II. Activity Previously Reviewed?	Yes	☐	No	X
Is this activity Fully bounded by an NRC approved 10 CFR 50.90 submittal or Alert and Notification System Design Report? If yes, identify bounding source document number or approval reference and ensure the basis for concluding the source document fully bounds the proposed change is documented below: Justification:	10 CFR 50.54(q) Effectiveness Evaluation is not required. Enter justification below and complete Attachment 4, Part V.		Continue to Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part III	

<< 10 CFR 50.54(q) Screening Evaluation Form >>

Bounding document attached (optional)				☐
Part III. Editorial Change	Yes	☐	No	X
Is this activity an editorial or typographical change only, such as formatting, paragraph numbering, spelling, or punctuation that does not change intent? Justification:	10 CFR 50.54(q) Effectiveness Evaluation is not required. Enter justification and complete Attachment 4, Part V & VI.		Continue to Attachment 4, Part IV and address non editorial changes	

Part IV. Emergency Planning Element and Function Screen (Reference Attachment 1, Considerations for Addressing Screening Criteria) Does this activity involve any of the following, including program elements from NUREG-0654/FEMA REP-1 Section II? If answer is yes, then check box.		
1	10 CFR 50.47(b)(1) Assignment of Responsibility (Organization Control)	
1a	Responsibility for emergency response is assigned.	☐
1b	The response organization has the staff to respond and to augment staff on a continuing basis (24-7 staffing) in accordance with the emergency plan.	☐
2	10 CFR 50.47(b)(2) Onsite Emergency Organization	
2a	Process ensures that onshift emergency response responsibilities are staffed and assigned	X
2b	The process for timely augmentation of onshift staff is established and maintained.	☐
3	10 CFR 50.47(b)(3) Emergency Response Support and Resources	
3a	Arrangements for requesting and using off site assistance have been made.	☐
3b	State and local staff can be accommodated at the EOF in accordance with the emergency plan. (NA for CR3)	☐
4	10 CFR 50.47(b)(4) Emergency Classification System	
4a	A standard scheme of emergency classification and action levels is in use. (Requires final approval of Screen and Evaluation by EP CFAM.)	X
5	10 CFR 50.47(b)(5) Notification Methods and Procedures	
5a	Procedures for notification of State and local governmental agencies are capable of initiating notification of the declared emergency within 15 minutes (60 minutes for CR3) after declaration of an emergency and providing follow-up notification.	☐
5b	Administrative and physical means have been established for alerting and providing prompt instructions to the public within the plume exposure pathway. (NA for CR3)	☐
5c	The public ANS meets the design requirements of FEMA-REP-10, Guide for Evaluation of Alert and Notification Systems for Nuclear Power Plants, or complies with the licensee's FEMA-approved ANS design report and supporting FEMA approval letter. (NA for CR3)	☐

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Part IV. Emergency Planning Element and Function Screen (cont.)		
6	10 CFR 50.47(b)(6) Emergency Communications	
6a	Systems are established for prompt communication among principal emergency response organizations.	☐
6b	Systems are established for prompt communication to emergency response personnel.	☐
7	10 CFR 50.47(b)(7) Public Education and Information	
7a	Emergency preparedness information is made available to the public on a periodic basis within the plume exposure pathway emergency planning zone (EPZ). (NA for CR3)	☐
7b	Coordinated dissemination of public information during emergencies is established.	☐
8	10 CFR 50.47(b)(8) Emergency Facilities and Equipment	
8a	Adequate facilities are maintained to support emergency response.	☐
8b	Adequate equipment is maintained to support emergency response.	X
9	10 CFR 50.47(b)(9) Accident Assessment	
9a	Methods, systems, and equipment for assessment of radioactive releases are in use.	X
10	10 CFR 50.47(b)(10) Protective Response	
10a	A range of public PARs is available for implementation during emergencies. (NA for CR3)	☐
10b	Evacuation time estimates for the population located in the plume exposure pathway EPZ are available to support the formulation of PARs and have been provided to State and local governmental authorities. (NA for CR3)	☐
10c	A range of protective actions is available for plant emergency workers during emergencies, including those for hostile action events.	☐
10d	KI is available for implementation as a protective action recommendation in those jurisdictions that chose to provide KI to the public.	☐
11	10 CFR 50.47(b)(11) Radiological Exposure Control	
11a	The resources for controlling radiological exposures for emergency workers are established.	☐
12	10 CFR 50.47(b)(12) Medical and Public Health Support	
12a	Arrangements are made for medical services for contaminated, injured individuals.	☐
13	10 CFR 50.47(b)(13) Recovery Planning and Post-accident Operations	
13a	Plans for recovery and reentry are developed.	☐
14	10 CFR 50.47(b)(14) Drills and Exercises	
14a	A drill and exercise program (including radiological, medical, health physics and other program areas) is established.	☐
14b	Drills, exercises, and training evolutions that provide performance opportunities to develop, maintain, and demonstrate key skills are assessed via a formal critique process in order to identify weaknesses.	☐
14c	Identified weaknesses are corrected.	☐
15	10 CFR 50.47(b)(15) Emergency Response Training	
15a	Training is provided to emergency responders.	☐

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Part IV. Emergency Planning Element and Function Screen (cont.)		
16	10 CFR 50.47(b)(16) Emergency Plan Maintenance	
16a	Responsibility for emergency plan development and review is established.	X
16b	Planners responsible for emergency plan development and maintenance are properly trained.	☐

PART IV. Conclusion If no Part IV criteria are checked, a 10 CFR 50.54(q) Effectiveness Evaluation is not required, then complete Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part V. Go to Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part VI for instructions describing the NRC required 30 day submittal.	☐
If any Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part IV criteria are checked, then complete Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part V and perform a 10 CFR 50.54(q) Effectiveness Evaluation. Shaded block requires final approval of Screen and Evaluation by EP CFAM.	X

Part V. Signatures:		
Preparer Name (Print): Pete Kuhlman	Preparer Signature: Pete J. Kuhlman	Date: 3/22/17
Reviewer Name (Print): Don Crowl	Reviewer Signature:	Date: 3-22-17
Approver (EP Manager Name (Print): Pat Street	Approver Signature:	Date: 3/22/17
Approver (CFAM, as required) Name (Print) John Overly	Approver Signature: SIGNATURE IN CAS	Date:

Part VI. NRC Emergency Plan and Implementing Procedure Submittal Actions

Create two EREG General Assignments.	
• One for EP to provide the 10 CFR 50.54(q) summary of the analysis, or the completed 10 CFR 50.54(q), to Licensing.	☐
• One for Licensing to submit the 10 CFR 50.54(q) information to the NRC within 30 days after the change is put in effect.	☐

QA RECORD

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Screening and Evaluation Number	Applicable Sites	
EREG #: 02089586	BNP	☐
	CNS	☐
	CR3	☐
	HNP	☐
5AD #: 02100608	MNS	☐
	ONS	X
	RNP	☐
	GO	☐
Document and Revision ONS Emergency Plan 2017-002	Sections B, D, H, I, P and Appendix 5	

Part I. Description of Proposed Change:

#	Page /Section	Current	Proposed Change
1.	Section B all pages	Rev. 2016-003 December 2016	Revision No. 2017-002 March 2017
2.	Section B page B-4	Oconee Rural Fire Association	Oconee County Emergency Services Fire/Chemical Spill
3.	Section B page B-6	12. Communications Operations Safety Assurance Training Nuclear Assurance	12. Communications Operations Organizational Effectiveness Training
4.	Section D all pages lower right	Revision No. 2017-001 March 2017	Revision No. 2017-002 March 2017
5.	Section D page D-14	Confinement Barrier	Confinement Boundary
6.	Section D p D-13(4.1.7) page D-14	OP/1,2,3/A/1502/000	OP/1,2,3/A/1502/009
7.	Section D Page D-126	Basis-Related Requirements from Appendix R Appendix R to 10 CFR 50, states in part: Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions." When considering the effects of fire,	*deleted, upgraded ONS Fire Program to be fully NFPA 805 compliant*

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		those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off. Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents. In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in HU4.2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.	
8.	Section D page D-136	Once the Control Room is evacuated, the objective is to establish control of important plant equipment and maintain knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to shutdown the reactor and maintain it shutdown), RCS inventory , and secondary heat removal .	Once the Control Room is evacuated, the objective is to establish control of important plant equipment and maintain knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to shutdown the reactor and maintain it shutdown), RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink).
9.	Section D / page 170 page 167	If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip) using a different switch).	If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor. Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a

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			subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.
10.	Section D / page 206	This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ES actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs that a HPI (makeup) pump be placed in service to restore and maintain pressurizer level.	This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ES actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs an action to restore and maintain pressurizer level due to exceeding NORMAL MAKEUP CAPABILITY.
11.	Section D / page 166 page 169 page 172	Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.	Note 8: A manual trip action is any Control Room operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.
12.	Section D / page 231 page 179(SU8.1) page 191	Containment pressure > 10 psig with < one full train of containment heat removal system (1 RBS with > 700 gpm spray flow **OR** 2 RBCUs) operating per design for ≥ 15 min. (Note 1)	Containment pressure > 10 psig with < one full train of containment heat removal system (1 RBS with > 700 gpm spray flow **AND** 2 RBCUs) operating per design for ≥ 15 min. (Note 1)
13.	Section H	Rev. 2016-002 September 2016	updated to Rev 2017-002 March 2017
14.	Section H page H-4	H. EMERGENCY FACILITIES AND EQUIPMENT	*inserted page break to move title to top of page H-5*
15.	Section H page H-24 Instrument Type column	Eberline RM14	Eberline RM14 or Ludlum 177
16.	Section H page H-24 Instrument Type column	Eberline RO20	Eberline RO20 or Ludlum 9-3
17.	Section H page H-24 Instrument Type column	Eberline RO7	Eberline RO7 or Ludlum 9-7
18.	Section H page H-24 Additional Information column	Has alarm setting. Speaker indication. 50 hr operation on fully charged battery.	Has alarm setting. Speaker indication. 50 hr operation on fully charged battery. Ludlum 177 has additional scale, x 1000 = 0-500 kcpm.
19.	Section I	Rev. 2015-002 March 2015	Rev. 2017-002 March 2017
20.	Section I	Emergency Action Level Procedures	Emergency Action Level Procedures

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	I-1	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/A/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.	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/A/1000/001) identifies plant parameters that can be used to determine emergency situations that require activation of the station emergency plan. Revision 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 (ADAMS Accession Number ML12326A805). ONS conducted an EAL implementation upgrade project that produced the EALs, see BASIS document Section D.
21.	Section I 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/A/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 AD-EP-ALL-0202. AD-EP-ALL-0202 uses release paths of unit vents and the main steam relief valves. Assumptions for the calculations are based on the following:	Source Term of Releases of Radioactive Material within Plant Systems Operations (Control Room Personnel) will use the EAL Wallchart of RP/0/A/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 AD-EP-ALL-0202. AD-EP-ALL-0202 uses release paths of unit vents and the main steam relief valves. Assumptions for the calculations are based on the following:
22.	Section P all pages	Rev. 2016-003 December 2016	Revision No. 2017-002 March 2017
23.	Section P page P-1 P.2 &P.3	The Manager of Emergency Preparedness 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	The Manager of Emergency Preparedness 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

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		exercise, drills and training sessions to test the Oconee Nuclear Site Emergency Plan. This person is employed in the Safety Assurance Group.	sessions to test the Oconee Nuclear Site Emergency Plan. This person is employed in the Organizational Effectiveness Group.
24.	Section P page P-16 O4.10	OCI507-N Appendix R Training	*deleted*
25.	Section P B7a-d C1a,b,c G3a,b G4a,b,c H4 M3	RP/0/A/1000/031 Joint Information Center Emergency Response Plan	AD-EP-ALL-0108 Activation and Operation of the Near-Site Joint Information Center
26.	Appendix 5 page 1	• DELETED - Dominion Nuclear Connecticut, Inc. (DNC) Superseded by letter from GO RP	*deleted*
27.	Appendix 5 page 2	• Safe Industries	*deleted*

Attachment 6, 10 CFR 50.54(q) Initiating Condition (IC) and Emergency Action Level (EAL) and EAL Bases Validation and Verification (V&V) Form , is attached (required for IC or EAL change)	Yes X No ☐

Part II. Description and Review of Licensing Basis Affected by the Proposed Change:

ONS Emergency Plan rev 2017-001

Section B

B.8 Contractor/Private Organizations - Technical Assistance

Private Contractors and/or companies that would be available to augment and support the emergency organization:

B.9 Local Agency Support

Agreements with local agency support groups have been made to assist the Oconee Nuclear Site during emergency situations.

Section D

Revision 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 (ADAMS Accession Number ML12326A805) (ref. 4.1.1), ONS conducted an EAL implementation upgrade project that produced the EALs discussed herein.

Section H

H.5.b Radiological Monitors (H-5)

Portable Monitors - sufficient numbers are available for use in assessing radiological conditions. (Figure H 6).

Section I

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

<< 10 CFR 50.54(q) Effectiveness Evaluation Form >>

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/A/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.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/A/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 AD-EP-ALL-0202. AD-EP-ALL-0202 uses release paths of unit vents and the main steam relief valves. Assumptions for the calculations are based on the following

Section P

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)

ONS Emergency Plan. 82-2

Section B

B.8 Contractor/Private Organizations - Technical Assistance

Private Contractors and/or companies that would be available to augment and support the emergency organization is as follows:

B.9 Local Agency Support

Agreement with local agency support groups have been made to assist the Oconee Nuclear Site during emergency situations.

Section D

Rev. 2017-001 implemented a standard emergency classification and action level scheme based on NEI 99-01 rev 6 and was SER approved by the NRC.

Section H

H.5.b Radiological Monitors (H-5)

Portable Monitors - sufficient numbers are available for use in assessing radiological conditions. Figure H 9.

Section I

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

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

Station Directive 3.8.5 is a control room procedure that is used to determine offsite dose projection by determining the relationship between the Containment Building RIA-4 monitor readings and the radioactive material available for release from containment. This is a four hour projection and can also be used if the instrumentation used for assessment are offscale or inoperable. (Figure I-1)

Section P

P.7 Implementing Procedures

All implementing procedures are indexed and cross referenced to the section applicable in NUREG 0654 Figure P 2. All implementing procedure are in a separate binder. Figure P-3 denotes those individuals who will receive Emergency Plans and/or Implementing Procedures.

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RIS 2005-02, Clarifying the Process for Making Emergency Plan Changes, Rev. 1 states,

2) Emergency plan

a) The document prepared and maintained by the licensee that identifies and describes the licensee's methods for maintaining emergency preparedness, and responding to emergencies. An emergency plan includes the plan as originally approved by the NRC, and all subsequent changes made by the licensee with, and without, prior NRC review and approval under 10 CFR 50.54(q).

i) The licensee's emergency plan consists of:

(1) The emergency plan as approved by the NRC via a Safety Evaluation Report (SE), or license amendment (LA) from the Office of Nuclear Reactor Regulation (NRR) or the Office of Federal and State Materials and Environmental Management Programs (FSME).

(2) Any subsequent changes to the emergency plan explicitly reviewed by the NRC through an SE, or LA from NRR or FSME, and found to meet the applicable regulations.

(3) Any subsequent changes made by the licensee without NRC review and approval after the licensee concluded that the change(s) do not constitute a decrease in effectiveness under 10 CFR 50.54(q).

The differences in ONS Emergency Plan, Revision 82-2, and the current revision of the ONS Emergency Plan have been determined to meet the regulatory requirements of the course of revisions. Each revision has been evaluated in the regulatory process has met the approval of the NRC during the inspection process. The changes do not reduce the effectiveness of the Emergency Plan.

Part III. Description of How the Proposed Change Complies with Regulation and Commitments.

If the emergency plan, modified as proposed, no longer complies with planning standards in 10 CFR 50.47(b) and the requirements in Appendix E to 10 CFR Part 50, then ensure the change is rejected, modified, or processed as an exemption request under 10 CFR 50.12, Specific Exemptions, rather than under 10 CFR 50.54(q):

10CFR50.47(b)(2) On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available and the interfaces among various onsite response activities and offsite support and response activities are specified.

Changes 1, 2, 3, 23, 25, 26 and 27 updates revision number and titles due to organizational changes. Appendix 5 letters that had been deleted in previous revisions but kept for reference were removed from the list of letters. The Letter of Agreement for SAFE industries is not required and was removed. SAFE Industries is a contract supplier for Beyond Design Basis Events (BDBE) and Fleet Supply Chain maintains contracts for resources needed for Beyond Design Basis Events SAFE Industries is also listed as a resource in paragraph B.8, if needed. The Emergency Plan continues to specify interfaces among support and response activities.

10CFR50.47(b)(4) A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.

Changes 4,5,6,7,8,9,10,11,12 are 4) changes the revision number, 5) Confinement Barrier updated to Boundary to align with terms used throughout the EAL scheme. 6) Procedure number was corrected 7) ONS has upgraded fire protection the NFPA 805 with the full implementation of the Protected Service Water (PSW) project, all references to Appendix 'R' are being removed.8) Enhanced the basis based on operations feedback to add examples of RCS inventory to cool the core and maintaining a heat sink to maintain secondary heat removal, no change in intent. 9)

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removed the bracketed reference to initiating a manual trip using a different switch, ONS does not have a second trip switch in the control room. Removed as technically inaccurate, does not change the intent as the manual action was left in the EAL and basis.10) Replaced starting a second HPI pump as the initiator with initiating actions due to exceeding NORMAL MAKEUP CAPABILITY, *Flow from one HPI pump through HP-120 with letdown isolated*, to align with the intent of the IC. ONS has procedural guidance to bypass the normal makeup valve and have RCS makeup in excess of 300gpm prior to starting an additional HPI pump.11) Enhanced the note to add verbiage in basis stating that the manual actions were performed in the control room, this information will now be available to the performer using only the wall board and will avoid confusion and save time by not having to reference the basis document to validate. No change of intent. 12) Corrected a technical inaccuracy, the EAL states 'one full train of containment heat removal' the parenthesis incorrectly stated one train as 'OR' instead of 'AND' by Tech Specs and the FSAR. Section D continues to maintain a standard emergency classification and action level scheme using the guidance provided in NEI 99-01 rev 6 and the NRC approved SER for implementation on 3/31/17.

10CFR50.47(b)(8) Adequate emergency facilities and equipment to support the emergency response are provided and maintained.

Changes 13 and 14 updates revision numbers and corrects heading format to top of the page. Changes 15,16 17 and 18 added available instruments due to standardization throughout the fleet and retirement of older models. Evaluation of added instruments were comparable to existing equipment.

	Instrument Type	Response Time	Detector Type	Ranges	Radiation Detected	Tube Saturation	Additional Information
Current instrument:	Eberline RM-14	2.2-22 seconds variable	Halogen quenched GM	x 1 = 0-500 cpm x 10 = 0-5000 cpm x 100 = 0-50,000 cpm	Beta & Gamma	Indicates offscale	Has alarm setting. Speaker indication. 50 hour operation on fully charged battery.
New RM-14 equivalent:	Ludlum 177	2.2-22 seconds variable	Halogen quenched GM	x 1 = 0-500 cpm x 10 = 0-5000 cpm x 100 = 0-50,000 cpm x 1000 = 0-500,000 cpm	Beta & Gamma	Indicates offscale	Has alarm setting. Speaker indication. 50 hour operation on fully charged battery.
Recommended entry for Figure H-6, page H-24:	Eberline RM-14 or Ludlum 177	2.2-22 seconds variable	Halogen quenched GM	x 1 = 0-500 cpm x 10 = 0-5000 cpm x 100 = 0-50,000 cpm	Beta & Gamma	Indicates offscale	Has alarm setting. Speaker indication. 50 hour operation on fully charged battery. Ludlum 177 has additional

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							scale, x 1000 = 0-500 kcpm.
Current instrument:	Eberline RO-20	5 seconds	Ion-chamber Air filed, Vented to atmosphere	0-50 Rad/hr	Beta & Gamma	Indicates offscale	Has battery check information
New RO-20 equivalent	Ludlum 9-3	5 seconds	Ion-chamber Air filed, Vented to atmosphere	0-50 Rad/hr	Beta & Gamma	Indicates offscale	Has battery check information
Recommended entry for Figure H-6, page H-24:	Eberline RO-20 or Ludlum 9-3	5 seconds	Ion-chamber Air filed, Vented to atmosphere	0-50 Rad/hr	Beta & Gamma	Indicates offscale	Has battery check information
Current instrument:	Eberline RO-7	Variable	Air filled ion chamber	Med range: 0.1-199.9 Rad/hr High range: 0.01-19,900 Rad/hr	Beta & Gamma	Indicates over range	Digital ion chamber with cables to extend detector up to 60" away or under water
New RO-7 equivalent	Ludlum 9-7	Variable	Air filled ion chamber	Med range: 0.1-199.9 Rad/hr High range: 0.01-19,900 Rad/hr	Beta & Gamma	Indicates over range	Digital ion chamber with cables to extend detector up to 60" away or under water
Recommended entry for Figure H-6, page H-24	Eberline RO-7 or Ludlum 9-7	Variable	Air filled ion chamber	Med range: 0.1-199.9 Rad/hr High range: 0.01-19.9 kRad/hr	Beta & Gamma	Indicates over range	Digital ion chamber with cables to extend detector up to 60" away or under water

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.

Changes 19, 20 and 21 updates revision number, and updates references to Section D, the EAL scheme upgrade to NEI 99-01 rev 6, which was implemented in rev 2017-001 of the ONS Emergency Plan. Operations personnel will use a wallchart instead of a procedure to classify based on events, Radiation Indicating Alarms (RIA) and setpoints required for assessing and classification are located on the wallcharts. No change in intent.

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10CFR50.47(b)(16) Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.

Changes 22, 23, 24 and 25 updates revision numbers, title changes and procedure references due to organizational changes. Deletes the reference for Appendix 'R' training that is no longer applicable due to the implementation of NFPA 805 standards for Fire Protection. Emergency Response Training Guide (ERTG-1) was evaluated for the change to remove Appendix R training under a separate EREG. Site procedure references were also corrected that were superseded by fleet procedures. No change in intent and procedural references to the Emergency Plan are still maintained.

10 CFR Appendix E to Part 50

B. Assessment Actions

1. The means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within and outside the site boundary to protect health and safety. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite and offsite monitoring. By June 20, 2012, for nuclear power reactor licensees, these action levels must include hostile action that may adversely affect the nuclear power plant. The initial emergency action levels shall be discussed and agreed on by the applicant or licensee and state and local governmental authorities, and approved by the NRC. Thereafter, emergency action levels shall be reviewed with the State and local governmental authorities on an annual basis.

E. Emergency Facilities and Equipment

Adequate provisions shall be made and described for emergency facilities and equipment, including:

1. Equipment at the site for personnel monitoring;
2. Equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment;

G. Maintaining Emergency Preparedness

Provisions to be employed to ensure that the emergency plan, its implementing procedures, and emergency equipment and supplies are maintained up to date shall be described.

The changes identified in this revision continue to comply with regulation. ONS continues to specify interfaces among support and response activities. ONS retains the capability required by regulation to assess, monitor and classify within 15 minutes of meeting an EAL threshold and declaring an emergency. ONS continues to maintain adequate emergency facilities and equipment to support the emergency response. ONS continues to maintain the Emergency Plan.

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Part IV. Description of Emergency Plan Planning Standards, Functions and Program Elements Affected by the Proposed Change (Address each function identified in Attachment 4, 10 CFR 50.54(q) Screening Evaluation Form, Part IV of associated Screen):

10CFR50.47(b)(2) On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available and the interfaces among various onsite response activities and offsite support and response activities are specified.

Two emergency planning **functions** have been defined for this planning standard:

(1) The process ensures that onshift emergency response responsibilities are staffed and assigned.

(2) The process for timely augmentation of onshift staff is established and maintained.

NUREG 0654

B.8 Each licensee shall specify the contractor and private organizations who may be requested to provide technical assistance to and augmentation of the emergency organization.

B.9 9. Each licensee shall identify the services to be provided by local agencies for handling emergencies, e.g., police, ambulance, medical, hospital, and fire-fighting organizations shall be specified. The licensee shall provide for transportation and treatment of injured personnel who may also be contaminated. Copies of the arrangements and agreements reached with contractor, private, and local support agencies shall be appended to the plan. The agreements shall delineate the authorities, responsibilities, and limits on the actions of the contractor, private organization, and local services support groups.

10CFR50.47(b)(4) A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.

The following emergency planning **function** has been defined for this planning standard:

A standard scheme of emergency classification and action levels is in use.

NUREG 0654

D.1 An emergency classification and emergency action level scheme as set forth in Appendix 1 must be established by the licensee. The specific instruments, parameters or equipment status shall be shown for establishing each emergency class, in the in-plant emergency procedures. The plan shall identify the parameter values and equipment status for each emergency class.

10CFR50.47(b)(8) Adequate emergency facilities and equipment to support the emergency response are provided and maintained.

Two emergency planning **functions** have been defined for this planning standard:

(1) Adequate facilities are maintained to support emergency response.

(2) Adequate equipment is maintained to support emergency response.

NUREG 0654

H.6 Each licensee shall make provision to acquire data from or for emergency access to offsite monitoring and analysis equipment including: (b) b. radiological monitors including ratemeters and sampling devices. Dosimetry shall be provided and shall meet, as a minimum, the NRC Radiological Assessment Branch Technical Position for the Environmental Radiological Monitoring Program; and....

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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.

The following emergency planning **function** has been defined for this planning standard:

Methods, systems, and equipment for assessment of radioactive releases are in use.

NUREG 0654

I.1 Each licensee shall identify plant system and effluent parameter values characteristic of a spectrum of off-normal conditions and accidents, and shall identify the plant parameter values or other information which correspond to the example initiating conditions of Appendix 1. Such parameter values and the corresponding emergency class shall be included in the appropriate facility emergency procedures. Facility emergency procedures shall specify the kinds of instruments being used and their capabilities.

I.3 3. Each licensee shall establish methods and techniques to be used for determining:

a. the source term of releases of radioactive material within plant systems. An example is the relationship between the containment radiation monitor(s) reading(s) and radioactive material available for release from containment.

b. the magnitude of the release of radioactive materials based on plant system parameters and effluent monitors.

10CFR50.47(b)(16) Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.

Two emergency planning **functions** have been defined for this planning standard:

(1) Responsibility for emergency plan development and review is established.

(2) Planners responsible for emergency plan development and maintenance are properly trained.

NUREG 0654

P.7 Each plan shall contain as an appendix listing, by title, procedures required to implement the plan. The listing shall include the section(s) of the plan to be implemented by each procedure

Part V. Description of Impact of the Proposed Change on the Effectiveness of Emergency Plan Functions:

10CFR50.47(b)(2) On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available and the interfaces among various onsite response activities and offsite support and response activities are specified.

Two emergency planning functions have been defined for this planning standard:

(1) The process ensures that onshift emergency response responsibilities are staffed and assigned.

(2) The process for timely augmentation of onshift staff is established and maintained.

The Emergency Plan changes do not change the methods or resources but changed titles due to organizational changes that were not captured in earlier revisions. The deletion of the Letter of Agreement does not affect the available resources but are maintained on a fleet level, referenced by contract for events beyond design bases.

10CFR50.47(b)(4) A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.

The following emergency planning function has been defined for this planning standard:

A standard scheme of emergency classification and action levels is in use.

The Emergency Plan continues to maintain a standard scheme of emergency classification and action level use, the changes corrected minor technical inaccuracies identified during the training on the new EAL scheme introduced in EPLAN rev 2017-001 but did not change the intent of any EALs. Enhancements to the basis to clarify

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the intent and aid in timely classification.

10CFR50.47(b)(8) Adequate emergency facilities and equipment to support the emergency response are provided and maintained.

Two emergency planning functions have been defined for this planning standard:

(1) Adequate facilities are maintained to support emergency response.

(2) Adequate equipment is maintained to support emergency response.

Emergency Plan added equipment that has been evaluated as comparable existing equipment to allow for standardization of equipment throughout the fleet and replace outdated equipment which ensures adequate equipment available to support emergency response.

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.

The following emergency planning function has been defined for this planning standard:

Methods, systems, and equipment for assessment of radioactive releases are in use.

The methods used in this function reference the assessment used in the standard scheme for emergency classification and action level use referenced in Section D, Emergency Classification. The same equipment used for assessment using procedures supporting the NUMARC model have been updated to reflect the use of the those instruments which are now procedurally located on a wallchart for easier tracking, identification and confirmation of meeting or exceeding an EAL threshold under the NEI 99-01 rev 6 methodology.

10CFR50.47(b)(16) Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.

Two emergency planning functions have been defined for this planning standard:

(1) Responsibility for emergency plan development and review is established.

(2) Planners responsible for emergency plan development and maintenance are properly trained.

The changes reflected in the Emergency Plan were due to maintenance of the Plan due to implementation of NFPA 805 to replace Appendix R, procedural title changes as well as organizational title changes

The changes identified in this revision maintain the effectiveness of the identified Emergency Plan functions.

Part VI. Evaluation Conclusion.

Answer the following questions about the proposed change.

1	Does the proposed change comply with 10 CFR 50.47(b) and 10 CFR 50 Appendix E?	Yes X	No ☐
2	Does the proposed change maintain the effectiveness of the emergency plan (i.e., no reduction in effectiveness)?	Yes X	No ☐
3	Does the proposed change maintain the current Emergency Action Level (EAL) scheme?	Yes X	No ☐
4	Choose one of the following conclusions:		
a	The activity does continue to comply with the requirements of 10 CFR 50.47(b) and 10 CFR 50, Appendix E, and the activity does not constitute a reduction in effectiveness or change in the current Emergency Action Level (EAL) scheme. Therefore, the activity can be implemented without prior NRC approval.		X
b	The activity does not continue to comply with the requirements of 10 CFR 50.47(b) or 10 CFR 50 Appendix E or the activity does constitute a reduction in effectiveness or EAL scheme change. Therefore, the activity cannot be implemented without prior NRC approval.		☐

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Part VII. Disposition of Proposed Change Requiring Prior NRC Approval		
Will the proposed change determined to require prior NRC approval be either revised or rejected?	Yes ☐	No ☐
If No, then initiate a License Amendment Request in accordance 10 CFR 50.90 and AD-LS-ALL-0002, Regulatory Correspondence, and include the tracking number:____.		

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Part VIII. Signatures: EP CFAM Final Approval is required for changes affecting risk significant planning standard 10 CFR 50.47(b)(4).		
Preparer Name (Print): Peter Kuhlman	Preparer Signature: Peter J. Kuhlman	Date: 3/22/17
Reviewer Name (Print): Don Crowl	Reviewer Signature:	Date: 3-22-17
Approver (EP Manager) Name (Print): Pat Street	Approver Signature:	Date: 3/22/17
Approver (CFAM, as required) Name (Print): John Overly	Approver Signature: SIGNATURE IN OAS	Date:

If the proposed activity is a change to the E-Plan or implementing procedures, then create two EREG General Assignments.	
• One for EP to provide the 10 CFR 50.54(q) summary of the analysis, or the completed 10 CFR 50.54(q), to Licensing.	☐
• One for Licensing to submit the 10 CFR 50.54(q) information to the NRC within 30 days after the change is put in effect.	☐

QA RECORD

<< 10 CFR 50.54(q) Initiating Condition (IC) and Emergency Action Level (EAL) and EAL Bases Validation and Verification (V&V) Form >>

Screening or Evaluation Number: **02089586**

Part I. Identification of ICs and EALs Affected by Proposed Change:

IC:FIRE potentially degrading the level of safety of the plant

HU4.2 Unusual Event

Receipt of a single fire alarm (i.e., no other indications of a FIRE)

AND

The fire alarm is indicating a FIRE within any Table H-1 area

AND

The existence of a FIRE is not verified within 30 min. of alarm receipt (Note 1)

IC: Inability to control a key safety function from outside the Control Room

HS6.1 Site Area Emergency

An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel or Standby Shutdown Facility

AND

Control of any of the following key safety functions is not re-established within 15 min. (Note 1):

- Reactivity (Modes 1, 2 and 3 only)
- Core cooling
- RCS heat removal

IC: Automatic or manual trip fails to shut down the reactor

SU6.1 Unusual Event

An automatic trip did not shut down the reactor as indicated by reactor power ≥ 5% after any RPS setpoint is exceeded

AND

A subsequent automatic trip or the manual trip pushbutton is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

IC: Automatic or manual trip fails to shut down the reactor

SU6.2 Unusual Event

A manual trip did not shut down the reactor as indicated by reactor power ≥ 5% after any manual trip action was initiated

AND

A subsequent automatic trip or the manual trip pushbutton is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

IC: Failure to isolate containment or loss of containment pressure control.

SU8.1 Unusual Event

Any penetration is not closed within 15 min. of a VALID ES actuation signal

OR

Containment pressure > 10 psig with < one full train of containment heat removal system (1 RBS with > 700 gpm spray flow ~~OR~~ AND 2 RBCUs) operating per design for ≥ 15 min. (Note 1)

<< 10 CFR 50.54(q) Initiating Condition (IC) and Emergency Action Level (EAL) and EAL Bases Validation and Verification (V&V) Form >>

Barrier: Containment
Category: D. CMT Integrity or Bypass
Degradation Threat: Potential Loss
Threshold:
3. Containment pressure > 10 psig with < one full train of containment heat removal system (1 RBS with > 700 gpm spray flow OR 2 RBCUs) operating per design for ≥ 15 min. (Note 1)

IC: Any loss or any potential loss of either Fuel Clad or RCS barrier

FA1.1 Alert

Any loss or any potential loss of either Fuel Clad or RCS barrier (Table F-1)

IC: Loss or potential loss of any two barriers

FS1.1 Site Area Emergency

Loss or potential loss of any two barriers (Table F-1)

IC: Loss of any two barriers and loss or potential loss of third barrier

FG1.1 General Emergency

Loss of any two barriers

AND

Loss or potential loss of third barrier (Table F-1)

Barrier: Reactor Coolant System
Category: A. RCS or SG Tube Leakage
Degradation Threat: Potential Loss
Threshold:
1. RCS leakage > normal makeup capacity due to EITHER:
- UNISOLABLE RCS leakage
- SG tube leakage

Part II. Determination of Validation Method by Site EP Manager:

In-Plant Walkdown	☐	Tabletop	X
Training	☐	Other (Specify)________	☐
Simulator	☐	NA	☐

EP Manager Name (Print):	EP Manager Signature:	Date:
Pat Street	[signature]	2/23/17

Part III. Validation. (Answers marked No require resolution)

Validation Question	Yes	No	NA	Resolution and Comments
Readouts, alarms, indications. etc., available in the Control Room?	☐	☐	☐	
Monitor, gauge, etc., designations are correct?	☐	☐	☐	
Are correct units of measure displayed on the monitor, gauge, etc.?	☐	☐	☐	

<< 10 CFR 50.54(q) Initiating Condition (IC) and Emergency Action Level (EAL) and EAL Bases Validation and Verification (V&V) Form >>

All values are within instrumentation display range?	☐	☐	☐	
Is instrument display finite enough to distinguish between values?	☐	☐	☐	
No miscellaneous issues were identified during walkdown correct?	☐	☐	☐	
Part IV. Verification (Answers marked No require resolution)				
Validation Question	Yes	No	NA	Resolution and Comments
Is the IC/EAL change easy to use and does it flow well? Is sequencing logical and correct? Is it written to appropriate level of detail and unambiguous?	☐	☐	☐	
Is the IC/EAL Matrix legible and easy to use?	☐	☐	☐	
Are correct units of measure displayed on the monitor, gauge, etc.?	☐	☐	☐	

Part IV. Verification (cont.) (No answers require resolution))				
Validation Question	Yes	No	NA	Resolution and Comments
Instrumentation; Plant Computer System (PCS); and/or Plant Process Computer System (PPCS) points specified? • Correct instrument? • Correct units • Adequate instrument range? • Display unit readable? • Proper significant digits? • Instrument number and noun name provided? • Consistent with operations procedures?	☐	☐	☐	
References specified in EAL Technical Basis current and updated and source documents for inputs have been identified and verified to be appropriate for use?	☐	☐	☐	
Does the change avoid human performance challenges, latent weaknesses, and human performance traps? • No vague or missing critical detail(s). • Decisions are not over-reliant on knowledge for successful performance	☐	☐	☐	

<< 10 CFR 50.54(q) Initiating Condition (IC) and Emergency Action Level (EAL) and EAL Bases Validation and Verification (V&V) Form >>

Modifications, Emergency Plan, EAL Technical Basis, reference manual and procedure revisions, setpoint changes, software changes, training, etc. are appropriately scheduled to correspond to the EAL revision?	☐	☐	☐	
Are alarm setpoints equal to or below EAL thresholds?	☐	☐	☐	
Do radiation monitor setpoints account for background?	☐	☐	☐	
Part V. Comments:				

ATTACHMENT 6

<< 10 CFR 50.54(q) Initiating Condition (IC) and Emergency Action Level (EAL) and EAL Bases Validation and Verification (V&V) Form >>

Part VI. Completion Review and Approval Signatures		
Validation and Verification (Print Names) [Note1]: Eric Doyle	Validation and Verification Signatures: Signature in CAS	Date:
Site EP Manager Review (Print Name): Pat Street	EP Manager Signature: Signature in CAS	Date:
Senior Operations License Holder (Print Name): Jeff Hrynda	Senior Operations License Holder Signature Signature in CAS	Date:
Qualified Emergency Coordinator (Print Name): Dean Hubbard	Qualified Emergency Coordinator Signature: Signature in CAS	Date:
Engineering Review (Print Name) [Note2]: N/A	Engineering Signature:	Date:
PSA Review (Print Name) [Note3]: N/A	PSA Signature:	Date:
EP CFAM Review (Print Name): John Overly	EP CFAM Signature: Signature in CAS	Date:
[BNP, CR3, HNP, RNP] Final PNSC Approval (Print Name) N/A	[BNP, CR3, HNP, RNP] PNSC Signature	Date:

<< 10 CFR 50.54(q) Initiating Condition (IC) and Emergency Action Level (EAL) and EAL Bases Validation and Verification (V&V) Form >>

Part VII. NRC Emergency Plan and Implementing Procedure Submittal Actions	
If the proposed activity is a change to the E-Plan or implementing procedures, then create two EREG General Assignments. • One for EP to provide the 10 CFR 50.54(q) summary of the analysis, or the completed 10 CFR50.54(q), to Licensing.	☐
• One for Licensing to submit the 10 CFR 50.54(q) information to the NRC within 30 days after the change is put in effect.	☐
Notes: 1. Validation and Verification can be performed by same individual but must be: • Qualified in the subject matter • Separate from the author of change • A cross-discipline reviewer 2. [BNP, CR3, HNP, RNP] System specific Engineering Review is required for EAL changes related to process equipment such as radiological instruments and environmental monitoring. 3. [BNP, HNP, RNP] PSA review is required for EAL changes to ensure any potential or actual impact to PSA calculations or assumptions are adequately addressed. (Not applicable to CR3)	

QA RECORD

Oconee Emergency Plan
List Of Effective Pages

SECTION	PAGE NUMBER	REVISION NO.	DATE
Emergency Plan Approval Cover Sheet		Rev. 2017-002	March 2107
List of Effective Pages	Page 1 - 3	Rev. 2017-002	March 2017
List Of Figures	Page 1 - 5	Rev. 2014-02	October 2014
Record Of Changes	Page 1 - 11	Rev. 2017-002	March 2017
Table of Contents	Page 1	Rev. 2012-05	December 2012
I. Introduction	Page 1 - 5	Rev. 2012-05	December 2012
	Page i-6	Rev. 2008-02	October 2008
	Page i-6a	Rev. 2012-05	December 2012
II. Planning Standards and Evaluation Criteria			
A. Assignment of Responsibility	Page A-1 - A-8	Rev. 2016-002	September 2016
B. Onsite Emergency Organization	Page B-1 - B-21	Rev. 2017-002	March 2017
C. Emergency Response Support And Resources	Page C-1 & C-2	Rev. 2012-05	December 2012
D. Emergency Classification System	Page D-1 - D-240	Rev. 2017-002	March 2017
E. Notification	Page E-1 & E-2	Rev. 2008-02	December 2008
F. Emergency Communications	Page F-1 - F-8	Rev. 2016-003	December 2016
G. Public Information and Education	Page G-1 - G-5	Rev. 2014-02	October 2014

Oconee Emergency Plan
List Of Effective Pages

Oconee Emergency Plan
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SECTION	PAGE NUMBER	REVISION NO.	DATE
APPENDIX 3 Alert And Notification System Description	Page 1 - 4	Rev. 2015-004	May 2015
APPENDIX 4 Evacuation Time Estimates	Page 1	Rev. 2013-01	October 2013
APPENDIX 5 Letters of Agreement	Page 1	Rev. 2017-002	March 2017
	Page 2	Rev. 2017-002	March 2017
APPENDIX 6 Distribution List	Page 1 - 4	Rev. 2014-02	October 2014
APPENDIX 7 Emergency Data Transmittal System	Page 1	Rev. 2012-01	June 2012
APPENDIX 8 Spill Prevention Control And Countermeasure Plan ONS Pollution Prevention Plan -cover sheet	Page 1	Rev. 2016-001	June 2016
APPENDIX 9 ONS Chemical Treatment Ponds 1, 2 and 3, Groundwater Monitoring Sampling And Analysis Plan	Page 1	Rev. 2016-003	December 2016
APPENDIX 10 Hazardous Materials Response Plan	Page 1	Rev. 2016-003	December 2016

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 throughout 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 mindset 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	06/12	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.
2014-01	03/14	• Section B - Removed reference to having home addresses listed in the emergency telephone directory as these were never listed in the telephone directory and clarified EOF Services Group actions. Updated titles of ERO positions in the TSC and OSC consistent with duty roster. • Section D - Added clarification for which RIA-45 is to be used. Respectively, it is expected that 1RIA-45, 2RIA-45 and 3RIA-45 would be used in connection with Enclosure 4.3, Abnormal Rad Level/Radiological Effluent. 4RIA-45 is not specifically related to a unit and therefore it is not applicable to Enclosure 4.3. • Section G - Removed Calendar and replaced with Note that the calendar is retained on file with EP Staff. • Section H - Eliminated drawings of Alternate TSC and Alternate OSC as these are for implementation and not needed in Emergency Plan. Removed implementation details from Primary TSC and Primary OSC drawings. Corrected Figure H-20 and shifted table alignment. • Section J - Provided editorial corrections to procedure numerical references where applicable. • Section M - Provided clarification of EOF Services listed on Figure M-2. • Section P - Provided editorial corrections to procedure numerical references where applicable, and changed a reference from the EP Functional Area Manual to a fleet administrative procedure reference (EP FAM to AD-EP-ALL-0001). Eliminated reference to HR Emergency Plan. • Appendix 5 - Removed all copies of the Letters of Agreement and indicated they are included by reference. The actual Letters of Agreement are retained on file by the EP Staff.

RECORD OF CHANGES (Continued)

REVISION NUMBER	EFFECTIVE DATE	REASON FOR REVISIONS
2014-02	10/14	• Section A - Revised for change from pagers to notify the ERO to using cell phones. Shift Manager delegates actual activation of notification device to Security if available or qualified operator if security is unable. • Section B - Revised responsibility for Radwaste function from Chemistry Group to Operations Group. • Section D - Revised responsibility for Radwaste function from Chemistry Group to Operations Group, including reference to chemistry procedures to operation procedures. • Section F - Revised for change from pagers to notify the ERO to using cell phones. Shift Manager delegates actual activation of notification device to Security if available or qualified operator if security is unable. • Section G - Procedure number changes • Section H - Removed specific locations of kits as these were insufficiently detailed and did not contain all kit locations. • Section I - Procedure number changes. • Section J - Procedure number changes. • Section M - Procedure number changes, title changes. • Section N - Changes to show new rules including 8 year cycle, consistency with fleet documents practices, and format. • Section P - Revised responsibility for independent audit from NSRB to NOS Manager, deleted duplicated paragraph and updated the listing of the implementing procedures. • Appendix 6 - Updated distribution list to reflect new format of E Plan and associated implementing procedures.
2014-003	12/14	Changes made associated with the modification from Raddose V to URI, and updates to WEBEOC.
2015-001	01/15	Changes made to Section F, EOF Communications and Figure F-1.
2015-002	03/15	Changes made as a result of superseding SH/0/B/2005/002, EP Fam 3.19 and Appendix 5.
2015-003	04/15	Changes made to Section D consisting of Protected Service Water replacing the Station Auxiliary Service Pump as a result of system modification. Replaced Selective Signaling with DEMNET .
2015-004	5/15	E-Plan changes made to Sections App 3, D, F, H, K, & P and the Table of Contents consisting of AD-EP ALL-0203 & AD-EP ALL-0204 procedure reference changes and changes from Selective Signaling to DEMNET. Also includes title changes from Operations Shift Manager to Shift Manager. • Appendix 3: Revised Selective Signaling to DEMNET • Section D: Revised Selective Signaling to DEMNET and removed OSM • Section F: Revised Selective Signaling to DEMNET, removed OSM, and SH/0/B/2005/002 to AD-EP ALL-0203 • Section H: Removed specific equipment reference to reduce E-plan revisions • Section K: Implementation for AD-EP-ALL-0204 for administration of KI, and title changes from Operations Shift Manager to Shift Manager • Section P: Reference updates SH/0/B/2005/003 to AD-EP ALL-0204
2015-006	9/15	E-Plan changes made to Section H and Section P, consisting of Emergency Operating Facility (EOF) changes and revised ONS floor plans to correctly indicate ERO positions within ERFs in Section H and update procedure references in Section P.

RECORD OF CHANGES (Continued)

REVISION NUMBER	EFFECTIVE DATE	REASON FOR REVISIONS
2015-007	4/16	Changes to update Letters of Agreements and references throughout; State of SC supersedes SC Highway Patrol and SC Law Enforcement Division, Greenville Health System has acquired Oconee Medical Center and Blue ridge Physicians. Change to enhance the BASIS of 4.6.U.1 to add damage to 'nearby' equipment in response to AR 01910385, clarifies damage isolated to non-safety components would not necessitate entry in to an Unusual Event. Enhancement in Section D to the terms 'Condition A' , which will be replaced by the term 'Imminent Failure' and 'Condition B' which will be replaced by 'Potential Failure' in 2016. The change was driven by FERC in order to minimize confusion when communicating with offsite organizations in accordance with Hydro Emergency Action Plans. Change corrected spelling from Manger to Manager on p. F-4. change 15 updated Figure J-5 to current Site Layout, added Site Support Complex, PSW Building and labeled RP Building.
2016-001	6/16	E-Plan change made to Appendix 8, Spill Prevention, Control and Countermeasure Plan (SPCC Plan), 1) revised contents of the SPCC due to change in inventory on site, addition of new used oil storage area, update to contact information and name change from Duke Power to Duke energy. 2) Moved the contents of the SPCC out of the Emergency Plan and include by reference the location as an implementing program of the Emergency Plan. Hard copy to be maintained by Environmental Services and retrievable electronically in Fusion. AR 02031855.
2016-002	9/16	Section A/Appendix 5; Revised wording to remove the requirement to update Letters of Agreement (LOA) and Memoranda of Understanding(MOA) every three years, changed to review annually and revise as necessary during Emergency Plan certification review Section B; Change to keep responsibility for 'Classification' on site by SM/EC or TSC/EC after EOF is activated. Section H; Title change from 'Backup' ERF's back to 'Alternate', changed in error in revision 2014-002 due to misinterpretation of definitions. This change will be in compliance with the definitions found in NEI 13-01, no change to the facilities. Section M; 1) Revised ONS Spill Response procedure to Fleet Spill Response Procedure. 2) Added description of recovery activities following Beyond Design Basis Natural Event from INPO L4-13-3 Section P; 1) Title change to revise ONS Spill Response procedure to Fleet Spill Response Procedure.
2016-003	12/16	Section B; Revised to clarify the requirements for minimum staffing in Figure B-8b, Offsite Communicators (2), one of the Communicators is the NRC Communicator. Section F, Clarify that ERO activation is at the discretion of the SM/EC for UE, required at Alert and higher. Clarify Site Assembly is required at an Alert or higher. Change Backup to Alternate TSC housekeeping issue per the guidance in NEI 13-01. (PRR 02080151) Section P; 1) Title change to revise ONS Spill Response procedure to Fleet Spill Response Procedure. Appendix 9, Removed the Ground Water Monitoring Plan from the Emergency Plan and implemented by reference, 50.54q statement added to the document and documented location in Fusion. Appendix 10, Removed the Hazardous Materials Response Plan from the Emergency Plan and implemented by reference, 50.54q statement added to the document and documented location in Fusion.
2017-001	3/17	Revised Section D, Emergency Classification System, 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. ONS is upgrading to an emergency action level (EAL) scheme based on the Nuclear Energy Institute (NEI) 99-01, Revision 6. MIL16109A093

RECORD OF CHANGES (Continued)

REVISION NUMBER	EFFECTIVE DATE	REASON FOR REVISIONS
2017-002	3/17	Section B, updated titles due to organizational changes. Section D, revised to correct technical inaccuracies in the basis identified during training on the new EAL scheme; removed second manual trip device wording, corrected containment heat removal train from 'or' to 'and', replaced starting a second HPI pump as initiator with initiating actions due to exceeding NORMAL MAKEUP CAPABILITY. Section H, added Ludlum to available equipment due to retirement of older equipment and fleet standardization. Section I revised to reflect upgraded EAL scheme in section D. Section P updated procedure references and removed reference to Appendix 'R' due to implementation of NFPA 805. Appendix 5 was updated to delete unused portions and removed letters for Beyond Design Basis Events that are maintained by fleet contracts.

B. ONSITE EMERGENCY ORGANIZATION

Adequate staffing to provide for initial emergency response in key functional areas is maintained at all times, timely augmentation of response capabilities is available, and the interfaces among various onsite response activities and offsite support and response activities are specified.

B.1. Emergency Response Organization

Figures B-1 through B-6 shows the Oconee Nuclear Site Emergency Response Organization that would be established during an incident along with the duties the different groups would assume. Designation of personnel to the Emergency Response organization is determined by the particular job expertise of an individual. This assures that these personnel are qualified to carry out their responsibilities during an emergency. The normal staffing assignments at the Oconee Nuclear Site include emergency response responsibilities. See Figures B-10 for emergency responsibilities for designated groups.

The Emergency Response Organization is outlined in Division/Section Directives. These directives establish the duties, responsibilities and alternates for each required emergency response position. Response procedures have been established for each section through these Directives.

B.2. Emergency Coordinator - (24 hour)

As a result of training and day-to-day experiences in the normal operating mode at the Oconee Nuclear Site, the Operations Shift Manager will assume authority and responsibility for any emergency that may arise at the site. He will assume control of the situation, alert and warn personnel and others, take necessary onsite remedial action, obtain necessary outside aid and notify management and appropriate offsite agencies. The authority vested in this position by Duke Energy management enables the Operations Shift Manager to declare an emergency as necessary to protect the plant, site personnel, and the general public. He is vested with the authority to provide protective action recommendations to state and local agencies for implementing offsite emergency measures.

The Operations Shift Manager will continue with these responsibilities until relieved by the Station Manager (alternate).

B.3. Alternates for Emergency Coordinator

In an emergency situation where the Station Manager/Emergency Coordinator is unavailable, for whatever reason, and an acting Station Manager has not been designated in writing, an alternate will be contacted. Alternates are personnel with intimate knowledge of plant operations and will fulfill the position as TSC Emergency Coordinator when assigned duty. Alternates are designees appointed by the Station Manager.

Note: Emergency Telephone Directory, provides names and phone numbers for the Emergency Coordinators listed above.

B.4 Functional Responsibilities of the Emergency Coordinator

Figure B-5 defines the duties and responsibilities of the Emergency Coordinator within the emergency response organization.

The Emergency Coordinator and the EOF Director are the individuals responsible for making protective action recommendations to the state and county agencies. Once the Emergency Operations Facility is activated, only the EOF Director has this responsibility. Prior to the activation of the EOF, the Emergency Coordinator is responsible for making protective action recommendations, classifying/ downgrading/ escalating/terminating events, and approving notification forms to offsite agencies. Once the EOF is activated, the Emergency Coordinator will retain responsibility for making classifications. These responsibilities may not be delegated.

IF AT ANY TIME the EOF is Activated, **THEN** the following applies:

- Classification of events are performed by either the TSC or Control Room
- Immediate communication to the EOF is required upon upgrade of a classification of an event either the TSC or Control Room
- Notification of Offsite Agencies are performed by the EOF

Protective Action Recommendations (PAR) are performed by the EOF.

B.5 Minimum Staffing Requirements for Nuclear Power Plant Emergencies

Figure B-8(a/b) identifies the positions by title and major tasks to be performed by the persons assigned to the functional areas of emergency activity within 45 to 75 minutes. The TSC and OSC will be activated within 75 minutes of event classification. The EOF (Figure B-9) will also be staffed and operational by a minimum staff within 75 minutes of event classification.

A detailed analysis demonstrating that on-shift personnel assigned emergency plan implementation functions are not assigned responsibilities that would prevent the timely performance of their assigned functions as specified in Figure B-8a is located in ONS-OSSA-12212012 Rev: 0.

B.6 Interface - Onsite, Offsite Organizations

(Figure A-1) shows the interfaces between and among the onsite functional areas of emergency activity, site support, local services support, and State and Local government response organizations. The onsite Technical Support Center, Operational Support Center, and the Emergency Operations Facility is shown.

B.7 Minimum Staffing Requirements - Corporate Level

Personnel from the corporate location will provide assistance to the Oconee Nuclear Station in the areas of dose assessment, coordination of news media and severe accident analysis. These people will respond on an as needed basis.

The Emergency Response Facility Organization is shown on Figures B-4A through B-4D. Division/Section Directives detail the emergency response organization and are included as implementing procedures.

Should an event occur at another Duke Energy nuclear facility concurrent with an event at Oconee Nuclear Site, additional resources would be required to support the EOF Organization. Figure B-11 shows the supplemental EOF organization that would be required to manage a multi-site event. An additional Assistant EOF Director would function as the lead manager to support the second site. The EOF Director would have overall emergency management responsibility for both events.

B.8 Contractor/Private Organizations - Technical Assistance

Private Contractors and/or companies that would be available to augment and support the emergency organization:

Waste Management	Chem Nuclear
Bartlett Nuclear	Emergency Equipment Supplier
SEG	(e.g. Safe Industries, Inc.)
Alaron	AREVA

Additional groups that could respond to emergencies would be contacted by the EOF Services Group at the Emergency Operations Facility.

B.9 Local Agency Support

Agreements with local agency support groups have been made to assist the Oconee Nuclear Site during emergency situations.

POLICE -

Oconee Sheriff's Department

Pickens Sheriff's Department

State of South Carolina - S. C. Highway Patrol and S. C. Law Enforcement Division

AMBULANCE -

Oconee Memorial Hospital - Emergency Medical Services

MEDICAL -

Oconee Memorial Hospital - Greenville Health System

FIRE FIGHTING -

Oconee County Emergency Services Fire/Chemical Spill

Keowee-Ebenezer Fire Department

Corinth-Shiloh Fire Department

Six-Mile Fire Department

Keowee Fire Department

RADIOLOGICAL MONITORING -

Pickens County Emergency Management Agency

Oconee County Emergency Management Agency

EVACUATION OF POPULATION -

Oconee County Emergency Management Agency

Pickens County Emergency Management Agency

HOSPITAL -

Oconee Memorial Hospital

USE OF BUILDINGS -

Oconee County School District

Pickens County School District

FIGURE B-1
OCONEE NUCLEAR STATION
EMERGENCY RESPONSE ORGANIZATION
FUNCTIONAL AREAS OF EMERGENCY RESPONSE

1. Emergency Response Coordination
   - Operations Shift Manager
   - Station Manager
   - Division Managers
   - Section Managers

2. Plant Systems Operations
   - Superintendent of Operations
   - Operations Shift Managers
   - Operations Engineers
   - On Shift Staff (Operations)
   - Engineering

3. Accident Assessment
   - Emergency Coordinators
   - Operations Shift Managers
   - Operations Engineers
   - Shift Managers
   - Site Engineering
   - Severe Accident Analysis Group (GO)

4. Radiological Environmental Survey and Monitoring
   - Contract service
   - Radiation Protection Section

5. First Aid/Rescue, Firefighting
   - Medical Emergency Response Team Members
   - Fire Brigade

6. Personnel Monitoring
   - Radiation Protection Section

7. Decontamination
   - Radiation Protection Section

FIGURE B-1
OCONEE NUCLEAR STATION
EMERGENCY RESPONSE ORGANIZATION

8. Security of Plant and Access Control
   Duke Security (ONS and Oconee JIC)
   Building Security/Access & Control (Charlotte EOF)

9. Repair/Corrective Actions
   Nuclear Supply Chain
   Site Services Group

10. Personnel Accountability
    Division Managers
    Section Managers
    Supervisors
    Duke Security

11. Radiological Accident Assessment
    Radiation Protection Section (ONS/GO/CNS/MNS)
    Site Engineering
    Operations Group
    Chemistry Group
    Severe Accident Analysis Group (GO)

12. Communications
    Operations
    Organizational Effectiveness
    Training

13. Radiation Protection Section
    Radiation Protection Section

14. Plant Chemistry
    Chemistry Group

FIGURE B-1
OCONEE NUCLEAR STATION
EMERGENCY RESPONSE ORGANIZATION

15. Radwaste Operations
    Operations Group

16. Technical Support
    Site Engineering

17. Manpower Planning and Logistical Support
    Work Control Group
    Nuclear Supply Chain
    Site Services Group

18. Public Information
    Corporate Communications - Joint Information Center

19. Licensee Representative to State County EOC
    Site Engineering

FIGURE B-2
OCONEE NUCLEAR STATION
TECHNICAL SUPPORT CENTER ORGANIZATION CHART

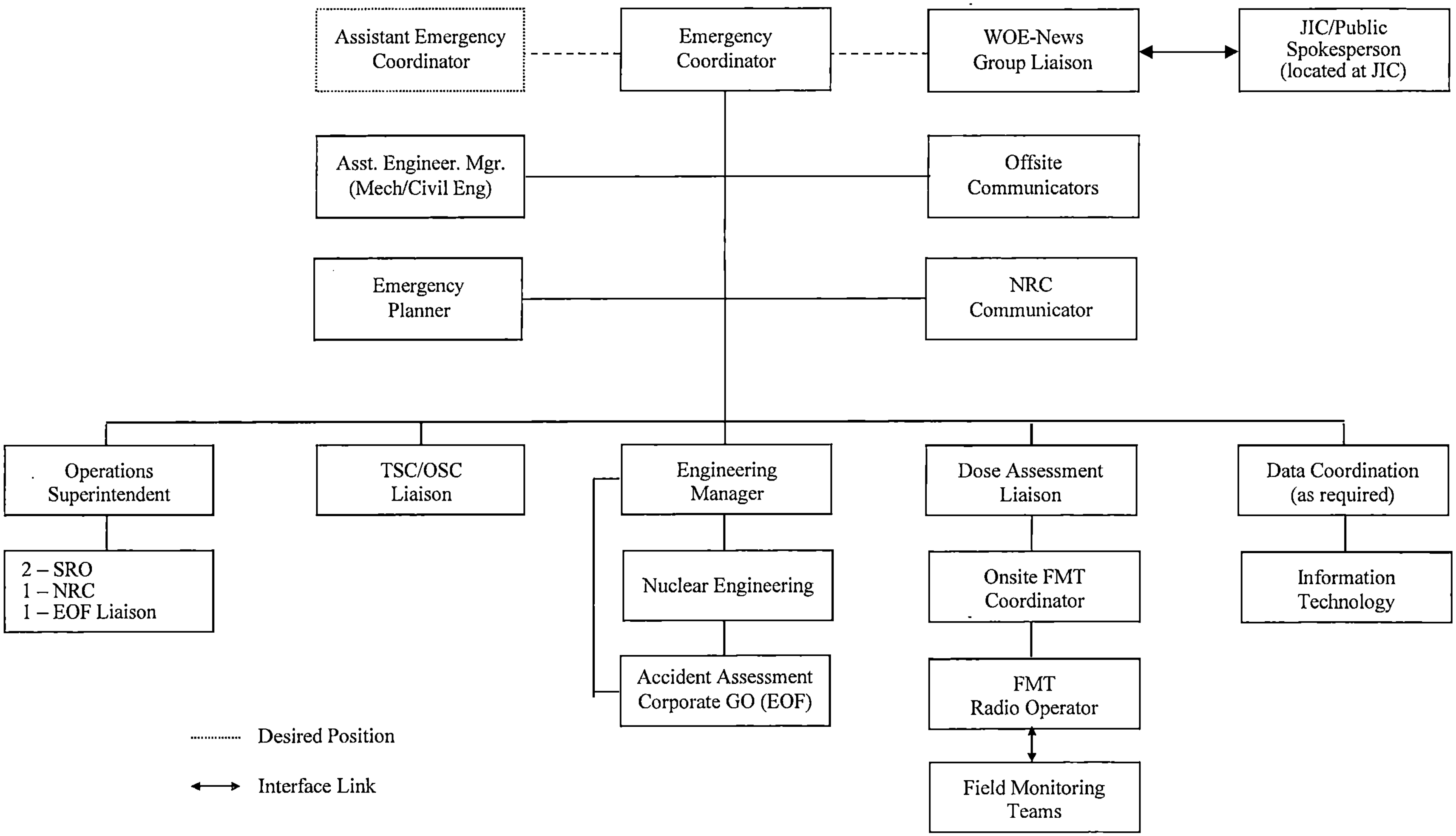


FIGURE B-3
OCONEE NUCLEAR STATION
OPERATIONAL SUPPORT CENTER ORGANIZATION CHART

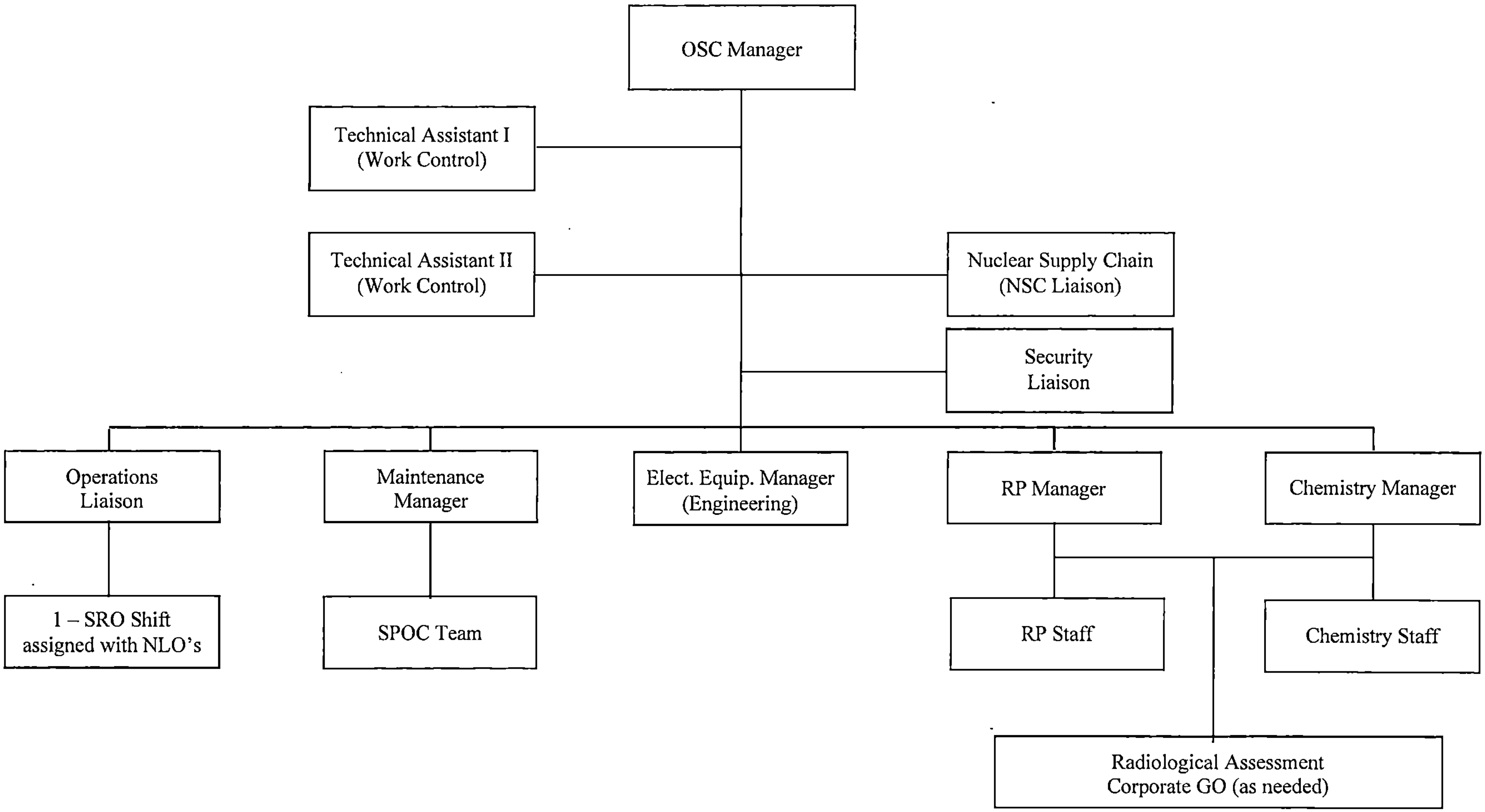


FIGURE B-4A
OCONEE NUCLEAR STATION
Emergency Operations Facility

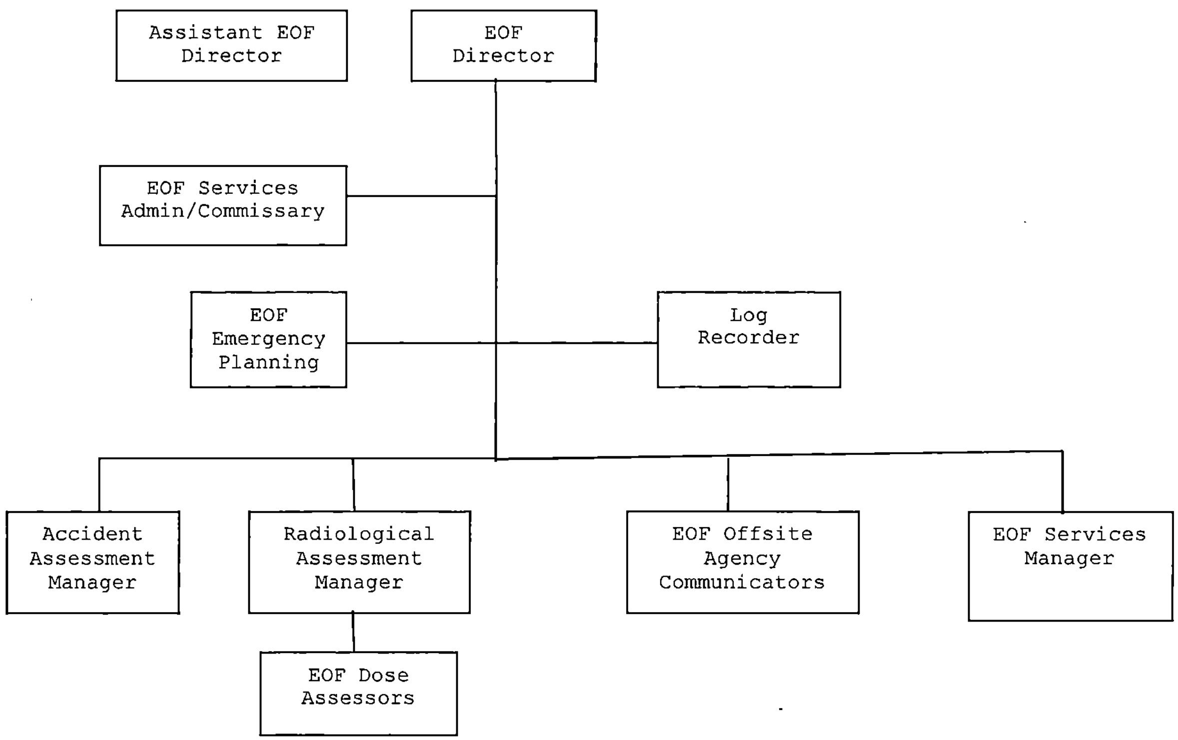


FIGURE B-4B
DUKE ENERGY COMPANY
OCONEE NUCLEAR STATION

EMERGENCY OPERATIONS FACILITY

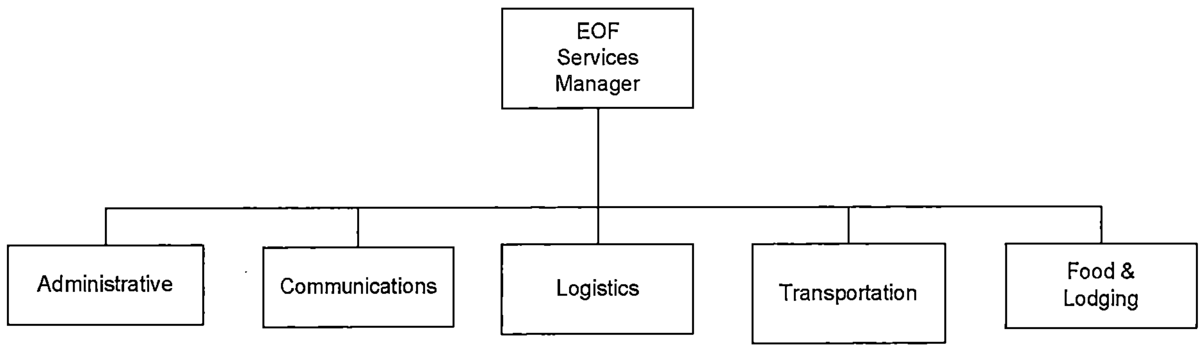


FIGURE B-4C
OCONEE NUCLEAR STATION
EMERGENCY OPERATIONS FACILITY

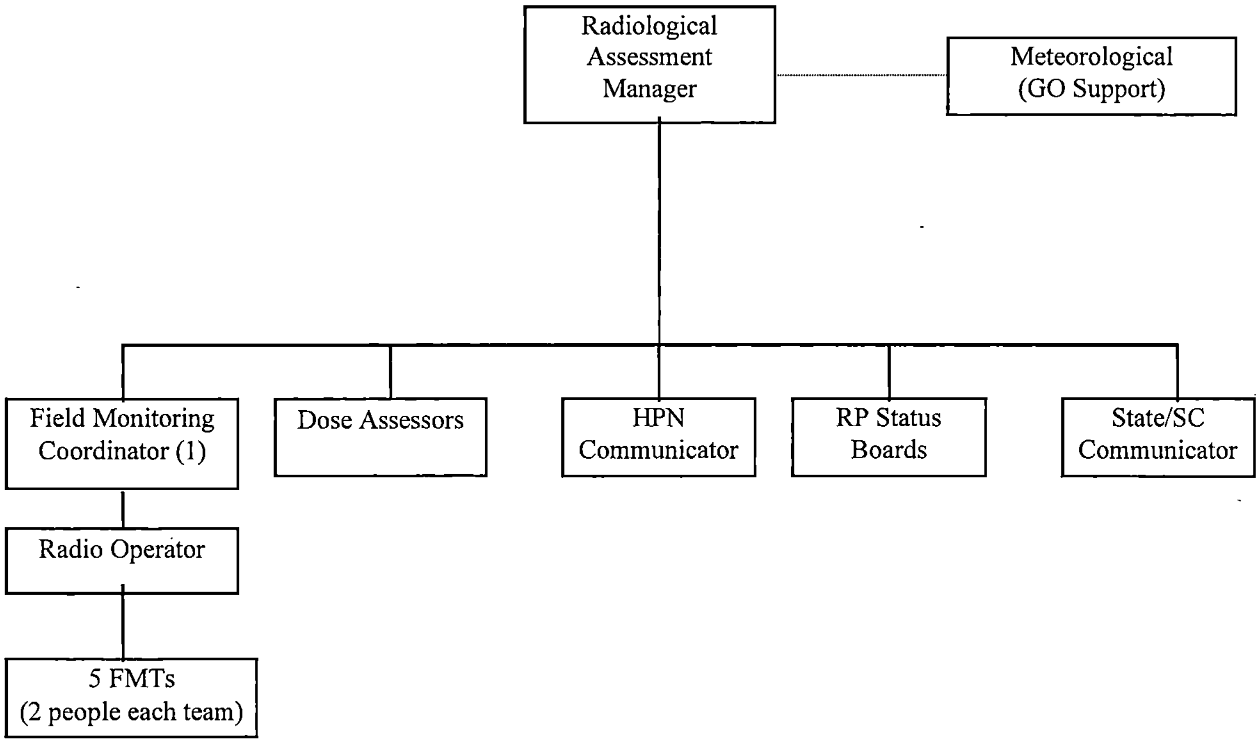


FIGURE B-5
OCONEE NUCLEAR STATION
EMERGENCY COORDINATOR DUTIES

Duties and Responsibilities:

1. Activate the Emergency Response Organization (Technical Support Center, Operational Support Center and the Emergency Operations Facility).

2. Coordinate technical assistance for remedial actions to mitigate circumstances surrounding plant operations.

3. Designate individual to communicate with offsite agencies promptly. Approve all information released via the emergency notification form. Approval (signature) of contents of message form may not be delegated to others in the emergency organization.

4. Initiate emergency actions within the provisions of the Oconee Nuclear Site Emergency Plan.

5. Classify emergency events. Escalate/de-escalate or terminate from an emergency status. Responsibility may not be delegated.

6. Make senior technical and management staff available onsite for consultation with NRC and State on periodic basis if EOF has not been activated.

7. Coordinate all emergency actions concerning the technical aspects of the corrective actions taken within the Technical Support Center.

8. Evacuate all non-essential personnel on site if a radiological emergency exists. Be aware of exposure guidelines of personnel.

9. Recommend protective action guides for the safety and welfare of the public to the appropriate offsite agency if the Emergency Operations Facility/EOF Director is not in a position to do so. This authority may not be delegated.

10. Authorize exposures in excess of routine yearly exposure limits for lifesaving and equipment repair missions in accordance with RP/0/B/1000/011. This responsibility can be delegated to the RP Manager in the OSC.

FIGURE B-6
OCONEE NUCLEAR STATION
EMERGENCY OPERATIONS FACILITY DIRECTOR

Duties and Responsibilities:

1. Overall Management of the offsite emergency response activities of Duke Energy (Oconee Nuclear Site).

2. Recommend protective action guides for the safety and welfare of the public to the appropriate offsite agency. This authority may not be delegated.

3. Approve all information released via the emergency notification form. Approval (signature) of contents of message form may not be delegated to others in the emergency organization.

4. Coordination with federal, state and local government agencies.

5. Provide approval of news releases if Public Spokesperson is unavailable.

FIGURE B-8a
OCONEE NUCLEAR STATION
MINIMUM ON-SHIFT STAFFING LEVELS

Functional Area	Major Tasks	Emergency Positions	Shift Staffing
1. Plant Operations and Assessment of Operational Aspects (a)	--	CR Supervisor (SRO) Control Room Operator (RO) Non-Licensed Operator (NLO)	3 6 3
2. Emergency Direction and Control	Command and Control	Ops Shift Manager	1
3. Notification & Communication	Licensee	Operator (SRO/RO/NLO)	1(b)
	Local/ State	Operator (SRO/RO/NLO)	1
	Federal	Operator (SRO)	1(b)
4. Radiological Assessment	Dose Assessment	RP Qualified Individual	1
	In-plant Surveys	RP Qualified Individual	1
	Onsite Surveys	RP Qualified Individual	1
	Chemistry	Chemistry Technician	1
5. Plant System Engineering, Repair, and Corrective Actions	Tech Support – OPs	Shift Technical Advisor	1
	– Core Damage	Shift Technical Advisor	1(b)
	Repair and Corrective Actions	Mechanical Maintenance IAE Maintenance	2 2
6. In-Plant PAs	Radiation Protection (such as access control, job coverage and personnel monitoring)	RP Qualified Individual	2(b)
7. Fire Fighting (c)	--	Fire Brigade Lead (SRO/NLO) Fire Brigade Member (NLO) Fire Brigade Member	1 4 5(b,c)
8. 1st Aid and Rescue	--	MERT (d)	2
9. Site Access Control and Accountability	Security & Accountability	SAS Operator Security Personnel	1 (e)
		Minimum # of Personnel:	**31**

(a) The Control Room staff complement is reflective of 3 Units in operation.

(b) May be performed by an individual filling another position provided they are qualified to do the collateral function.

(c) The Fire Brigade requirement of ten members is met by using five personnel from Operations (including the Fire Brigade Leader) and five personnel from either SPOC, Radiation Protection, Chemistry or Security (SLC 16.13.1).

(d) The Medical Emergency Response Team (MERT) can be filled by any qualified technician.

(e) Per Duke Energy ONS Security Plan.

FIGURE B-8b
OCONEE NUCLEAR STATION
MINIMUM AUGMENTED ERO STAFFING LEVELS

MAJOR FUNCTIONAL AREA	MAJOR TASKS	POSITION TITLE OR EXPERTISE	CAPABILITY FOR ADDITIONS* WITHIN 45 MINUTES	WITHIN 75 MINUTES
Emergency Director and Control (Emergency. Coordinator) ***		Station Manager	—	1
Notification/Communication	Notify Company Personnel, State, County, Federal Agencies and maintain communication	State/County Communicator NRC Communicator		1 1
EOF/Radiological Accident Assessment and Offsite Agency Support	EOF Director (Protective Action Recommendation, Offsite Agency Interface, ENF approval)	DPC Senior Manager	—	1
	EOF Plant Assessment	Accident Assessment Manager	—	1
Dose Assessment and Protective Action Recommendations	EOF Offsite Dose Assessment/ Protective Action Recommendations	Radiological Assessment Manager	—	1
Offsite Notifications	EOF Offsite Agency Notifications	Offsite Agency Communicator Offsite Agency Communicator	— —	1 1
	EOF Access Control	Electronic Card Reader	—	#
Dose Assessment and Protective Action Recommendations	TSC Dose Assessment/Protective Action Recommendations	Radiological Assessment	—	1
	Offsite Surveys Onsite Surveys (Out-of-Plant) In-Plant Surveys Chemistry/Radio Chemistry	Field Monitoring Teams (2) RP Qualified Individuals RP Qualified Individuals Rad/Chem Technician	 1 1 —	4 **** 1 1 1
Plant System Engineering, Repair and Corrective Actions	Technical Support	Core/Thermal Hydraulics Electrical Mechanical	— — —	1 1 1
	Repair and Corrective Actions	Mechanical Maintenance Rad/Waste Operator I&E Technician	— — —	1 1 2
Protective Actions (In-Plant)	Radiation Protection A. Access Control B. RP Coverage for Repair Corrective Actions, Search and Rescue, First Aid & Firefighting C. Personnel Monitoring D. Dosimetry E. On-Shift Dose Assessment	RP Qualified Individuals	—	4
Firefighting	—	Fire Brigade		Local Support
Rescue Operations & First-Aid	—	MERT Team		Local Support

FIGURE B-8b
OCONEE NUCLEAR STATION
MINIMUM STAFFING LEVELS

* Consideration is given to the fact that many of the Oconee Nuclear Site Emergency Response Organization personnel do not live within a radius of the station which will allow a response time of 30 minutes or less under ideal conditions. Factors such as weather conditions, road capacity and traffic density, and the distance to travel from residence to the emergency response facility, indicate a realistic response time from a few minutes to 1 hour and 15 minutes for most employees. Consideration is also given to personnel on shift who are qualified and sufficient in number to handle any emergency condition until response personnel begin to arrive on site.

*** Management of the Offsite Emergency Response will be assumed by the EOF Director when the Emergency Operations Facility is activated.

Management of the Onsite Emergency Response is assumed by the Station Manager/alternate acting as the Emergency Coordinator when the Technical Support Center and Operational Support Centers are activated.

**** The Field Monitoring Teams will initially report to the Body Burden Analysis (BBA) Room. If needed, the Field Monitoring Teams will dispatch from the Body Burden Analysis (BBA) Room. Once the Emergency Operations Facility (EOF) Field Monitoring Coordinator is ready he/she will assume control of the Field Monitoring Teams. A FMT consists of one RP qualified individual and one vehicle driver.

An electronic card reader in conjunction with a posted building security officer fulfills the function for controlling access to the EOF during emergencies.

FIGURE B-9
OCONEE NUCLEAR STATION
MINIMUM STAFFING REQUIREMENTS

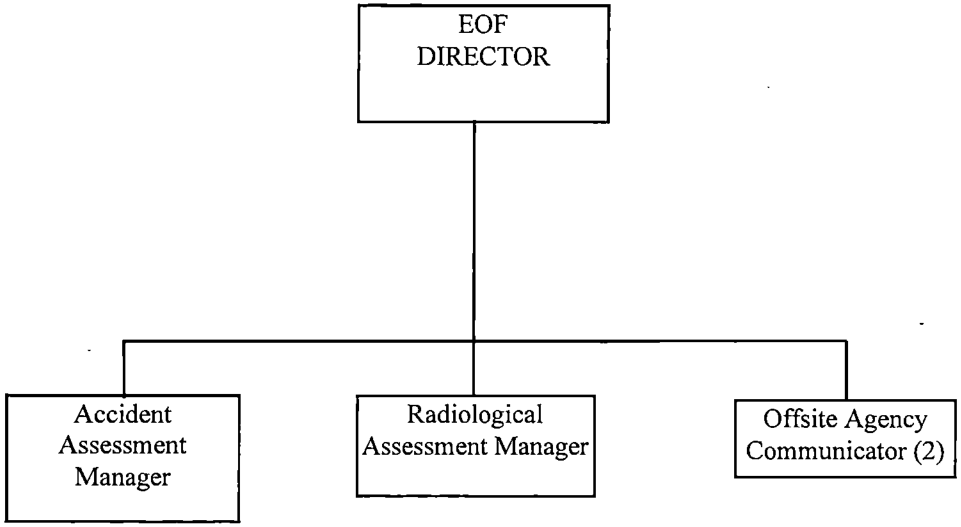


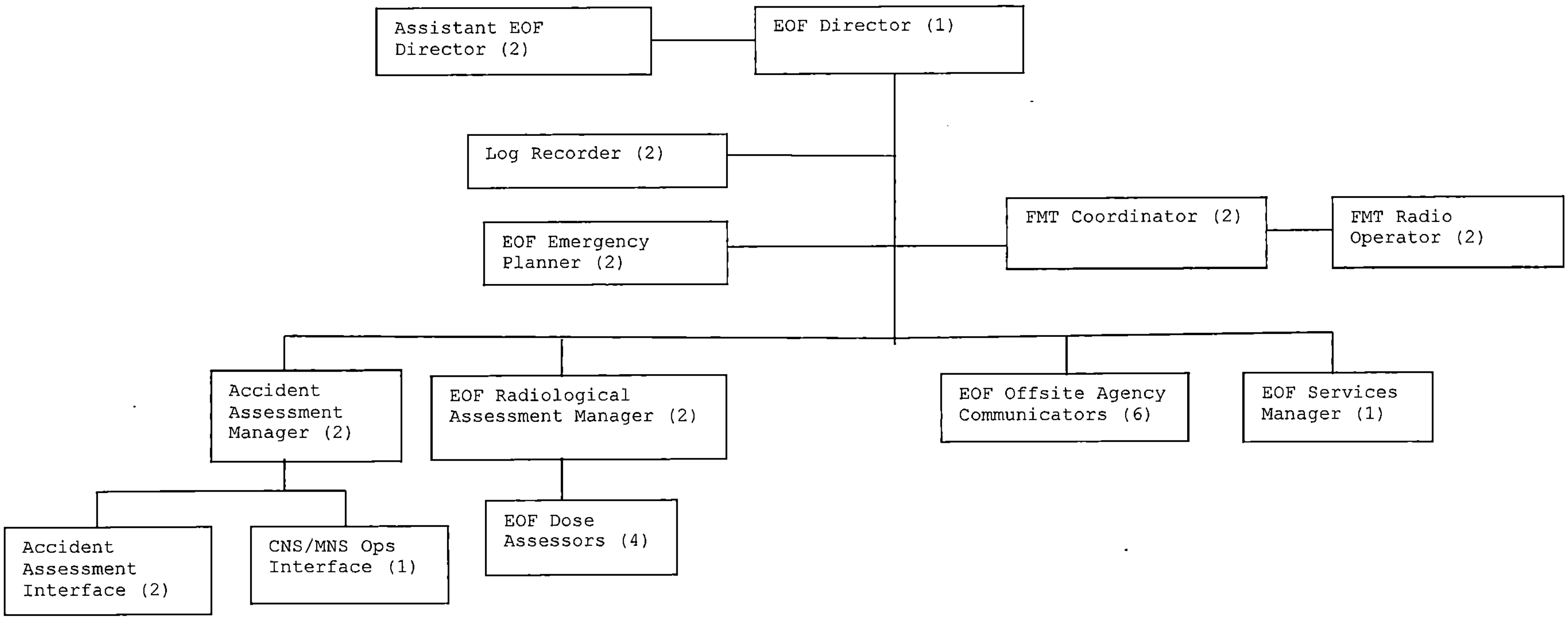
FIGURE B-10
OCONEE NUCLEAR STATION
ONSITE EMERGENCY RESPONSE DUTIES

RADIATION PROTECTION	CHEMISTRY	MAINTENANCE	OPERATIONS	WORK CONTROL	BUSINESS GROUP
Onsite Monitoring	Post-accident liquid sampling	Repair/calibration of electrical and mechanical equipment	Plant Operations	OSC Coordination	Provide personnel for field monitoring teams
Initial dose assessment	Chemistry analysis	Staff OSC with onshift personnel	Accident assessment	Recovery implementation	
Decontamination		Augment OSC with additional personnel	Safe shut-down of reactor	Provide personnel for field monitoring teams	
Post-accident gaseous sampling			Liaison with OSC		
Dose Control			Fire brigade response		
Provide personnel for field monitoring teams			Initial Emergency Management		
Evacuation coordination			NRC Communications		
RP support for work tasks in the field			Fire Support		
			Tank Farm Operation		
			Radwaste Operation		

FIGURE B-10
OCONEE NUCLEAR STATION
ONSITE EMERGENCY RESPONSE DUTIES

Organizational Effectiveness	**ENGINEERING**	**SECURITY**	**SAFETY**	**NUCLEAR SUPPLY CHAIN SITE SERVICES GROUP**
	TSC LOG	ACCESS & CONTROL		COMMUNICATIONS EQUIPMENT
OFF SITE COMMUNICATIONS	TSC/OSC LIAISON TECHNICAL ASSISTANT	EVACUATION COORDINATION		SUPPLY/PARTS
ASSIST WITH ACCOUNTABILITY REPORTING	ENGINEERING SUPPORT	MEDICAL (MERT)	PERSONNEL SAFETY	HEAVY EQUIPMENT OPERATORS
ASSIST WITH EVACUATION	LIAISON WITH CORPORATE ACCIDENT ASSESSMENT GROUP			COMMISSARY
EMERGENCY PLANNING	TSC STATUS BOARDS			

FIGURE B-11
OCONEE NUCLEAR STATION
Common EOF - Multi-Site Event Staffing



1. Notification Devices activated for second unit - all call response
2. Assistant EOF Director assumes responsibility as lead manager for designated Site
3. Additional Log Keeper retained to support 2nd Site
4. Additional Accident Assessment Manager retained to support 2nd Site
5. Additional Dose Assessor retained to support 2nd Site
6. Additional FMT Coordinator retained to support 2nd Site
7. Additional FMT Radio Operator retained to support 2nd Site
8. Four additional Offsite Communicators as needed to support both Sites
9. Additional Emergency Planner as needed to support 2nd Site
10. Additional Radiological Assessment Manager as needed to support 2nd Site
11. Additional Assistant EOF Director as needed to support 2nd Site
12. Oconee Ops Interface position is staffed in the ONS TSC
13. Additional Accident Assessment Interface as needed to support 2nd Site

EMERGENCY PLAN
EMERGENCY ACTION LEVEL TECHNICAL BASES

Table of Contents

SECTION PAGE

1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Oconee Nuclear Station (ONS). It should be used to facilitate review of the ONS EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of RP/0/A/1000/001, Emergency Classification, may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Coordinator in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training and for explaining event classifications to off-site officials.

The expectation is that emergency classifications are to be made as soon as conditions are present and recognizable for the classification, but within 15 minutes or less in all cases of conditions present. Use of this document for assistance is not intended to delay the emergency classification.

Because the information in a basis document can affect emergency classification decision-making (e.g., the Emergency Coordinator refers to it during an event), the NRC staff expects that changes to the basis document will be evaluated in accordance with the provisions of 10 CFR 50.54(q).

2.0 DISCUSSION

2.1 Background

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the ONS Emergency Plan.

In 1992, the NRC endorsed NUMARC/NESP-007 "Methodology for Development of Emergency Action Levels" as an alternative to NUREG-0654 EAL guidance.

NEI 99-01 (NUMARC/NESP-007) Revisions 4 and 5 were subsequently issued for industry implementation. Enhancements over earlier revisions included:

- Consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur during plant shutdown conditions.
- Initiating conditions and example emergency action levels that fully address conditions that may be postulated to occur at permanently Defueled Stations and Independent Spent Fuel Storage Installations (ISFSIs).
- Simplifying the fission product barrier EAL threshold for a Site Area Emergency.

Subsequently, Revision 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 (ADAMS Accession Number ML12326A805) (ref. 4.1.1), ONS conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

Fission product barrier thresholds represent threats to the defense in depth design concept that precludes the release of radioactive fission products to the environment. This concept relies on multiple physical barriers, any one of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment.

Many of the EALs derived from the NEI methodology are fission product barrier threshold based. That is, the conditions that define the EALs are based upon thresholds that represent the loss or potential loss of one or more of the three fission product barriers. "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. A "Loss" threshold means the barrier no longer assures containment of radioactive materials. A "Potential Loss" threshold implies an increased probability of barrier loss and decreased certainty of maintaining the barrier.

The primary fission product barriers are:

A. <u>Fuel Clad (FC):</u> The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.

B. <u>Reactor Coolant System (RCS):</u> 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.

C. <u>Containment (CMT):</u> The Containment (Reactor Building) Barrier includes the Reactor Building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the Reactor Building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the Emergency Classification Level (ECL) from Alert to a Site Area Emergency or a General Emergency.

2.3 Fission Product Barrier Classification Criteria

The following criteria are the bases for event classification related to fission product barrier loss or potential loss:

<u>Alert:</u>

Any loss or any potential loss of either Fuel Clad or RCS Barrier

<u>Site Area Emergency:</u>

Loss or potential loss of any two barriers

<u>General Emergency:</u>

Loss of any two barriers and loss or potential loss of the third barrier

2.4 EAL Organization

The ONS EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
  - EALs applicable under <u>any</u> plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
  - EALs applicable only under <u>hot</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Hot Standby, Startup, or Power Operation mode.
  - EALs applicable only under <u>cold</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling mode or No Mode.

  The purpose of the groups is to avoid review of hot condition EALs when the plant is in a cold condition and avoid review of cold condition EALs when the plant is in a hot condition. This approach significantly minimizes the total number of EALs that must be reviewed by the EAL-user for a given plant condition, reduces EAL-user reading burden and, thereby, speeds identification of the EAL that applies to the emergency.

- Within each group, assignment of EALs to categories and subcategories:

Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. The ONS EAL categories are aligned to and represent the NEI 99-01 "Recognition Categories." Subcategories are used in the ONS scheme as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The ONS EAL categories and subcategories are listed below.

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
<u>Any Operating Mode:</u>	
R – Abnormal **R**ad Levels / Rad Effluent	1 – Radiological Effluent 2 – Irradiated Fuel Event 3 – Area Radiation Levels
H – **H**azards and Other Conditions Affecting Plant Safety	1 – Security 2 – Seismic Event 3 – Natural or Technological Hazard 4 – Fire 5 – Hazardous Gas 6 – Control Room Evacuation 7 – Emergency Coordinator Judgment
E – Independent Spent Fuel Storage Installation (ISFSI)	1 – Confinement Boundary
<u>Hot Conditions:</u>	
S – **S**ystem Malfunction	1 – Loss of Essential AC Power 2 – Loss of Vital DC Power 3 – Loss of Control Room Indications 4 – RCS Activity 5 – RCS Leakage 6 – RPS Failure 7 – Loss of Communications 8 – Containment Failure 9 – Hazardous Event Affecting Safety Systems
F – **F**ission Product Barrier Degradation	None
<u>Cold Conditions:</u>	
C – **C**old Shutdown / Refueling System Malfunction	1 – RCS Level 2 – Loss of Essential AC Power 3 – RCS Temperature 4 – Loss of Vital DC Power 5 – Loss of Communications 6 – Hazardous Event Affecting Safety Systems

The primary tool for determining the emergency classification level is the EAL Classification Matrix. The user of the EAL Classification Matrix may (but is not required to) consult the EAL Technical Bases in order to obtain additional information concerning the EALs under classification consideration. The user should consult Section 3.0 and Attachments 1 & 2 of this document for such information.

2.5 Technical Bases Information

EAL technical bases are provided in Attachment 1 for each EAL according to EAL group (Any, Hot, Cold), EAL category (R, C, H, S, E and F) and EAL subcategory. A summary explanation of each category and subcategory is given at the beginning of the technical bases discussions of the EALs included in the category. For each EAL, the following information is provided:

<u>Category Letter & Title</u>

<u>Subcategory Number & Title</u>

<u>Initiating Condition (IC)</u>

Site-specific description of the generic IC given in NEI 99-01 Rev. 6.

<u>EAL Identifier (enclosed in rectangle)</u>

Each EAL is assigned a unique identifier to support accurate communication of the emergency classification to onsite and offsite personnel. Four characters define each EAL identifier:

1. First character (letter): Corresponds to the EAL category as described above (R, C, H, S, E or F)
2. Second character (letter): The emergency classification (G, S, A or U)

   G = General Emergency
   S = Site Area Emergency
   A = Alert
   U = Unusual Event

3. Third character (number): Subcategory number within the given category. Subcategories are sequentially numbered beginning with the number one (1). If a category does not have a subcategory, this character is assigned the number one (1).
4. Fourth character (number): The numerical sequence of the EAL within the EAL subcategory. If the subcategory has only one EAL, it is given the number one (1).

<u>Classification (enclosed in rectangle):</u>

Unusual Event (U), Alert (A), Site Area Emergency (S) or General Emergency (G)

<u>EAL (enclosed in rectangle)</u>

Exact wording of the EAL as it appears in the EAL Classification Matrix

<u>Mode Applicability</u>

One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Power Operation, 2 - Startup, 3 – Hot Standby, 4 - Hot Shutdown, 5 - Cold Shutdown, 6 - Refueling, NM – No Mode, or Any. (See Section 2.6 for operating mode definitions)

<u>Definitions:</u>

If the EAL wording contains a defined term, the definition of the term is included in this section. These definitions can also be found in Section 5.1.

Basis:

A basis section that provides ONS-relevant information concerning the EAL as well as a description of the rationale for the EAL as provided in NEI 99-01 Rev. 6.

ONS Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.6 Operating Mode Applicability (ref. 4.1.6)

1 Power Operation

$K_{eff} \geq 0.99$ and reactor thermal power > 5%

2 Startup

$K_{eff} \geq 0.99$ and reactor thermal power $\leq$ 5%

3 Hot Standby

$K_{eff} < 0.99$ and average coolant temperature $\geq$ 250°F

4 Hot Shutdown

$K_{eff} < 0.99$ and average coolant temperature 250°F > T_{avg} > 200°F and all reactor vessel head closure bolts fully tensioned

5 Cold Shutdown

$K_{eff} < 0.99$ and average coolant temperature $\leq$ 200°F and all reactor vessel head closure bolts fully tensioned

6 Refueling

One or more reactor vessel head closure bolts less than fully tensioned

NM No Mode

Reactor vessel contains no irradiated fuel

The plant operating mode that exists at the time that the event occurs (prior to any protective system or operator action being initiated in response to the condition) should be compared to the mode applicability of the EALs. If a lower or higher plant operating mode is reached before the emergency classification is made, the declaration shall be based on the mode that existed at the time the event occurred.

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Emergency Coordinator must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes, and the informing basis information. In the Recognition Category F matrices, EALs are based on loss or potential loss of Fission Product Barrier Thresholds.

3.1.1 Classification Timeliness

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, "Interim Staff Guidance, Emergency Planning for Nuclear Power Plants" (ref. 4.1.9).

3.1.2 Valid Indications

All emergency classification assessments shall be based upon valid indications, reports or conditions. A valid indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy.

An indication, report, or condition is considered to be valid when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

3.1.3 Imminent Conditions

For ICs and EALs that have a stipulated time duration (e.g., 15 minutes, 30 minutes, etc.), the Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time. If an ongoing radiological release is detected and the release start time is unknown, it should be assumed that the release duration specified in the IC/EAL has been exceeded, absent data to the contrary.

3.1.4 Planned vs. Unplanned Events

A planned work activity that results in an expected event or condition which meets or exceeds an EAL does not warrant an emergency declaration provided that: 1) the activity proceeds as planned, and 2) the plant remains within the limits imposed by the operating license. Such activities include planned work to test, manipulate, repair, maintain or modify a system or component. In these cases, the controls associated with the planning, preparation and execution of the work will ensure that compliance is maintained with all aspects of the operating license provided that the activity proceeds and concludes as expected. Events or conditions of this type may be subject to the reporting requirements of 10 § CFR 50.72 (ref. 4.1.4).

3.1.5 Classification Based on Analysis

The assessment of some EALs is based on the results of analyses that are necessary to ascertain whether a specific EAL threshold has been exceeded (e.g., dose assessments, chemistry sampling, RCS leak rate calculation, etc.). For these EALs, the EAL wording or the associated basis discussion will identify the necessary analysis. In these cases, the 15-minute declaration period starts with the availability of the analysis results that show the threshold to be exceeded (i.e., this is the time that the EAL information is first available). The NRC expects licensees to establish the capability to initiate and complete EAL-related analyses within a reasonable period of time (e.g., maintain the necessary expertise on-shift).

3.1.6 Emergency Coordinator Judgment

While the EALs have been developed to address a full spectrum of possible events and conditions which may warrant emergency classification, a provision for classification based on operator/management experience and judgment is still necessary. The NEI 99-01 EAL scheme provides the Emergency Coordinator with the ability to classify events and conditions based upon judgment using EALs that are consistent with the ECL definitions (refer to Category H). The Emergency Coordinator will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated in the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

3.2 Classification Methodology

To make an emergency classification, the user will compare an event or condition (i.e., the relevant plant indications and reports) to an EAL(s) and determine if the EAL has been met or exceeded. The evaluation of an EAL must be consistent with the related Operating Mode Applicability and Notes. If an EAL has been met or exceeded, the associated IC is likewise met, the emergency classification process "clock" starts, and the ECL must be declared in accordance with plant procedures no later than fifteen minutes after the process "clock" started.

When assessing an EAL that specifies a time duration for the off-normal condition, the "clock" for the EAL time duration runs concurrently with the emergency classification process "clock." For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01 (ref. 4.1.9).

3.2.1 Classification of Multiple Events and Conditions

When multiple emergency events or conditions are present, the user will identify all met or exceeded EALs. The highest applicable ECL identified during this review is declared. For example:

- If an Alert EAL and a Site Area Emergency EAL are met, whether at one unit or at two different units, a Site Area Emergency should be declared.

There is no "additive" effect from multiple EALs meeting the same ECL. For example:

- If two Alert EALs are met, whether at one unit or at two different units, an Alert should be declared.

Related guidance concerning classification of rapidly escalating events or conditions is provided in Regulatory Issue Summary (RIS) 2007-02, *Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events* (ref. 4.1.2).

3.2.2 Consideration of Mode Changes During Classification

The mode in effect at the time that an event or condition occurred, and prior to any plant or operator response, is the mode that determines whether or not an IC is applicable. If an event or condition occurs, and results in a mode change before the emergency is declared, the emergency classification level is still based on the mode that existed at the time that the event or condition was initiated (and not when it was declared). Once a different mode is reached, any new event or condition, not related to the original event or condition, requiring emergency classification should be evaluated against the ICs and EALs applicable to the operating mode at the time of the new event or condition.

For events that occur in Cold Shutdown or Refueling, escalation is via EALs that are applicable in the Cold Shutdown or Refueling modes, even if Hot Shutdown (or a higher mode) is entered during the subsequent plant response. In particular, the fission product barrier EALs are applicable only to events that initiate in the Hot Shutdown mode or higher.

3.2.3 Classification of Imminent Conditions

Although EALs provide specific thresholds, the Emergency Coordinator must remain alert to events or conditions that could lead to meeting or exceeding an EAL within a relatively short period of time (i.e., a change in the ECL is IMMINENT). If, in the judgment of the Emergency Coordinator, meeting an EAL is IMMINENT, the emergency classification should be made as if the EAL has been met. While applicable to all emergency classification levels, this approach is particularly important at the higher emergency classification levels since it provides additional time for implementation of protective measures.

3.2.4 Emergency Classification Level Upgrading and Downgrading

An ECL may be downgraded when the event or condition that meets the highest IC and EAL no longer exists, and other site-specific downgrading requirements are met. If downgrading the ECL is deemed appropriate, the new ECL would then be based on a lower applicable IC(s) and EAL(s). The ECL may also simply be terminated.

As noted above, guidance concerning classification of rapidly escalating events or conditions is provided in RIS 2007-02 (ref. 4.1.2).

3.2.5 Classification of Short-Lived Events

Event-based ICs and EALs define a variety of specific occurrences that have potential or actual safety significance. By their nature, some of these events may be short-lived and, thus, over before the emergency classification assessment can be completed. If an event occurs that meets or exceeds an EAL, the associated ECL must be declared regardless of its continued presence at the time of declaration. Examples of such events include an earthquake or a failure of the reactor protection system to automatically trip the reactor followed by a successful manual trip.

3.2.6 Classification of Transient Conditions

Many of the ICs and/or EALs employ time-based criteria. These criteria will require that the IC/EAL conditions be present for a defined period of time before an emergency declaration is warranted. In cases where no time-based criterion is specified, it is recognized that some transient conditions may cause an EAL to be met for a brief period of time (e.g., a few seconds to a few minutes). The following guidance should be applied to the classification of these conditions.

EAL momentarily met during expected plant response - In instances where an EAL is briefly met during an expected (normal) plant response, an emergency declaration is not warranted provided that associated systems and components are operating as expected, and operator actions are performed in accordance with procedures.

EAL momentarily met but the condition is corrected prior to an emergency declaration – If an operator takes prompt manual action to address a condition, and the action is successful in correcting the condition prior to the emergency declaration, then the applicable EAL is not considered met and the associated emergency declaration is not required. For illustrative purposes, consider the following example:

> An ATWS occurs and the high pressure ECCS systems fail to automatically start. Reactor vessel level rapidly decreases and the plant enters an inadequate core cooling condition (a potential loss of both the Fuel Clad and RCS Barriers). If an operator manually starts a high pressure ECCS system in accordance with an EOP step and clears the inadequate core cooling condition prior to an emergency declaration, then the classification should be based on the ATWS only.

It is important to stress that the 15-minute emergency classification assessment period (process clock) is not a "grace period" during which a classification may be delayed to allow the performance of a corrective action that would obviate the need to classify the event. Emergency classification assessments must be deliberate and timely, with no undue delays. The provision discussed above addresses only those rapidly evolving situations when an operator is able to take a successful corrective action prior to the Emergency Coordinator completing the review and steps necessary to make the emergency declaration. This provision is included to ensure that any public protective actions resulting from the emergency classification are truly warranted by the plant conditions.

3.2.7 After-the-Fact Discovery of an Emergency Event or Condition

In some cases, an EAL may be met but the emergency classification was not made at the time of the event or condition. This situation can occur when personnel discover that an event or condition existed which met an EAL, but no emergency was declared, and the event or condition no longer exists at the time of discovery. This may be due to the event or condition not being recognized at the time or an error that was made in the emergency classification process.

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 (ref. 4.1.3) is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR § 50.72 (ref. 4.1.4) within one hour of the discovery of the undeclared event or condition. The licensee should also notify appropriate State and local agencies in accordance with the agreed upon arrangements.

3.2.8 Retraction of an Emergency Declaration

Guidance on the retraction of an emergency declaration reported to the NRC is discussed in NUREG-1022 (ref. 4.1.3).

4.0 REFERENCES

4.1 Developmental

4.1.1 NEI 99-01 Revision 6, Methodology for the Development of Emergency Action Levels for Non-Passive Reactors, ADAMS Accession Number ML12326A805

4.1.2 RIS 2007-02 Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events, February 2, 2007.

4.1.3 NUREG-1022 Event Reporting Guidelines: 10CFR50.72 and 50.73

4.1.4 10 § CFR 50.72 Immediate Notification Requirements for Operating Nuclear Power Reactors

4.1.5 10 § CFR 50.73 License Event Report System

4.1.6 Technical Specifications Table 1.1-1 Modes

4.1.7 OP/1,2,3/A/1502/009 Containment Closure Control

4.1.8 Procedure Writer's Manual, Revision 012

4.1.9 NSIR/DPR-ISG-01 Interim Staff Guidance, Emergency Planning for Nuclear Power Plants

4.1.10 Oconee Nuclear Site Emergency Plan

4.1.11 S.D.1.3.5 Shutdown Protection Plan

4.1.12 Duke Energy Physical Security Plan for ONS

4.2 Implementing

4.2.1 RP/0/A/1000/001 Emergency Classification

4.2.2 NEI 99-01 Rev. 6 to ONS EAL Comparison Matrix

4.2.3 ONS EAL Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

Selected terms used in Initiating Condition and Emergency Action Level statements are set in all capital letters (e.g., ALL CAPS). These words are defined terms that have specific meanings as used in this document. The definitions of these terms are provided below.

Alert

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 small fractions of the EPA Protective Action Guideline exposure levels.

Confinement Boundary

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the ONS ISFSI, Confinement Boundary is comprised of the DSC (dry shielded canister) shell, inner bottom cover plate, inner top cover plate, siphon & vent block, siphon & vent port cover plate, and the welds that join them together.

Containment Closure

The action to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under all plant conditions up to and including a loss of decay heat removal or fuel handling accident inside containment (ref. 4.1.11).

As applied to ONS, Containment Closure is established when the requirements of OP/1,2,3/A/1502/009, Containment Closure Control, are met (ref. 4.1.7).

Emergency Action Level (EAL)

A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.

Emergency Classification Level (ECL)

One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are:

- Unusual Event (UE)
- Alert
- Site Area Emergency (SAE)
- General Emergency (GE)

EPA PAGs

Environment Protection Agency Protective Action Guidelines. The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs requires ONS to recommend protective actions for the general public to offsite planning agencies.

Explosion

A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

Faulted

The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.

Fire

Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

Fission Product Barrier Threshold

A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.

Flooding

A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

General Emergency

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 actions that result 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.

Hostage

A person(s) held as leverage against the station to ensure that demands will be met by the station.

Hostile Action

An act toward ONS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on ONS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

Hostile Force

One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

Imminent

The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

Independent Spent Fuel Storage Installation (ISFSI)

A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

Impede(d)

Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

Initiating Condition (IC)

An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.

Intrusion

The act of entering without authorization. Discovery of a bomb in a specified area is indication of intrusion into that area by a hostile force.

Maintain

Take appropriate action to hold the value of an identified parameter within specified limits.

Normal Levels

As applied to radiological IC/EALs, the highest reading in the past twenty-four hours excluding the current peak value.

Owner Controlled Area

Area outside the PROTECTED AREA fence that immediately surrounds the plant. Access to this area is generally restricted to those entering on official business.

Projectile

An object directed toward a Nuclear Power Plant that could cause concern for its continued operability, reliability, or personnel safety.

Protected Area

That part of the nuclear station property consisting of the Reactor, Auxiliary, Turbine, and Service Building and grounds, contained within the owner controlled security fence (ref. 4.1.10).

RCS Intact

The RCS should be considered intact when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams, pressurizer manway and safeties installed).

Reduced Inventory

Condition with fuel in the reactor vessel and the level lower than approximately three feet below the reactor vessel flange (RCS level < 50" on LT-5) (ref. 4.1.11).

Refueling Pathway

The spent fuel pool and/or fuel transfer canal comprise the refueling pathway.

Restore

Take the appropriate action required to return the value of an identified parameter to the applicable limits.

Ruptured

The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

Safety System

A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Security Condition

Any security event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A security condition does not involve a hostile action.

Site Area Emergency

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 actions that result 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 Guidelines exposure levels beyond the site boundary.

Site Boundary

That area, including the PROTECTED AREA, in which Duke Energy has the authority to control all activities including exclusion or removal of personnel and property (1 mile radius) from the center of Unit 2. (ref. 4.1.10).

Unisolable

An open or breached system line that cannot be isolated, remotely or locally.

Unplanned

A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Unusual Event

Events are in progress or have occurred which indicate a potential degradation in 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.

Valid

An indication, report, or condition, is considered to be valid when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

Visible Damage

Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

5.2 Acronyms and Abbreviations

°F	Degrees Fahrenheit
°	Degrees
AC	Alternating Current
AP	Abnormal Operating Procedure
ATWS	Anticipated Transient Without Scram
BWST	Borated Water Storage Tank
CETC	Core Exit Thermocouple
CDE	Committed Dose Equivalent
CFR	Code of Federal Regulations
CMT	Containment
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DC	Direct Current
DSC	Dry Shielded Canister
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
ECL	Emergency Classification Level
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
ERG	Emergency Response Guideline
EPIP	Emergency Plan Implementing Procedure
ESF	Engineered Safety Feature
FAA	Federal Aviation Administration
FBI	Federal Bureau of Investigation
FEMA	Federal Emergency Management Agency
GE	General Emergency
HPI	High Pressure Injection
IC	Initiating Condition
IPEEE	Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI	Independent Spent Fuel Storage Installation
K_{eff}	Effective Neutron Multiplication Factor
LCO	Limiting Condition of Operation

LEC .... Law Enforcement Center
LER .... Licensee Event Report
LOCA .... Loss of Coolant Accident
LWR .... Light Water Reactor
MPC .... Maximum Permissible Concentration/Multi-Purpose Canister
mR, mRem, mrem, mREM .... milli-Roentgen Equivalent Man
MSL .... Main Steam Line
MW .... Megawatt
NEI .... Nuclear Energy Institute
NESP .... National Environmental Studies Project
NM .... No Mode
NPP .... Nuclear Power Plant
NRC .... Nuclear Regulatory Commission
NORAD .... North American Aerospace Defense Command
(NO)UE .... Notification of Unusual Event
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM .... Off-site Dose Calculation Manual
ORO .... Offsite Response Organization
PA .... Protected Area
PAG .... Protective Action Guideline
PRA .... Probabilistic Risk Assessment
PSA .... Probabilistic Safety Assessment
PWR .... Pressurized Water Reactor
PSIG .... Pounds per Square Inch Gauge
PSW .... Protected Service Water
R .... Roentgen
RCS .... Reactor Coolant System
Rem, rem, REM .... Roentgen Equivalent Man
Rep CET .... Representative Core Exit Thermocouples
RETS .... Radiological Effluent Technical Specifications
RPS .... Reactor Protective System
RV .... Reactor Vessel
RVLIS .... Reactor Vessel Level Indicating System

SAR .... Safety Analysis Report
SBO .... Station Blackout
SCBA .... Self-Contained Breathing Apparatus
SG .... Steam Generator
SLC .... Selected License Commitment
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
TEDE .... Total Effective Dose Equivalent
TSC .... Technical Support Center
UFSAR .... Updated Final Safety Analysis Report

6.0 ONS-TO-NEI 99-01 Rev. 6 EAL CROSS-REFERENCE

This cross-reference is provided to facilitate association and location of an ONS EAL within the NEI 99-01 IC/EAL identification scheme. Further information regarding the development of the ONS EALs based on the NEI guidance can be found in the EAL Comparison Matrix.

ONS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
RU1.1	AU1	1, 2
RU1.2	AU1	3
RU2.1	AU2	1
RA1.1	AA1	1
RA1.2	AA1	2
RA1.3	AA1	3
RA1.4	AA1	4
RA2.1	AA2	1
RA2.2	AA2	2
RA2.3	AA2	3
RA3.1	AA3	1
RA3.2	AA3	2
RS1.1	AS1	1
RS1.2	AS1	2
RS1.3	AS1	3
RS2.1	AS2	1
RG1.1	AG1	1
RG1.2	AG1	2
RG1.3	AG1	3
RG2.1	AG2	1
CU1.1	CU1	1
CU1.2	CU1	2

ONS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
CU2.1	CU2	1
CU3.1	CU3	1
CU3.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1, 2, 3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA3.1	CA3	1, 2
CA6.1	CA6	1
CS1.1	CS1	3
CG1.1	CG1	2
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1
HU1.2	HU1	2
HU1.3	HU1	3
HU2.1	HU2	1
HU3.1	HU3	1
HU3.2	HU3	2
HU3.3	HU3	3
HU3.4	HU3	4
HU4.1	HU4	1
HU4.2	HU4	2

ONS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
HU4.3	HU4	3
HU4.4	HU4	4
HU7.1	HU7	1
HA1.1	HA1	1
HA1.2	HA1	2
HA5.1	HA5	1
HA6.1	HA6	1
HA7.1	HA7	1
HS1.1	HS1	1
HS6.1	HS6	1
HS7.1	HS7	1
HG1.1	HG1	1
HG7.1	HG7	1
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	2
SU5.1	SU4	1, 2, 3
SU6.1	SU5	1
SU6.2	SU5	2
SU7.1	SU6	1, 2, 3
SU8.1	SU7	1, 2
SA1.1	SA1	1
SA3.1	SA2	1
SA6.1	SA5	1
SA9.1	SA9	1

ONS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1
SG1.1	SG1	1
SG1.2	SG8	1
EU1.1	EU1	1

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Fission Product Barrier Matrix and Basis

7.3 Attachment 3, Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases

Category R – Abnormal Rad Release / Rad Effluent

EAL Group: ANY (EALs in this category are applicable to any plant condition, hot or cold.)

Many EALs are based on actual or potential degradation of fission product barriers because of the elevated potential for offsite radioactivity release. Degradation of fission product barriers though is not always apparent via non-radiological symptoms. Therefore, direct indication of elevated radiological effluents or area radiation levels are appropriate symptoms for emergency classification.

At lower levels, abnormal radioactivity releases may be indicative of a failure of containment systems or precursors to more significant releases. At higher release rates, offsite radiological conditions may result which require offsite protective actions. Elevated area radiation levels in the plant may also be indicative of the failure of containment systems or preclude access to plant vital equipment necessary to ensure plant safety.

Events of this category pertain to the following subcategories:

1. Radiological Effluent

Direct indication of effluent radiation monitoring systems provides a rapid assessment mechanism to determine releases in excess of classifiable limits. Projected offsite doses, actual offsite field measurements or measured release rates via sampling indicate doses or dose rates above classifiable limits.

2. Irradiated Fuel Event

Conditions indicative of a loss of adequate shielding or damage to irradiated fuel may preclude access to vital plant areas or result in radiological releases that warrant emergency classification.

3. Area Radiation Levels

Sustained general area radiation levels which may preclude access to areas requiring continuous occupancy also warrant emergency classification.

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the SLC/TS limits for 60 minutes or longer

EAL:

RU1.1	**Unusual Event**
Reading on **any** Table R-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3)	

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Unit 1/2/3 Plant Vent	RIA-45	----	----	----	1.41E+5 cpm
Gaseous	Unit 1/2/3 Plant Vent	RIA-46	3.00E+5 cpm	3.00E+4 cpm	3.00E+3 cpm	----
Liquid	Liquid Radwaste Discharge	RIA-33	----	----	----	4.79E+5 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

The column "UE" release values in Table R-1 represent two times the appropriate SLC and Technical Specification release rate and concentration limits associated with the specified monitors (ref. 1, 2, 3, 4, 5, 6).

<u>Gaseous Releases</u>

Instrumentation that may be used to assess this EAL: (ref. 1):

- Unit 1/2/3 Plant Vent Noble Gas Low Monitor – RIA-45(L)

Liquid Releases

Instrumentation that may be used to assess this EAL: (ref. 1):

- Liquid Radwaste Discharge Monitor – RIA-33 (batch release)

This IC addresses a potential decrease in the level of safety of the plant as indicated by a low-level radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release). It includes any gaseous or liquid radiological release, monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.

Nuclear power plants incorporate design features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, and to control and monitor intentional releases. The occurrence of an extended, uncontrolled radioactive release to the environment is indicative of degradation in these features and/or controls.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Releases should not be prorated or averaged. For example, a release exceeding 4 times release limits for 30 minutes does not meet the EAL.

This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways.

Escalation of the emergency classification level would be via IC RA1.

ONS Basis Reference(s):

1. UFSAR Section 11.5, Process and Effluent Radiological Monitoring and Sampling Systems
2. Oconee Nuclear Station Units 1, 2 and 3 Offsite Dose Calculation Manual
3. ONS-SLC 16.11.1 Radioactive Liquid Effluents
4. ONS-SLC 16.11.2 Radioactive Gaseous Effluents
5. EP-EALCALC-ONS-1401 ONS Radiological Effluent EAL Values, Rev. 0
6. Technical Specification Section 5.5.5
7. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the SLC/TS limits for 60 minutes or longer.

EAL:

RU1.2	**Unusual Event**
Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x SLC/TS limits for ≥ 60 min. (Notes 1, 2)	

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

None

Basis:

This IC addresses a potential decrease in the level of safety of the plant as indicated by a low-level radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release). It includes any gaseous or liquid radiological release, monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.

Nuclear power plants incorporate design features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, and to control and monitor intentional releases. The occurrence of an extended, uncontrolled radioactive release to the environment is indicative of degradation in these features and/or controls.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

Releases should not be prorated or averaged. For example, a release exceeding 4 times release limits for 30 minutes does not meet the EAL.

This EAL addresses uncontrolled gaseous or liquid releases that are detected by sample analyses or environmental surveys, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.).

Escalation of the emergency classification level would be via IC RA1.

ONS Basis Reference(s):

1. UFSAR Section 11.5, Process and Effluent Radiological Monitoring and Sampling Systems
2. Oconee Nuclear Station Units 1, 2 and 3 Offsite Dose Calculation Manual
3. ONS-SLC 16.11.1 Radioactive Liquid Effluents
4. ONS-SLC 16.11.2 Radioactive Gaseous Effluents
5. AD-RP-ALL-2003 Investigation of Unusual Radiological Occurrences
6. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.1 Alert

Reading on **any** Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4 The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Unit 1/2/3 Plant Vent	RIA-45	----	----	----	1.41E+5 cpm
Gaseous	Unit 1/2/3 Plant Vent	RIA-46	3.00E+5 cpm	3.00E+4 cpm	3.00E+3 cpm	----
Liquid	Liquid Radwaste Discharge	RIA-33	----	----	----	4.79E+5 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL addresses gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either:

- 10 mRem TEDE
- 50 mRem CDE Thyroid

The column "ALERT" gaseous effluent release values in Table R-1 correspond to calculated doses of 1% (10% of the SAE thresholds) of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1, 2, 3, 4).

Instrumentation that may be used to assess this EAL: (ref. 1):

- Unit 1/2/3 Plant Vent Noble Gas Medium Monitor – RIA-46(M)

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

ONS Basis Reference(s):

1. UFSAR Section 11.5 Process and Effluent Radiological Monitoring and Sampling Systems
2. Oconee Nuclear Station Units 1, 2 and 3 Offsite Dose Calculation Manual
3. EP-EALCALC-ONS-1401 ONS Radiological Effluent EAL Values, Rev. 0
4. SDQA-70400-COM, "Unified RASCAL Interface (URI)"
5. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.2	**Alert**
Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)	

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That area, including the PROTECTED AREA, in which Duke Energy has the authority to control all activities including exclusion or removal of personnel and property (1 mile radius) from the center of Unit 2.

Basis:

Dose assessments are performed by computer-based methods (ref. 1, 2).

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

ONS Basis Reference(s):

1. RP/0/A/1000/001 Emergency Classification
2. AD-EP-ALL-0202 Emergency Response Offsite Dose Assessment
3. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.3 Alert

Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure (Notes 1, 2)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That area, including the PROTECTED AREA, in which Duke Energy has the authority to control all activities including exclusion or removal of personnel and property (1 mile radius) from the center of Unit 2.

Basis:

Dose assessments based on liquid releases are performed per Offsite Dose Calculation Manual (ref. 1).

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

ONS Basis Reference(s):

1. Oconee Nuclear Station Units 1, 2 and 3 Offsite Dose Calculation Manual
2. NEI 99-01 AA1

ATTACHMENT 1
EAL Bases

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.4 Alert

Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation.

(Notes 1, 2)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That area, including the PROTECTED AREA, in which Duke Energy has the authority to control all activities including exclusion or removal of personnel and property (1 mile radius) from the center of Unit 2.

Basis:

AD-EP-ALL-0203 Protocol for the Field Monitoring Coordinator During Emergency Conditions provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

ONS Basis Reference(s):

1. AD-EP-ALL-0203 Protocol for the Field Monitoring Coordinator During Emergency Conditions
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE

EAL:

RS1.1 Site Area Emergency

Reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Unit 1/2/3 Plant Vent	RIA-45	----	----	----	1.41E+5 cpm
Gaseous	Unit 1/2/3 Plant Vent	RIA-46	3.00E+5 cpm	3.00E+4 cpm	3.00E+3 cpm	----
Liquid	Liquid Radwaste Discharge	RIA-33	----	----	----	4.79E+5 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL addresses gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either:

- 100 mRem TEDE
- 500 mRem CDE Thyroid

The column "SAE" gaseous effluent release value in Table R-1 corresponds to calculated doses of 10% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1, 3).

Instrumentation that may be used to assess this EAL: (ref. 2):

- Unit 1/2/3 Plant Vent Noble Gas Medium Monitor – RIA-46(M)

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RG1.

ONS Basis Reference(s):

1. EP-EALCALC-ONS-1401 ONS Radiological Effluent EAL Values, Rev. 0
2. UFSAR Section 11.5 Process and Effluent Radiological Monitoring and Sampling Systems
3. SDQA-70400-COM, "Unified RASCAL Interface (URI)"
4. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE

EAL:

RS1.2	**Site Area Emergency**
Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)	

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That area, including the PROTECTED AREA, in which Duke Energy has the authority to control all activities including exclusion or removal of personnel and property (1 mile radius) from the center of Unit 2.

Basis:

Dose assessments are performed by computer-based methods (ref. 1, 2).

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RG1.

ONS Basis Reference(s):

1. RP/0/A/1000/001 Emergency Classification
2. AD-EP-ALL-0202 Emergency Response Offsite Dose Assessment
3. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE

EAL:

RS1.3 Site Area Emergency

Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation.

(Notes 1, 2)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That area, including the PROTECTED AREA, in which Duke Energy has the authority to control all activities including exclusion or removal of personnel and property (1 mile radius) from the center of Unit 2.

Basis:

AD-EP-ALL-0203 Protocol for the Field Monitoring Coordinator During Emergency Conditions provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Escalation of the emergency classification level would be via IC RG1.

ONS Basis Reference(s):

1. AD-EP-ALL-0203 Protocol for the Field Monitoring Coordinator During Emergency Conditions
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

EAL:

RG1.1	**General Emergency**
Reading on **any** Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4)	

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Unit 1/2/3 Plant Vent	RIA-45	----	----	----	1.41E+5 cpm
Gaseous	Unit 1/2/3 Plant Vent	RIA-46	3.00E+5 cpm	3.00E+4 cpm	3.00E+3 cpm	----
Liquid	Liquid Radwaste Discharge	RIA-33	----	----	----	4.79E+5 cpm

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL addresses gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either:

- 1000 mRem TEDE
- 5000 mRem CDE Thyroid

The column "GE" gaseous effluent release values in Table R-1 correspond to calculated doses of 100% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1, 3).

Instrumentation that may be used to assess this EAL: (ref. 2):

- Unit 1/2/3 Plant Vent Noble Gas Medium Monitor – RIA-46(M)

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

ONS Basis Reference(s):

1. EP-EALCALC-ONS-1401 ONS Radiological Effluent EAL Values, Rev. 0
2. UFSAR Section 11.5 Process and Effluent Radiological Monitoring and Sampling Systems
3. SDQA-70400-COM, "Unified RASCAL Interface (URI)"
4. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

EAL:

RG1.2	**General Emergency**
Dose assessment using actual meteorology indicates doses > 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)	

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That area, including the PROTECTED AREA, in which Duke Energy has the authority to control all activities including exclusion or removal of personnel and property (1 mile radius) from the center of Unit 2.

Basis:

Dose assessments are performed by computer-based methods (ref. 1, 2).

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

ONS Basis Reference(s):

1. RP/0/A/1000/001 Emergency Classification
2. AD-EP-ALL-0202 Emergency Response Offsite Dose Assessment
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

EAL:

RG1.3 General Emergency

Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates > 1,000 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 5,000 mrem for 60 min. of inhalation.

(Notes 1, 2)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That area, including the PROTECTED AREA, in which Duke Energy has the authority to control all activities including exclusion or removal of personnel and property (1 mile radius) from the center of Unit 2.

Basis:

AD-EP-ALL-0203 Protocol for the Field Monitoring Coordinator During Emergency Conditions provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem while the 5,000 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

ONS Basis Reference(s):

1. AD-EP-ALL-0203 Protocol for the Field Monitoring Coordinator During Emergency Conditions
2. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: UNPLANNED loss of water level above irradiated fuel

EAL:

RU2.1 Unusual Event

UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication

AND

UNPLANNED rise in corresponding area radiation levels as indicated by **any** of the following radiation monitors:

- RIA-3 RB Refueling Deck Shield Wall
- RIA-6 Spent Fuel Building Wall
- Portable area monitors on the main bridge or SFP bridge

Mode Applicability:

All

Definition(s):

UNPLANNED -. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

REFUELING PATHWAY- The spent fuel pool and/or fuel transfer canal comprise the refueling pathway.

Basis:

The spent fuel pool low water level alarm setpoint is actuated at -1.8 ft. below normal level (ref. 1). Water level restoration instructions are performed in accordance with Abnormal Operating Procedures (APs) (ref. 2).

The specified radiation monitors are those expected to see increase area radiation levels as a result of a loss of REFUELING PATHWAY inventory (ref. 3). Increasing radiation indications on these monitors in the absence of indications of decreasing water level are not classifiable under this EAL. Radiation levels in the Reactor Building refueling area are monitored by RIA-3. Radiation levels in the Spent Fuel Pool area are monitored RIA-6. When a fuel bridge is being used to handle fuel, radiation levels are monitored by a portable area monitor mounted on the bridge. (ref. 3, 4)

When the spent fuel pool and reactor cavity are connected, there could exist the possibility of uncovering irradiated fuel. Therefore, this EAL is applicable for conditions in which irradiated fuel is being transferred to and from the reactor vessel and spent fuel pool.

This IC addresses a decrease in water level above irradiated fuel sufficient to cause elevated radiation levels. This condition could be a precursor to a more serious event and is also indicative of a minor loss in the ability to control radiation levels within the plant. It is therefore a potential degradation in the level of safety of the plant.

A water level decrease will be primarily determined by indications from available level instrumentation. Other sources of level indications may include reports from plant personnel (e.g., from a refueling crew) or video camera observations (if available). A significant drop in the water level may also cause an increase in the radiation levels of adjacent areas that can be detected by monitors in those locations.

The effects of planned evolutions should be considered. For example, a refueling bridge area radiation monitor reading may increase due to planned evolutions such as lifting of the reactor vessel head or movement of a fuel assembly. Note that this EAL is applicable only in cases where the elevated reading is due to an unplanned loss of water level.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC RA2.

ONS Basis Reference(s):

1. OP/1/A/6101/009 Alarm Response Guide 1SA-09, A-5; OP/2/A/6102/009; OP/3/A/6103/009
2. AP/1-2,3/A/1700/035 Loss of SPF Cooling and/or Level
3. UFSAR Table 12-3 Area Radiation Monitors
4. OP/1,2,3/A/1502/007, Enclosure 1, Defueling/Refueling Prerequisites
5. NEI 99-01 AU2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel

EAL:

RA2.1	**Alert**
Uncovery of irradiated fuel in the REFUELING PATHWAY	

Mode Applicability:

All

Definition(s):

REFUELING PATHWAY- The spent fuel pool and/or fuel transfer canal comprise the refueling pathway.

Basis:

This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

This EAL escalates from RU2.1 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used (e.g., a boil-off curve). Classification of an event using this EAL should be based on the totality of available indications, reports and observations.

While an area radiation monitor could detect an increase in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC RS1.

ONS Basis Reference(s):

1. AP/1-2,3/A/1700/035 Loss of SPF Cooling and/or Level
2. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel

EAL:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity

AND

HIGH alarm on **any** of the following radiation monitors:

- RIA-3 RB Refueling Deck Shield Wall
- RIA-6 Spent Fuel Building Wall
- RIA-41 Spend Fuel Pool Gas
- RIA-49 RB Gas
- Portable area monitors on the main bridge or SFP bridge

Mode Applicability:

All

Definition(s):

None

Basis:

The specified radiation monitors are those expected to see increase area radiation levels as a result of damage to irradiated fuel. Radiation levels in the Reactor Building refueling area are monitored by RIA-3. Radiation levels in the Spent Fuel Pool area are monitored RIA-6. When a fuel bridge is being used to handle fuel, radiation levels are monitored by a portable area monitor mounted on the bridge. (ref. 1, 2, 3)

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.

This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

This EAL applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with EAL EU1.1.

Escalation of the emergency would be based on either Recognition Category R or C ICs.

This EAL addresses a release of radioactive material caused by mechanical damage to irradiated fuel. Damaging events may include the dropping, bumping or binding of an assembly, or dropping a heavy load onto an assembly. A rise in readings on radiation monitors should be considered in conjunction with in-plant reports or observations of a potential fuel damaging event (e.g., a fuel handling accident).

Escalation of the emergency classification level would be via IC RS1.

ONS Basis Reference(s):

1. OP/1/A/6101/008, Alarm Response Guide 1SA-08 B-9; OP/2/A/6101/008; OP/3/A/6101/008
2. AP/1,2,3/A/1700/018, Abnormal Release of Radioactivity
3. OP/1,2,3/A/1502/007, Enclosure 1, Defueling/Refueling Prerequisites
4. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel

EAL:

RA2.3	**Alert**
Lowering of spent fuel pool level to -13.5 ft.	

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL is applicable once the post-Fukushima level instrumentation becomes operational on its associated unit.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3).

SFP level instruments 1/2/3SFP0010 (primary) and 011 (backup) measure SFP level relative to normal water level (El. 840 ft.) from + 1 ft. to -23.5 ft. (El. 816.4 ft).

For ONS Level 2 corresponds to an indicated water level of -13.5 ft. (El. 826.5 ft.) (ref. 1).

This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

Escalation of the emergency would be based on either Recognition Category R or C ICs.

Spent fuel pool water level at this value is within the lower end of the level range necessary to prevent significant dose consequences from direct gamma radiation to personnel performing operations in the vicinity of the spent fuel pool. This condition reflects a significant loss of spent fuel pool water inventory and thus it is also a precursor to a loss of the ability to adequately cool the irradiated fuel assembles stored in the pool.

Escalation of the emergency classification level would be via IC RS1.

ONS Basis Reference(s):

1. Engineering Change EC 105805 & 105806
2. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Spent fuel pool level at the top of the fuel racks

EAL:

RS2.1	**Site Area Emergency**
Lowering of spent fuel pool level to -23.5 ft.	

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL is applicable once the post-Fukushima level instrumentation becomes operational on its associated unit.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3).

SFP level instruments 1/2/3SFP0010 (primary) and 011 (backup) measure SFP level relative to normal water level (El. 840 ft.) from + 1 ft. to -23.5 ft. (El. 816.4 ft).

For ONS Level 3 corresponds to an indicated water level of -23.5 ft. (El. 816.5 ft.) (ref. 1).

This EAL addresses a significant loss of spent fuel pool inventory control and makeup capability leading to IMMINENT fuel damage. This condition entails major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

It is recognized that this IC would likely not be met until well after another Site Area Emergency IC was met; however, it is included to provide classification diversity.

Escalation of the emergency classification level would be via IC AG1 or RG2.

ONS Basis Reference(s):

1. Engineering Change EC 105805 & 105806
2. NEI 99-01 AS2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer

EAL:

RG2.1	**General Emergency**
Spent fuel pool level **cannot** be restored to at least -23.5 ft. for ≥ 60 min. (Note 1)	

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL is applicable once the post-Fukushima level instrumentation becomes operational on its associated unit.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3).

SFP level instruments 1/2/3SFP0010 (primary) and 011 (backup) measure SFP level relative to normal water level (El. 840 ft.) from + 1 ft. to -23.5 ft. (El. 816.4 ft).

For ONS Level 3 corresponds to an indicated water level of -23.5 ft. (El. 816.5 ft.) (ref. 1).

This EAL addresses a significant loss of spent fuel pool inventory control and makeup capability leading to a prolonged uncovery of spent fuel. This condition will lead to fuel damage and a radiological release to the environment.

It is recognized that this IC would likely not be met until well after another General Emergency IC was met; however, it is included to provide classification diversity.

ONS Basis Reference(s):

1. Engineering Change EC 105805 & 105806
2. NEI 99-01 AG2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

Initiating Condition: Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

RA3.1	**Alert**

Dose rates > 15 mR/hr in **EITHER** of the following areas:

- Control Room (RIA-1)
- Central Alarm Station (by survey)

Mode Applicability:

All

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

Basis:

Areas that meet this threshold include the Control Room (CR) and the Central Alarm Station (CAS). RIA-1 monitors the Control room for area radiation (ref. 1). The CAS is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations.

There are no permanently installed area radiation monitors in the CAS that may be used to assess this EAL threshold. Therefore, this threshold is evaluated using local radiation survey for this area (ref. 1).

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Emergency Coordinator should consider the cause of the increased radiation levels and determine if another IC may be applicable.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

ONS Basis Reference(s):

1. UFSAR Table 12-3 Area Radiation Monitors
2. NEI 99-01 AA3

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

Initiating Condition: Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

RA3.2 Alert

An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 rooms or areas (Note 5)

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then **no** emergency classification is warranted.

Table R-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Turbine Building	1, 2, 3
Equipment and Cable Rooms	1, 2, 3
Auxiliary Building	1, 2, 3, 4, 5
Reactor Buildings	3, 4, 5

Mode Applicability:

All

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

The list of plant rooms or areas with entry-related mode applicability identified specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Emergency Coordinator should consider the cause of the increased radiation levels and determine if another IC may be applicable.

For RA3.2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the increased radiation levels. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., installing temporary shielding, requiring use of non-routine protective equipment, requesting an extension in dose limits beyond normal administrative limits).

An emergency declaration is not warranted if any of the following conditions apply:

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the elevated radiation levels). For example, the plant is in Mode 1 when the radiation increase occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The increased radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.
- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

ONS Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Rooms/Areas Tables R-3 & H-2 Bases
2. NEI 99-01 AA3

Category E – Independent Spent Fuel Storage Installation (ISFSI)

EAL Group: ANY (EALs in this category are applicable to any plant condition, hot or cold.)

An independent spent fuel storage installation (ISFSI) is a complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage. A significant amount of the radioactive material contained within a canister must escape its packaging and enter the biosphere for there to be a significant environmental effect resulting from an accident involving the dry storage of spent nuclear fuel.

A Notification of Unusual Event is declared on the basis of the occurrence of an event of sufficient magnitude that a loaded cask confinement boundary is damaged or violated.

The ONS ISFSI is contained wholly within the plant Protected Area. Therefore a security event related to the ISFSI would be applicable to EALs HU1.1, HA1.1and HS1.1

Category: ISFSI

Subcategory: Confinement Boundary

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Unusual Event

Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask > **any** Table E-1 ISFSI dose limit

Table E-1 ISFSI Dose Limits			
Location	**24PHB**	**37PTH**	**69BTH**
HSM front bird screen	1,050 mrem/hr	1,050 mrem/hr	500 mrem/hr
Outside HSM door	40 mrem/hr	4 mrem/hr	4 mrem/hr
End shield wall exterior	550 mrem/hr	8 mrem/hr	8 mrem/hr

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY - The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the ONS ISFSI, Confinement Boundary is comprised of the DSC (dry shielded canister) shell, inner bottom cover plate, inner top cover plate, siphon & vent block, siphon & vent port cover plate, and the welds that join them together.

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) - A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

Basis:

The ONS ISFSI utilizes the NUHOMS System dry spent fuel storage system for dry spent fuel storage.

The Standardized NUHOMS® System is a horizontal canister system composed of a steel dry shielded canister (DSC), a reinforced concrete horizontal storage module (HSM), and a transfer cask (TC). The welded DSC provides confinement and criticality control for the storage and transfer of irradiated fuel. The concrete module provides radiation shielding while allowing cooling of the DSC and fuel by natural convection during storage (ref. 1, 2). The ONS ISFSI utilizes the 24PHB, 37PTH and 69BTH DSC designs.

Confinement boundary is defined as the barrier(s) between areas containing radioactive substances and the environment. Therefore, damage to a confinement boundary must be a confirmed physical breach between the spent fuel and the environment for the TSC.

The Table E-1 values shown are 2 times the limits specified in the ISFSI Certificate of Compliance Technical Specifications for radiation external to the applicable loaded DSC (ref. 1, 2).

This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage.

The existence of "damage" is determined by radiological survey. The technical specification multiple of "2 times", which is also used in Recognition Category R IC RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

Security-related events for ISFSIs are covered under ICs HU1 and HA1.

ONS Basis Reference(s):

1. USNRC Certificate of Compliance for Spent Fuel Storage Casks, No. 1004, Amendment 13, Attachment A, Technical Specifications for Transnuclear, Inc., Standardized NUHOMS Horizontal Modular Storage System
2. OSC-8716, Oconee ISFSI Dose Rate Evaluations, Rev. 0 (4/29/05)
3. NEI 99-01 E-HU1

Category C – Cold Shutdown / Refueling System Malfunction

EAL Group: Cold Conditions (RCS temperature ≤ 200°F); EALs in this category are applicable only in one or more cold operating modes.

Category C EALs are directly associated with cold shutdown or refueling system safety functions. Given the variability of plant configurations (e.g., systems out-of-service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The cold shutdown and refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, containment closure, and fuel clad integrity for the applicable operating modes (5 - Cold Shutdown, 6 - Refueling, NM – No Mode).

The events of this category pertain to the following subcategories:

1. RCS Level

RCS water level is directly related to the status of adequate core cooling and, therefore, fuel clad integrity.

2. Loss of Essential AC Power

Loss of essential plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite power sources for 4160V AC essential buses.

3. RCS Temperature

Uncontrolled or inadvertent temperature or pressure increases are indicative of a potential loss of safety functions.

4. Loss of Vital DC Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of power to or degraded voltage on the 125V DC vital buses.

5. Loss of Communications

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

6. Hazardous Event Affecting Safety Systems

Certain hazardous natural and technological events may result in visible damage to or degraded performance of safety systems warranting classification.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory for 15 minutes or longer

EAL:

CU1.1	**Unusual Event**
UNPLANNED loss of reactor coolant results in RCS water level less than a required lower limit for ≥ 15 min. (Note 1)	

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

RCS water level less than a required lower limit is meant to be less than the lower end of the level control band being procedurally maintained for the current condition or evolution.

RCS water level instrumentation requirements to begin an RCS inventory reduction with fuel in the core to below 80" (lowered inventory) or 50" (reduced inventory) are the following (ref. 1):

- Both channels of LT-5 prior to reducing RCS inventory below 80".
- Both channels of LT-5 and both hot leg and cold leg ultrasonic monitors prior to reducing RCS inventory below 50".

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor RCS level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

This EAL recognizes that the minimum required RCS level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.

The 15-minute threshold duration allows sufficient time for prompt operator actions to restore and maintain the expected water level. This criterion excludes transient conditions causing a brief lowering of water level.

Continued loss of RCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA3.

ONS Basis Reference(s):

1. S. D. 1.3.5 Shutdown Protection Plan, Section 5.2.7
2. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: UNPLANNED loss of RCS inventory for 15 minutes or longer

EAL:

CU1.2 Unusual Event

RCS level **cannot** be monitored

AND EITHER

- UNPLANNED increase in **any** Table C-1 sump/tank level due to loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

Table C-1 Sumps / Tanks
• RB Normal Sumps
• RB Emergency Sumps
• Core Flood Tank
• Quench Tank
• Low Activity Waste Tank
• High Activity Waste Tank
• Miscellaneous Waste Holdup Tank
• LPI Room Sumps

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

A RCS leak should be considered UNISOLABLE if the leak cannot be isolated within 15 min.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available.

In the Refuel mode, the RCS is not intact and reactor vessel level may be monitored by different means, including the ability to monitor level visually.

In this EAL, all water level indication is unavailable and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of significant leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor RCS level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

This EAL addresses a condition where all means to determine level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels (Table C-1). Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

Continued loss of RCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA3.

ONS Basis Reference(s):

1. AP/1,2,3/A/1700/002 Excessive RCS Leakage
2. AP/1-2,3/A/1700/030 Auxiliary Building Flood
3. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory

EAL:

CA1.1	**Alert**
Loss of RCS inventory as indicated by RCS level < 10" (LT-5)	

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

None

Basis:

RCS water level of 10" as indicated on LT-5 is the lowest level for continued operation of LPI pumps for decay heat removal (ref. 1). Two LPI pumps and two coolers normally perform the decay heat removal function for each unit (ref. 2).

The threshold was chosen because a loss of suction to decay heat removal systems may occur. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS Barrier.

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

For this EAL, a lowering of RCS water level below 10 in. indicates that operator actions have not been successful in restoring and maintaining RCS water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

Although related, this EAL is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Decay Heat Removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

If RCS water level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

ONS Basis Reference(s):

1. AP/1,2,3/A/1700/026 Loss of Decay Heat Removal
2. UFSAR Section 9.3.3 Low Pressure Injection System
3. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory

EAL:

CA1.2 Alert

RCS level **cannot** be monitored for ≥ 15 min. (Note 1)

AND EITHER

- UNPLANNED increase in **any** Table C-1 Sump / Tank level due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-1 Sumps / Tanks
• RB Normal Sumps • RB Emergency Sumps • Core Flood Tank • Quench Tank • Low Activity Waste Tank • High Activity Waste Tank • Miscellaneous Waste Holdup Tank • LPI Room Sumps

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

A RCS leak should be considered UNISOLABLE if the leak cannot be isolated within 15 minutes.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available.

In the Refuel mode, the RCS is not intact and RCS level may be monitored by different means, including the ability to monitor level visually.

In this EAL, all RCS water level indication would be unavailable for greater than 15 minutes, and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of significant leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

For this EAL, the inability to monitor RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

The 15-minute duration for the loss of level indication was chosen because it is half of the EAL duration specified in IC CS1.

If the RCS inventory level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

ONS Basis Reference(s):

1. AP/1,2,3/A/1700/002 Excessive RCS Leakage
2. AP/1-2,3/A/1700/030 Auxiliary Building Flood
3. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory affecting core decay heat removal capability

EAL:

CS1.1 Site Area Emergency

RCS level **cannot** be monitored for ≥ 30 min. (Note 1)

AND

Core uncovery is indicated by **any** of the following:

- UNPLANNED increase in **any** Table C-1 sump/tank level
- Visual observation of UNISOLABLE RCS leakage
- High alarm on RIA-3 RB Refueling Deck Shield Wall
- Erratic Source Range Monitor Indication

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-1 Sumps / Tanks
• RB Normal Sumps
• RB Emergency Sumps
• Core Flood Tank
• Quench Tank
• Low Activity Waste Tank
• High Activity Waste Tank
• Miscellaneous Waste Holdup Tank
• LPI Room Sumps

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

A RCS leak should be considered UNISOLABLE if the leak cannot be isolated within 15 minutes.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available.

In the Refueling mode, the RCS is not intact and RCS level may be monitored by different means, including the ability to monitor level visually.

In this EAL, all RCS water level indication would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of significant leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

Sump or tank level increases should be of a magnitude that correlates to a volume sufficient to indicate fuel has been uncovered or uncovery is imminent.

The Reactor Vessel inventory loss may be detected by a reduction in water shielding that causes a high alarm on the Refueling Deck Shield Wall area radiation monitor (ref. 3).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations (ref. 4, 5, 6).

This IC addresses a significant and prolonged loss of reactor vessel/RCS inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RCS level cannot be restored, fuel damage is probable.

The 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

This EAL addresses concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal; SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues; NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States; and NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management.

Escalation of the emergency classification level would be via IC CG1 or RG1.

ONS Basis Reference(s):

1. AP/1,2,3/A/1700/002 Excessive RCS Leakage
2. AP/1-2,3/A/1700/030 Auxiliary Building Flood
3. UFSAR Table 12-3 Area Radiation Monitors
4. UFSAR Section 7.4.1 Nuclear Instrumentation
5. OP/1,2,3/A/5102/002 Alarm Response Guide 1,2,3SA-02, A-6
6. Nuclear Safety Analysis Center (NSAC), 1980, "Analysis of Three Mile Island - Unit 2 Accident," NSAC-1
7. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory affecting fuel clad integrity with containment challenged

EAL:

CG1.1 General Emergency

RCS level **cannot** be monitored for ≥ 30 min. (Note 1)

AND

Core uncovery is indicated by **any** of the following:

- UNPLANNED increase in **any** Table C-1 sump/tank level due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage
- High alarm on RIA-3 RB Refueling Deck Shield Wall
- Erratic Source Range Monitor Indication

AND

Any Containment Challenge indication, Table C-2

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is **not** required.

Table C-1 Sumps / Tanks
• RB Normal Sumps
• RB Emergency Sumps
• Core Flood Tank
• Quench Tank
• Low Activity Waste Tank
• High Activity Waste Tank
• Miscellaneous Waste Holdup Tank
• LPI Room Sumps

Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • Containment hydrogen concentration ≥ 4% • Unplanned rise in containment pressure

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under all plant conditions up to and including a loss of decay heat removal or fuel handling accident inside containment.

As applied to ONS, Containment Closure is established when the requirements of OP/1,2,3/A/1502/009, Containment Closure Control, are met.

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

A RCS leak should be considered UNISOLABLE if the leak cannot be isolated within 15 minutes.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS level monitoring means are available.

In the Refueling mode, the RCS is not intact and RCS level may be monitored by different means, including the ability to monitor level visually.

In this EAL, all RCS water level indication would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of significant leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

Sump or tank level increases should be of a magnitude that correlates to a volume sufficient to indicate fuel has been uncovered or uncovery is imminent.

The Reactor Vessel inventory loss may be detected by a reduction in water shielding that causes a high alarm on the Refueling Deck Shield Wall area radiation monitor (ref. 3).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations (ref. 4, 5, 6).

Three conditions are associated with a challenge to Containment integrity:

1. CONTAINMENT CLOSURE not established - The status of containment closure is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 7). If containment closure is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to GE would not occur.
2. Containment hydrogen ≥ 4% - The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen combustion. ONS is equipped with a Containment Hydrogen Monitoring System (CHMS) that provides continuous indication of hydrogen concentration in the containment atmosphere. The measurement capability is provided over the range of 0% to 10%. A continuous indication of the hydrogen concentration is not required in the control room at all times during normal operation. If continuous indication of the hydrogen concentration is not available at all times, continuous indication and recording shall be functioning within 90 minutes of the initiation of the safety injection. (ref. 8, 9)
3. UNPLANNED rise in containment pressure - An unplanned pressure rise in containment while in cold shutdown or refueling modes can threaten Containment Closure capability and thus containment potentially cannot be relied upon as a barrier to fission product release.

This IC addresses the inability to restore and maintain reactor vessel level above the top of active fuel with containment challenged. This condition represents actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RCS level cannot be restored, fuel damage is probable.

With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to Containment integrity.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.

The 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor RCS level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

This EAL addresses concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management*.

ONS Basis Reference(s):

1. AP/1,2,3/A/1700/002 Excessive RCS Leakage
2. AP/1-2,3/A/1700/030 Auxiliary Building Flood
3. UFSAR Table 12-3 Area Radiation Monitors
4. UFSAR Section 7.4.1 Nuclear Instrumentation
5. OP/1/A/6101/002; OP/2/A/6102/002; OP/3/A/6103/002 Alarm Response Guide 1,2,3SA-02, A-6
6. Nuclear Safety Analysis Center (NSAC), 1980, "Analysis of Three Mile Island - Unit 2 Accident," NSAC-1
7. OP/1,2,3/A/1502/009 Containment Closure Control
8. UFSAR Section 9.3.7 Containment Hydrogen Monitoring System
9. UFSAR Section 15.16.3 Evaluation of Hydrogen Concentrations
10. NEI 99-01 CG1

ATTACHMENT 1
EAL Bases

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Essential AC Power

Initiating Condition: Loss of all but one AC power source to essential buses for 15 minutes or longer

EAL:

CU2.1 Unusual Event

AC power capability, Table C-3, to essential 4160 V buses MFB-1 and MFB-2 reduced to a single power source for ≥ 15 min. (Note 1)

AND

Any additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-3 AC Power Sources
Offsite: • Unit Normal Transformer (backcharged) • Unit Startup Transformer (SWYD) • Another Unit Startup Transformer (aligned) (SWYD) • CT5 (Central/energizing Standby Bus)
Emergency: • Unit Startup Transformer (Keowee) • Another Unit Startup Transformer (aligned) (Keowee) • CT4 • CT5 (dedicated line/energizing Standby Bus)

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling, NM – No Mode

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

For emergency classification purposes, "capability" means that an AC power source is available to and capable of powering the emergency bus(es) within 15 min, whether or not the buses are currently powered from it.

The condition indicated by this EAL is the degradation of the offsite and onsite power sources such that any additional single failure would result in a loss of all AC power to the emergency buses. 4160 V buses MFB-1 and MFB-2 are the essential buses (ref. 1).

Each unit is provided with two physically independent circuits from the switching station. One is the circuit from the 230 kV switching station through the startup transformer, which is designed to be available within a few seconds following a loss of coolant accident. The second circuit is the path from the switchyard through the main step-up transformer, the main generator bus and the unit auxiliary transformer with the generator disconnected from the main bus. The second circuit is currently used during refueling as an additional power feed for the shutdown unit(s) from the 230 kV switchyard. Whenever there is inadequate power from the generating units, the 230 kV switching station and the hydro units, power is available to the standby power buses either directly from the 100 kV Central Tie Substation or from Lee Steam Station via a Transformer CT5 (ref. 2).

Upon loss of power from the Oconee generating unit and 230 kV switchyard, power is supplied from both Keowee Hydro Station units through two separate and independent routes. One route is an underground feeder to Transformer CT4 which supplies the two redundant Main Feeder Buses (MFB-1 and MFB-2). The other route is an overhead feeder to the 230 KV switching station which supplies each unit startup transformer (ref. 3).

The Standby Shutdown Facility (SSF) consists of standby systems for use in an extreme emergency and is equipped with a manually started diesel generator that can supply power necessary to maintain hot shutdown of the reactors of each unit in the event of loss of power. However, the SSF cannot supply power to the essential buses and therefore not credited in this EAL (ref. 3).

This cold condition EAL is equivalent to the hot condition EAL SA1.1.

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment.

When in the cold shutdown, refueling, or no mode, this condition is not classified as an Alert because of the increased time available to restore another power source to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

An "AC power source" is a source recognized in APs and EOPs, and capable of supplying required power to an essential bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one essential power source (e.g., CT4, CT5, CT1 (Keowee).

- A loss of essential power sources (e.g., CT4, CT5, CT1, 2, 3 (Keowee)) with a single train of essential buses being back-fed from an offsite power source.

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

The subsequent loss of the remaining single power source would escalate the event to an Alert in accordance with IC CA2.

ONS Basis Reference(s):

1. UFSAR Figure 8.1 Single Line Diagram
2. UFSAR Section 8.2 Offsite Power System
3. UFSAR Section 8.3 Onsite Power Systems
4. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Essential AC Power

Initiating Condition: Loss of **all** offsite and **all** emergency AC power to essential buses for 15 minutes or longer

EAL:

CA2.1 **Alert**
Loss of **all** offsite and **all** emergency AC power capability to essential 4160 V buses MFB-1 and MFB-2 for ≥ 15 min. (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, NM – No Mode

Basis:

For emergency classification purposes, "capability" means that an AC power source is available to and capable of powering the emergency bus(es) within 15 min, whether or not the buses are currently powered from it.

The condition indicated by this EAL is the degradation of the offsite and onsite power sources resulting in a loss of all AC power to the emergency buses. 4160 V buses MFB-1 and MFB-2 are the essential buses (ref. 1).

Each unit is provided with two physically independent circuits from the switching station. One is the circuit from the 230 kV switching station through the startup transformer, which is designed to be available within a few seconds following a loss of coolant accident. The second circuit is the path from the switchyard through the main step-up transformer, the main generator bus and the unit auxiliary transformer with the generator disconnected from the main bus. The second circuit is currently used during refueling as an additional power feed for the shutdown unit(s) from the 230 kV switchyard. Whenever there is inadequate power from the generating units, the 230 kV switching station and the hydro units, power is available to the standby power buses either directly from the 100 kV Central Tie Substation or from Lee Steam Station via a Transformer CT5. (ref. 2)

Upon loss of power from the Oconee generating unit and 230 kV switchyard, power is supplied from both Keowee Hydro Station units through two separate and independent routes. One route is an underground feeder to Transformer CT4 which supplies the two redundant Main Feeder Buses (MFB-1 and MFB-2). The other route is an overhead feeder to the 230 KV switching station which supplies each unit startup transformer. (ref. 3)

The Standby Shutdown Facility (SSF) consists of standby systems for use in an extreme emergency and is equipped with a manually started diesel generator that can supply power necessary to maintain hot shutdown of the reactors of each unit in the event of loss of power. However, the SSF cannot supply power to the essential buses and therefore not credited in this EAL (ref. 3).

This cold condition EAL is equivalent to the hot condition loss of all offsite AC power EAL SS1.1.

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink.

When in the cold shutdown, refueling, or no mode, this condition is not classified as a Site Area Emergency because of the increased time available to restore an emergency bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition represents an actual or potential substantial degradation of the level of safety of the plant.

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

Escalation of the emergency classification level would be via IC CS1 or RS1.

ONS Basis Reference(s):

1. UFSAR Figure 8.1 Single Line Diagram
2. UFSAR Section 8.2 Offsite Power System
3. UFSAR Section 8.3 Onsite Power Systems
4. NEI 99-01 CA2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.1 **Unusual Event**
UNPLANNED increase in RCS temperature to > 200°F due to loss of decay heat removal capability (Note 10)

Note 10: In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on time to boil data.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1). These include cold leg (T_c) temperature indications, hot leg (T_h) temperature indications with RCPs running, CETCs and LPI cooler outlet temperature indications (ref. 2).

However, if Low Pressure Injection (LPI) flow is lost, the normal temperature elements used to monitor RCS temperature are not accurate indicators of RCS temperature. The CETCs are the design instruments for these conditions. For some periods of time the CETCs may not be available. The current practices concerning determining time to boil can be used in the evaluation of these EALs. Without CETC indication and with a loss of LPI flow the following guidance should be used (ref. 2):

- Use the predetermined "time to boil" data for evaluating these EALs. This approach reflects the relatively small numerical difference between the typical Technical Specification cold shutdown temperature limit of 200°F and the boiling temperature of RCS water with the plant in Mode 5 or 6.
- Alternately, the Control Room staff may use a procedure or user aid to determine when RCS temperature will likely exceed 200°F given the actual plant conditions (e.g., using a heat-up curve).

This IC addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit and represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Coordinator should also refer to IC CA3.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

This EAL involves a loss of decay heat removal capability, or an addition of heat to the RCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

During an outage, the level in the reactor vessel will normally be maintained at or above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid increase in reactor coolant temperature depending on the time after shutdown.

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

ONS Basis Reference(s):

1. ONS Technical Specifications Table 1.1-1
2. AP/1,2,3/A/1700/026 Loss of Decay Heat Removal
3. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.2	**Unusual Event**
Loss of **all** RCS temperature and RCS level indication for ≥ 15 min. (Note 1)	

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6- Refueling

Definition(s):

None

Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1). These include cold leg (T_c) temperature indications, hot leg (T_h) temperature indications with RCPs running, CETCs and LPI cooler outlet temperature indications (ref. 2).

Several instruments are capable of providing indication of RCS level including pressurizer level, RVLIS, LT-5 and local monitor (ref. 3).

This EAL addresses the inability to determine RCS temperature and level, and represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Coordinator should also refer to IC CA3.

This EAL reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

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

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

ONS Basis Reference(s):

1. ONS Technical Specifications Table 1.1-1
2. AP/1,2,3/A/1700/026 Loss of Decay Heat Removal
3. UFSAR Section 7.5.2.2 Inadequate Core Cooling Instruments
4. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.1 Alert

UNPLANNED increase in RCS temperature to > 200°F for > Table C-4 duration (Notes 1, 10)

OR

UNPLANNED RCS pressure increase > 10 psig due to a loss of RCS cooling (this EAL does not apply during water-solid plant conditions)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

Note 10: In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on time to boil data.

Table C-4: RCS Heat-up Duration Thresholds		
RCS Status	**CONTAINMENT CLOSURE Status**	**Heat-up Duration**
Intact (but **not** REDUCED INVENTORY)	N/A	60 min.*
Not intact **OR** REDUCED INVENTORY	established	20 min.*
	not established	0 min.
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable.		

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The action to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under all plant conditions up to and including a loss of decay heat removal or fuel handling accident inside containment.

As applied to ONS, Containment Closure is established when the requirements of OP/1,2,3/A/1502/009, Containment Closure Control, are met.

UNPLANNED -. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

REDUCED INVENTORY - Condition with fuel in the reactor vessel and the level lower than three feet below the reactor vessel flange (RCS level < 50" on LT-5)

Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1). These include cold leg (T_c) temperature indications, hot leg (T_h) temperature indications with RCPs running, CETCs and LPI cooler outlet temperature indications (ref. 2).

However, if Low Pressure Injection (LPI) flow is lost, the normal temperature elements used to monitor RCS temperature are not accurate indicators of RCS temperature. The CETCs are the design instruments for these conditions. For some periods of time the CETCs may not be available. The current practices concerning determining time to boil can be used in the evaluation of these EALs. Without CETC indication and with a loss of LPI flow the following guidance should be used (ref. 2):

- Use the predetermined "time to boil" data for evaluating these EALs. This approach reflects the relatively small numerical difference between the typical Technical Specification cold shutdown temperature limit of 200°F and the boiling temperature of RCS water with the plant in Mode 5 or 6.
- Alternately, the Control Room staff may use a procedure or user aid to determine when RCS temperature will likely exceed 200°F given the actual plant conditions (e.g., using a heat-up curve).

Numerous RCS pressure instruments are capable of measuring pressure to less than 10 psia including RCS low range cooldown pressure indicators RC-P-0086A/B (ref. 3).

This IC addresses conditions involving a loss of decay heat removal capability or an addition of heat to the RCS in excess of that which can currently be removed. Either condition represents an actual or potential substantial degradation of the level of safety of the plant.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

The RCS Heat-up Duration Thresholds table addresses an increase in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not intact, or RCS inventory is reduced (e.g., mid-loop operation). The 20-minute criterion was included to allow time for operator action to address the temperature increase.

The RCS Heat-up Duration Thresholds table also addresses an increase in RCS temperature with the RCS intact. The status of CONTAINMENT CLOSURE is not crucial in this condition since the intact RCS is providing a high pressure barrier to a fission product release. The 60-minute time frame should allow sufficient time to address the temperature increase without a substantial degradation in plant safety.

Finally, in the case where there is an increase in RCS temperature, the RCS is not intact or is at reduced inventory, and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because 1) the evaporated reactor coolant may be released directly into the containment atmosphere and subsequently to the environment, and 2) there is reduced reactor coolant inventory above the top of irradiated fuel.

The RCS pressure increase threshold provides a pressure-based indication of RCS heat-up in the absence of RCS temperature monitoring capability.

Escalation of the emergency classification level would be via IC CS1 or RS1.

ONS Basis Reference(s):

1. ONS Technical Specifications Table 1.1-1
2. AP/1,2,3/A/1700/026 Loss of Decay Heat Removal
3. IP/1,2,3/A/0200/047A Reactor Coolant System LTOP Instrument Calibration
4. NEI 99-01 CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – Loss of Vital DC Power

Initiating Condition: Loss of Vital DC power for 15 minutes or longer

EAL:

CU4.1	**Unusual Event**
Indicated voltage is < 105VDC on vital DC buses **required** by Technical Specifications for ≥ 15 min. (Note 1)	

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

None

Basis:

The purpose of this EAL 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. The fifteen minute interval is intended to exclude transient or momentary power losses.

For each unit, two independent and physically separated 125 volt DC batteries and DC buses are provided for the vital instrumentation and control power system. (ref. 1, 2). Minimum DC bus voltage is 110 VDC (ref. 3).

This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS2.1.

This IC addresses a loss of vital DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or refueling mode. In these modes, the core decay heat load has been significantly reduced, and coolant system temperatures and pressures are lower; these conditions increase the time available to restore a vital DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, "required" means the vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if Train A is out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an Unusual Event. A loss of Vital DC power to Train A would not warrant an emergency classification.

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

Depending upon the event, escalation of the emergency classification level would be via IC CA1 or CA3, or an IC in Recognition Category R.

ONS Basis Reference(s):

1. UFSAR Figure 8.5 Typical DC and AC Vital Power System - Single Line
2. UFSAR Section 8.3.2 DC Power Systems
3. EP/*/A/1800/001 Blackout Tab
4. Technical Specifications 3.8.4 DC Sources - Shutdown
5. NEI 99-01 CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

CU5.1 Unusual Event

Loss of **all** Table C-5 onsite communication methods

OR

Loss of **all** Table C-5 offsite communication methods

OR

Loss of **all** Table C-5 NRC communication methods

Table C-5 Communication Methods

System	Onsite	Offsite	NRC
Commercial phone service	X	X	X
ONS site phone system	X	X	X
EOF phone system		X	X
Public Address system	X		
Onsite radio system	X		
DEMNET		X	
Offsite radio system		X	
NRC Emergency Telephone System			X
Satellite Phone		X	

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, NM – No Mode

Definition(s):

None

Basis:

Onsite, offsite and NRC communications include one or more of the systems listed in Table C-5 (ref. 1).

1. Commercial phone service

   The Commercial phone service does not go through the site telephone system.

2. ONS site phone system

   The site phone system is generator and battery backed with:

   - Fiber-Optic to Charlotte GO (65 lines)
   - Telephone line to Easley (6 circuits)
   - Anderson (4 lines)
   - Six Mile (4 lines)
   - Site Telephone System: Inward and outward direct dial available from the Control Room, TSC, and OSC

3. EOF phone system

   The emergency communications systems at the Charlotte EOF are designed to ensure the reliable, timely flow of information between all parties having an emergency response role.

4. Public Address (Paging) system

   The paging system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

5. Onsite radio system

   The onsite radio system receives emergency backup power from Keowee Hydro Units supporting communications with: Control Room 1&2, 3, Fire Brigade, Chemistry, Safety, Radiation Protection, Maintenance, Medical Emergency Response Team, and Hazardous Materials Response Team.

6. DEMNET

   DEMNET is the primary means of offsite communication. This circuit allows intercommunication among the EOF, TSC, control room, counties, and states. DEMNET operates as an internet based (VoIP) communications system with a satellite back-up. Should the internet transfer rate become slow or unavailable, the DEMNET will automatically transfer to satellite mode.

7. Offsite radio system

   The offsite radio system is battery backed supporting communications with: Control Room Units 1&2, TSC, Field Monitoring Teams, EOF, counties and State of South Carolina.

8. NRC Emergency Telephone System (ETS)

   The NRC uses a Duke Energy dedicated telephone line which allows direct telephone communications from the plant to NRC regional and national offices. The Duke Energy communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the Oconee Control Rooms, Technical Support Center, and Emergency Operations Facility and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

9. Satellite Phone

Satellite Phones can be used for external communications

This EAL is the cold condition equivalent of the hot condition EAL SU7.1.

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the State EOC and FEO, Pickens County LEC and EOC, and Oconee County LEC and EOC.

The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

ONS Basis Reference(s):

1. ONS Emergency Plan, Section 7.2 Communications Systems
2. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting Safety Systems

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

EAL:

CA6.1 Alert

The occurrence of **any** Table C-6 hazardous event

AND EITHER:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Table C-6 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

VISIBLE DAMAGE - Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

Basis:

- The significance of seismic events are discussed under EAL HU2.1 (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps.
- External flooding at ONS is highly unlikely since the probable maximum flood (PMF) would be contained by the Keowee Reservoir. Plant grade elevation is 796.0 ft MSL. The minimum external access elevation for the Auxiliary, Turbine, and Service Buildings is 796.5 ft MSL which provides a 6 inch water sill. (ref. 2)
- High winds in excess of design (95 mph) or tornado strikes can cause significant structural damage (ref. 3).
- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area (ref. 4, 5).
- An explosion that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

The first conditional addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

The second conditional addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC CS1 or RS1.

ONS Basis Reference(s):

1. AP/0/A/1700/005 Earthquake
2. UFSAR Section 3.4.1.1 Flood Protection Measures for Seismic Class 1 Structures
3. UFSAR Section 3.3.1.1 Design Wind Velocity
4. OSS-0254.00-00-4008 Design Bases Specification for Fire Protection
5. AP/1,2,3/A/1700/050 Challenging Plant Fire
6. NEI 99-01 CA6

Category H – Hazards and Other Conditions Affecting Plant Safety

EAL Group: ANY (EALs in this category are applicable to any plant condition, hot or cold.)

Hazards are non-plant, system-related events that can directly or indirectly affect plant operation, reactor plant safety or personnel safety.

1. Security

Unauthorized entry attempts into the PROTECTED AREA, bomb threats, sabotage attempts, and actual security compromises threatening loss of physical control of the plant.

2. Seismic Event

Natural events such as earthquakes have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety.

3. Natural or Technology Hazard

Other natural and non-naturally occurring events that can cause damage to plant facilities include tornados, FLOODING, hazardous material releases and events restricting site access warranting classification.

4. Fire

Fires can pose significant hazards to personnel and reactor safety. Appropriate for classification are fires within the Plant PROTECTED AREA or which may affect operability of equipment needed for safe shutdown

5. Hazardous Gas

Toxic, corrosive, asphyxiant or flammable gas leaks can affect normal plant operations or preclude access to plant areas required to safely shutdown the plant.

6. Control Room Evacuation

Events that are indicative of loss of Control Room habitability. If the Control Room must be evacuated, additional support for monitoring and controlling plant functions is necessary through the emergency response facilities.

7. Emergency Coordinator Judgment

The EALs defined in other categories specify the predetermined symptoms or events that are indicative of emergency or potential emergency conditions and thus warrant classification. While these EALs have been developed to address the full spectrum of possible emergency conditions which may warrant classification and subsequent implementation of the Emergency Plan, a provision for classification of emergencies based on operator/management experience and judgment is still necessary. The EALs of this category provide the Emergency Coordinator the latitude to classify emergency conditions consistent with the established classification criteria based upon Emergency Coordinator judgment.

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1	**Unusual Event**
A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervision	

Mode Applicability:

All

Definition(s):

SECURITY CONDITION - Any security event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A security condition does not involve a hostile action.

HOSTILE ACTION - An act toward ONS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on ONS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

Basis:

This EAL is based on the Duke Energy Physical Security Plan for ONS (ref. 1).

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program].*

This EAL references the Security Shift Supervision because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR §

2.39 information.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Duke Energy Physical Security Plan for ONS (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

ONS Basis Reference(s):

1. Duke Energy Physical Security Plan for ONS
2. AP/0/A/1700/045 Site Security Threats
3. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.2	**Unusual Event**
Notification of a credible security threat directed at the site	

Mode Applicability:

All

Definition(s):

SECURITY CONDITION - Any security event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A security condition does not involve a hostile action.

Basis:

This EAL is based on the Duke Energy Physical Security Plan for ONS (ref. 1).

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program].*

This EAL addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the Duke Energy Physical Security Plan for ONS.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Duke Energy Physical Security Plan for ONS (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

ONS Basis Reference(s):

1. Duke Energy Physical Security Plan for ONS
2. AP/0/A/1700/045 Site Security Threats
3. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.3	**Unusual Event**
A validated notification from the NRC providing information of an aircraft threat	

Mode Applicability:

All

Definition(s):

SECURITY CONDITION - Any security event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A security condition does not involve a hostile action.

Basis:

This EAL is based on the Duke Energy Physical Security Plan for ONS (ref. 1).

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program].*

This EAL addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with AP/0/A/1700/045 Site Security Threats (ref. 2).

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Duke Energy Physical Security Plan for ONS (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

ONS Basis Reference(s):

1. Duke Energy Physical Security Plan for ONS
2. AP/0/A/1700/045 Site Security Threats
3. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

EAL:

HA1.1	**Alert**
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervision	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward ONS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on ONS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

OWNER CONTROLLED AREA - Area outside the PROTECTED AREA fence that immediately surrounds the plant. Access to this area is generally restricted to those entering on official business.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of Offsite Response Organizations (OROs), allowing them to be better prepared should it be necessary to consider further actions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

This EAL is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes any action directed against an ISFSI that is located outside the plant PROTECTED AREA.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Duke Energy Physical Security Plan for ONS (ref. 1).

Escalation of the emergency classification level would be via IC HS1.

ONS Basis Reference(s):

1. Duke Energy Physical Security Plan for ONS
2. AP/0/A/1700/045 Site Security Threats
3. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

EAL:

HA1.2	**Alert**
A validated notification from NRC of an aircraft attack threat within 30 min. of the site	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward ONS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on ONS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program].*

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of Offsite Response Organizations (OROs), allowing them to be better prepared should it be necessary to consider further actions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

This EAL addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and OROs are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with AP/0/A/1700/045 Site Security Threats (ref. 2).

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may be provided by NORAD through the NRC.

In some cases, it may not be readily apparent if an aircraft impact within the OWNER CONTROLLED AREA was intentional (i.e., a HOSTILE ACTION). It is expected, although not certain, that notification by an appropriate Federal agency to the site would clarify this point. In this case, the appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. The emergency declaration, including one based on other ICs/EALs, should not be unduly delayed while awaiting notification by a Federal agency.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Duke Energy Physical Security Plan for ONS (ref. 1).

Escalation of the emergency classification level would be via IC HS1.

ONS Basis Reference(s):

1. Duke Energy Physical Security Plan for ONS
2. AP/0/A/1700/045 Site Security Threats
3. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA

EAL:

HS1.1	**Site Area Emergency**
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward ONS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on ONS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

PROTECTED AREA - That part of the nuclear station property consisting of the Reactor, Auxiliary, Turbine, and Service Building and grounds, contained within the owner controlled security fence.

Basis:

The Security Shift Supervision are the designated on-site personnel 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 Duke Energy Physical Security Contingency Plan for ONS (Safeguards) information. (ref. 1)

This IC addresses the occurrence of a HOSTILE ACTION within the PROTECTED AREA. This event will require rapid response and assistance due to the possibility for damage to plant equipment.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1, 2).

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program].*

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Site Area Emergency declaration will mobilize Offsite Response Organization (ORO) resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

This IC does not apply to a HOSTILE ACTION directed at an ISFSI PROTECTED AREA located outside the PROTECTED AREA; such an attack should be assessed using IC HA1. It also does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Duke Energy Physical Security Plan for ONS (ref. 1).

Escalation of the emergency classification level would be via IC HG1.

ONS Basis Reference(s):

1. Duke Energy Physical Security Plan for ONS
2. AP/0/A/1700/045 Site Security Threats
3. NEI 99-01 HS1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility

EAL:

HG1.1 General Emergency

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

AND EITHER of the following has occurred:

Any of the following safety functions cannot be controlled or maintained

- Reactivity
- Core cooling
- RCS heat removal

OR

Damage to spent fuel has occurred or is IMMINENT

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward ONS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on ONS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

IMMINENT - The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions

PROTECTED AREA - That part of the nuclear station property consisting of the Reactor, Auxiliary, Turbine, and Service Building and grounds, contained within the owner controlled security fence.

Basis:

Indications of damaged spent fuel are provided in AP/1,2,3/A/1700/009 Spent Fuel Damage (ref. 4).

This IC addresses an event in which a HOSTILE FORCE has taken physical control of the facility to the extent that the plant staff can no longer operate equipment necessary to maintain key safety functions. It also addresses a HOSTILE ACTION leading to a loss of physical control that results in actual or IMMINENT damage to spent fuel due to 1) damage to a spent fuel pool cooling system (e.g., pumps, heat exchangers, controls, etc.) or, 2) loss of spent fuel pool integrity such that sufficient water level cannot be maintained.

Timely and accurate communications between the Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3).

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Duke Energy Physical Security Plan for ONS (ref.1).

ONS Basis Reference(s):

1. Duke Energy Physical Security Plan for ONS
2. AP/0/A/1700/045 Site Security Threats
3. AP/0/A/1700/046 Extensive Damage Mitigation
4. AP/1,2,3/A/1700/009 Spent Fuel Damage
5. NEI 99-01 HG1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE levels

EAL:

HU2.1 Unusual Event

Seismic event > OBE as indicated by **EITHER** of the following:

- 1SA-9/E-1 (SEISMIC TRIGGER) alarm
- 3SA-9/E-1 (SEISMIC TRIGGER) alarm

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL is based on a VALID receipt of either of the specified seismic trigger alarms.

For ONS, the Operating Basis Earthquake (OBE) is equivalent to the Design Basis Earthquake (DBE). The design basis earthquake ground acceleration at the site is 0.05g. The maximum hypothetical earthquake ground acceleration is 0.10g and 0.15g for Class 1 structures founded on bedrock and overburden respectively. (ref. 1)

If an earthquake of ≥ 0.05 g has occurred on site, all units are required to be shut down to Mode 5 once a plant damage assessment is complete along with the completion of any needed repairs to support the units ability to achieve safe shutdown. (ref. 2)

Earthquake instrumentation is the SMA-3 system consisting of a central recording system, control panel, one TS-3 triaxial seismic trigger package, and two force-balance triaxial accelerometer packages. The seismic trigger and one accelerometer of the SMA-3 system are located in the Unit 1 Tendon Gallery. Also, a second accelerometer is located directly above at elevation 797' +6" in the Oconee 1 Reactor Building. The recorder for the system is located in the Unit 1 Cable Room. Also, a seismic trigger/switch is located in the Unit 1 tendon gallery. The TS-3 has a preset acceleration threshold of 0.05g which activates the statalarm in Units 1 and 3 control rooms, when design conditions occur. (ref. 3)

To avoid inappropriate emergency classification resulting from spurious actuation of the seismic instrumentation or felt motion not attributable to seismic activity, an offsite agency (USGS, National Earthquake Information Center (NEIC)) can confirm that an earthquake has occurred in the area of the plant. Such confirmation should not, however, preclude a timely emergency declaration based on receipt of the OBE alarm. The NEIC can be contacted by calling **(303) 273-8500** (ref. 2). Select **option #1** and inform the analyst you wish to confirm recent seismic activity in the vicinity of ONS. If requested, provide the analyst with the following ONS coordinates: **34° 47' 38.2" north latitude, 82° 53' 55.4" west longitude** (ref. 4). Alternatively, near real-time seismic activity can be accessed via the NEIC website:

http://earthquake.usgs.gov/eqcenter/

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE). An earthquake greater than an OBE but less than a Safe Shutdown Earthquake (SSE) should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., performs walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event (e.g., lateral accelerations in excess of 0.05g). The Shift Manager or Emergency Coordinator may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

ONS Basis Reference(s):

1. UFSAR Section 3.2.1.3 Seismic Loading Conditions
2. AP/0/A/1700/005 Earthquake
3. UFSAR Section 3.7.4 Seismic Instrumentation Program
4. UFSAR Section 2.1.1.1 Specification of Location
5. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.1	**Unusual Event**
A tornado strike within the PROTECTED AREA	

Mode Applicability:

All

Definition(s):

PROTECTED AREA - That part of the nuclear station property consisting of the Reactor, Auxiliary, Turbine, and Service Building and grounds, contained within the owner controlled security fence.

Basis:

Response actions associated with a tornado onsite is provided in AP/0/A/1700/006, Natural Disaster (ref. 1).

If damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under EAL CA6.1 or SA9.1.

A tornado striking (touching down) within the PROTECTED AREA warrants declaration of an Unusual Event regardless of the measured wind speed at the meteorological tower. A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

EAL HU3.1 addresses a tornado striking (touching down) within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

ONS Basis Reference(s):

1. AP/0/A/1700/006 Natural Disaster
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.2	**Unusual Event**
Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode	

Mode Applicability:

All

Definition(s):

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

Areas susceptible to internal flooding are the Turbine Building and Auxiliary Building (ref.1, 2). Refer to EAL CA6.1 or SA9.1 for internal flooding affecting one or more SAFETY SYSTEM trains.

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses FLOODING of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

ONS Basis Reference(s):

1. AP/1,2,3/A/1700/010 Turbine Building Flood
2. AP/1-2,3/A/1700/030 Auxiliary Building Flood
3. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.3	**Unusual Event**
Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)	

Mode Applicability:

All

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

PROTECTED AREA - That part of the nuclear station property consisting of the Reactor, Auxiliary, Turbine, and Service Building and grounds, contained within the owner controlled security fence.

Basis:

As used here, the term "offsite" is meant to be areas external to the ONS PROTECTED AREA.

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

ONS Basis Reference(s):

1. NEI 99-01 HU3

ATTACHMENT 1
EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.4 Unusual Event

A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)

Note 7: This EAL does **not** apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Mode Applicability:

All

Definition(s):

None

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site FLOODING caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.

This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

ONS Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.1 Unusual Event

A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):

- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

The FIRE is located within **any** Table H-1 area

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Reactor Building • Auxiliary Building • Turbine Building • Standby Shutdown Facility • Intake Structure • Electrical Blockhouse • Keowee Hydro & associated transformers • Transformer Yard • Protected Service Water Building • Essential Siphon Vacuum Building

Mode Applicability:

All

Definition(s):

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

Basis:

The 15 minute requirement begins with a credible notification that a fire is occurring, or receipt of multiple valid fire detection system alarms or field validation of a single fire alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field.

Table H-1 Fire Areas are those areas that contain equipment necessary for safe operation and shutdown of the plant (ref. 1, 2).

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

For EAL HU4.1 the intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

ONS Basis Reference(s):

1. OSS-0254.00-00-4008 Design Bases Specification for Fire Protection
2. AP/1,2,3/A/1700/050 Challenging Plant Fire
3. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.2 Unusual Event

Receipt of a single fire alarm (i.e., **no** other indications of a FIRE)

AND

The fire alarm is indicating a FIRE within **any** Table H-1 area

AND

The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Reactor Building • Auxiliary Building • Turbine Building • Standby Shutdown Facility • Intake Structure • Electrical Blockhouse • Keowee Hydro & associated transformers • Transformer Yard • Protected Service Water Building • Essential Siphon Vacuum Building

Mode Applicability:

All

Definition(s):

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

Basis:

The 30 minute requirement begins upon receipt of a single valid fire detection system alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field. Actual field reports must be made within the 30 minute time limit or a classification must be made. If a fire is verified to be occurring by field report, classification shall be made based on EAL HU4.1.

Control Room indications that may be used to validate a single fire alarm include (ref. 3):

- Remote camera system
- CRD service structure air temperature
- PZR tailpipe temperature
- RB dome temperature
- RBCU inlet and outlet temperatures
- RCP parameters
- Status lights of components located inside RB

Table H-1 Fire Areas are those areas that contain equipment necessary for safe operation and shutdown of the plant (ref. 1, 2).

The ONS Fire Protection Program is based on 10 CFR 50.48 (a) and (c) requiring compliance with NFPA 805. The NFPA 805 based Fire Protection Program requirements provide are consistent with the NEI 99-01 basis stated below (ref. 1, 4).

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

ONS Basis Reference(s):

1. OSS-0254.00-00-4008 Design Bases Specification for Fire Protection
2. AP/1,2,3/A/1700/050 Challenging Plant Fire
3. OP/1,2,3/A/6101/003
4. NRC Letter to T. Preston Gillespie (Duke); ONS Units 1, 2, and 3, Issuance of Amendments Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance With 10 CFR 50.48(c); dated December 29, 2010
5. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.3	**Unusual Event**
A FIRE within the PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)	

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

All

Definition(s):

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

PROTECTED AREA - That part of the nuclear station property consisting of the Reactor, Auxiliary, Turbine, and Service Building and grounds, contained within the owner controlled security fence.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

In addition to a FIRE addressed by EAL HU4.1 or HU4.2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

ONS Basis Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.4	**Unusual Event**
A FIRE within the PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish	

Mode Applicability:

All

Definition(s):

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

PROTECTED AREA - That part of the nuclear station property consisting of the Reactor, Auxiliary, Turbine, and Service Building and grounds, contained within the owner controlled security fence.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

If a FIRE within the PLANT PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA9.

ONS Basis Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gases

Initiating Condition: Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

HA5.1 Alert

Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 rooms or areas

AND

Entry into the room or area is prohibited or IMPEDED (Note 5)

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then **no** emergency classification is warranted.

Table H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Turbine Building	1, 2, 3
Equipment and Cable Rooms	1, 2, 3
Auxiliary Building	1, 2, 3, 4, 5
Reactor Buildings	3, 4, 5

Mode Applicability:

All

Definition(s):

IMPEDE(D) - Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

Basis:

If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

The list of plant rooms or areas with entry-related mode applicability identified specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

This IC addresses an event involving a release of a hazardous gas that precludes or impedes access to equipment necessary to maintain normal plant operation, or required for a normal plant cooldown and shutdown. This condition represents an actual or potential substantial degradation of the level of safety of the plant.

An Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.

Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the Emergency Coordinator's judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

An emergency declaration is not warranted if any of the following conditions apply:

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.
- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

This EAL does not apply to firefighting activities that automatically or manually activate a fire suppression system in an area..

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

ONS Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Rooms/Areas Tables R-3 & H-2 Bases
2. NEI 99-01 HA5

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations

EAL:

HA6.1 **Alert**
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel or Standby Shutdown Facility

Mode Applicability:

All

Definition(s):

None

Basis:

The Control Room Supervisor (CRS) determines if the Control Room is uninhabitable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions.

AP/1,2,3/A/1700/008, Loss of Control Room, provides the instructions for tripping the unit, and maintaining RCS inventory and Hot Shutdown conditions from outside the Control Room due to reasons other than fire (ref. 1).

AP/1,2,3/A/1700/050, Challenging Plant Fire, provides the instructions for tripping the unit, and maintaining RCS inventory and Hot Shutdown conditions from outside the Control Room due to a fire (ref. 2).

If normal post-trip conditions cannot be maintained with the Auxiliary Shutdown Panel or there is a challenging fire in an SSF risk area, plant shutdown may be directed from the Standby Shutdown Facility (ref. 3, 4).

Inability to establish plant control from outside the Control Room escalates this event to a Site Area Emergency per EAL HS6.1.

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Escalation of the emergency classification level would be via IC HS6.

ONS Basis Reference(s):

1. AP/1,2,3/A/1700/008 Loss of Control Room
2. AP/1,2,3/A/1700/050 Challenging Plant Fire
3. AP/0/A/1700/025 Standby Shutdown Facility Emergency Operating Procedure
4. AP/0/A/1700/043 Fire Brigade Response Procedure
5. NEI 99-01 HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

Initiating Condition: Inability to control a key safety function from outside the Control Room

EAL:

HS6.1 Site Area Emergency

An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel or Standby Shutdown Facility

AND

Control of **any** of the following key safety functions is **not** re-established within 15 min. (Note 1):

- Reactivity (Modes 1, 2 and 3 **only**)
- Core cooling
- RCS heat removal

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Hot Shutdown, 5 – Cold Shutdown, 6 - Refuel

Definition(s):

None

Basis:

The Control Room Supervisor (CRS) determines if the Control Room is uninhabitable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions.

AP/1,2,3/A/1700/008, Loss of Control Room, provides the instructions for tripping the unit, and maintaining RCS inventory and Hot Shutdown conditions from outside the Control Room due to reasons other than fire (ref. 1).

AP/1,2,3/A/1700/050, Challenging Plant Fire, provides the instructions for tripping the unit, and maintaining RCS inventory and Hot Shutdown conditions from outside the Control Room due to a fire (ref. 2).

If normal post-trip conditions cannot be maintained with the Auxiliary Shutdown Panel or there is a challenging fire in an SSF risk area, plant shutdown may be directed from the Standby Shutdown Facility (ref. 3, 4).

The intent of this EAL is to capture events in which control of the plant cannot be reestablished in a timely manner. The fifteen minute time for transfer starts when the Control Room is evacuated (when CRS leaves the Control Room, not when AP/1,2,3/A/1700/008 or AP/1,2,3/A/1700/050 is entered). The time interval is based on how quickly control must be reestablished without core uncovery and/or core damage. The determination of whether or not control is established from outside the Control Room is based on Emergency Coordinator judgment. The Emergency Coordinator is expected to make a reasonable, informed judgment that control of the plant from outside the Control Room cannot be established within the fifteen minute interval.

Once the Control Room is evacuated, the objective is to establish control of important plant equipment and maintain knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to shutdown the reactor and maintain it shutdown), RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink).

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on Emergency Coordinator judgment. The Emergency Coordinator is expected to make a reasonable, informed judgment within 15 minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s).

Escalation of the emergency classification level would be via IC FG1 or CG1

ONS Basis Reference(s):

1. AP/1,2,3/A/1700/008 Loss of Control Room
2. AP/1,2,3/A/1700/050 Challenging Plant Fire
3. AP/0/A/1700/025 Standby Shutdown Facility Emergency Operating Procedure
4. AP/0/A/1700/043 Fire Brigade Response Procedure
5. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a UE

EAL:

HU7.1	**Unusual Event**
Other conditions exist which in the judgment of the Emergency Coordinator 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 offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.	

Mode Applicability:

All

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the ONS Emergency Plan (ref. 1). The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but plant management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

This IC addresses 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 emergency classification level description for an Unusual Event.

ONS Basis Reference(s):

1. ONS Emergency Plan Section A Assignment of Responsibility
2. RP/0/A/1000/001 Emergency Classification
3. NEI 99-01 HU7

ATTACHMENT 1
EAL Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions exist that in the judgment of the Emergency Coordinator warrant declaration of an Alert

EAL:

HA7.1	**Alert**
Other conditions exist which, in the judgment of the Emergency Coordinator, 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.	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward ONS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on ONS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

Basis:

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the ONS Emergency Plan (ref. 1). The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

This IC addresses 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 emergency classification level description for an Alert.

ONS Basis Reference(s):

1. ONS Emergency Plan Section A Assignment of Responsibility
2. RP/0/A/1000/001 Emergency Classification
3. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a Site Area Emergency

EAL:

HS7.1 **Site Area Emergency**
Other conditions exist which in the judgment of the Emergency Coordinator 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

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward ONS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on ONS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

SITE BOUNDARY - That area, including the PROTECTED AREA, in which Duke Energy has the authority to control all activities including exclusion or removal of personnel and property (1 mile radius) from the center of Unit 2.

Basis:

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the ONS Emergency Plan (ref. 1). The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

This IC addresses 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 emergency classification level description for a Site Area Emergency.

ONS Basis Reference(s):

1. ONS Emergency Plan Section A Assignment of Responsibility
2. RP/0/A/1000/001 Emergency Classification
3. NEI 99-01 HS7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Coordinator warrant declaration of a General Emergency

EAL:

HG7.1	**General Emergency**
Other conditions exist which in the judgment of the Emergency Coordinator 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 offsite for more than the immediate site area.	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward ONS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on ONS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

IMMINENT - The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

Basis:

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the ONS Emergency Plan (ref. 1). The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

Releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the Site Boundary.

This IC addresses 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 emergency classification level description for a General Emergency.

ONS Basis Reference(s):

1. ONS Emergency Plan Section A Assignment of Responsibility
2. RP/0/A/1000/001 Emergency Classification
3. NEI 99-01 HG7

Category S – System Malfunction

EAL Group: Hot Conditions (RCS temperature > 210°F); EALs in this category are applicable only in one or more hot operating modes.

Numerous system-related equipment failure events that warrant emergency classification have been identified in this category. They may pose actual or potential threats to plant safety.

The events of this category pertain to the following subcategories:

1. Loss of Essential AC Power

Loss of essential electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite sources for 4160V AC essential buses.

2. Loss of Vital DC Power

Loss of emergency electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of vital plant 125V DC power sources.

3. Loss of Control Room Indications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant increase from these base-line levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

5. RCS Leakage

The reactor vessel provides a volume for the coolant that covers the reactor core. The reactor pressure vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail. Excessive RCS leakage greater than Technical Specification limits indicates potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and containment integrity.

6. RPS Failure

This subcategory includes events related to failure of the Reactor Protective System (RPS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the

RPS to complete a reactor trip comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean any trip failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and containment integrity.

7. Loss of Communications

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

8. Containment Failure

Failure of containment isolation capability (under conditions in which the containment is not currently challenged) warrants emergency classification. Failure of containment pressure control capability also warrants emergency classification.

9. Hazardous Event Affecting Safety Systems

Various natural and technological events that result in degraded plant safety system performance or significant visible damage warrant emergency classification under this subcategory.

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of **all** offsite AC power capability to essential buses for 15 minutes or longer

EAL:

SU1.1	**Unusual Event**
Loss of **all** offsite AC power capability, Table S-1, to essential 4160 V buses MFB-1 and MFB-2 for ≥ 15 min. (Note 1)	

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 AC Power Sources
Offsite:
• Unit Normal Transformer (backcharged)
• Unit Startup Transformer (SWYD)
• Another Unit Startup Transformer (aligned) (SWYD)
• CT5 (Central/energizing Standby Bus)
Emergency:
• Unit Startup Transformer (Keowee)
• Another Unit Startup Transformer (aligned) (Keowee)
• CT4
• CT5 (dedicated line/energizing Standby Bus)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

Basis:

The 4160 V AC System provides the power requirements for operation and safe shutdown of the plant. 4160 V buses MFB-1 and MFB-2 are the essential buses (ref. 1).

The condition indicated by this EAL is the degradation of all offsite AC power sources such that only onsite AC power capability exists for 15 minutes or longer.

Each unit is provided with two physically independent circuits from the switching station. One is the circuit from the 230 kV switching station through the startup transformer, which is designed to be available within a few seconds following a loss of coolant accident. The second circuit is the path from the switchyard through the main step-up transformer, the main generator bus and the unit auxiliary transformer with the generator disconnected from the main bus. The second circuit is currently used during refueling as an additional power feed for the shutdown unit(s) from the 230 kV switchyard. Whenever there is inadequate power from the generating units, the 230 kV switching station and the hydro units, power is available to the standby power buses either directly from the 100 kV Central Tie Substation or from Lee Steam Station via Transformer CT5. (ref. 2)

Upon loss of power from the Oconee generating unit and 230 kV switchyard, power is supplied from both Keowee Hydro Station units through two separate and independent routes. One route is an underground feeder to Transformer CT4 which supplies the two redundant Main Feeder Buses (MFB-1 and MFB-2). The other route is an overhead feeder to the 230 KV switching station which supplies each unit startup transformer. (ref. 3)

The Standby Shutdown Facility (SSF) consists of standby systems for use in an extreme emergency and is equipped with a manually started diesel generator that can supply power necessary to maintain hot shutdown of the reactors of each unit in the event of loss of power. However, the SSF cannot supply power to the essential buses and therefore not credited in this EAL (ref. 3).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses.

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC essential buses. This condition represents a potential reduction in the level of safety of the plant.

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the essential buses, whether or not the buses are powered from it.

Escalation of the emergency classification level would be via IC SA1.

ONS Basis Reference(s):

1. UFSAR Figure 8.1 Single Line Diagram
2. UFSAR Section 8.2 Offsite Power System
3. UFSAR Section 8.3 Onsite Power Systems
4. NEI 99-01 SU1

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of **all but one** AC power source to essential buses for 15 minutes or longer

EAL:

SA1.1 Alert

AC power capability, Table S-1, to essential 4160 V buses MFB-1 and MFB-2 reduced to a single power source for ≥ 15 min. (Note 1)

AND

Any additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 AC Power Sources
Offsite:
• Unit Normal Transformer (backcharged)
• Unit Startup Transformer (SWYD)
• Another Unit Startup Transformer (aligned) (SWYD)
• CT5 (Central/energizing Standby Bus)
Emergency:
• Unit Startup Transformer (Keowee)
• Another Unit Startup Transformer (aligned) (Keowee)
• CT4
• CT5 (dedicated line/energizing Standby Bus)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

For emergency classification purposes, "capability" means that an AC power source is available to and capable of powering the emergency bus(es) within 15 min, whether or not the buses are currently powered from it.

The 4160 V AC System provides the power requirements for operation and safe shutdown of the plant. 4160 V buses MFB-1 and MFB-2 are the essential buses (ref. 1).

The condition indicated by this EAL is the degradation of the offsite and onsite power sources such that any additional single failure would result in a loss of all AC power to the essential buses.

Each unit is provided with two physically independent circuits from the switching station. One is the circuit from the 230 kV switching station through the startup transformer, which is designed to be available within a few seconds following a loss of coolant accident. The second circuit is the path from the switchyard through the main step-up transformer, the main generator bus and the unit auxiliary transformer with the generator disconnected from the main bus. The second circuit is currently used during refueling as an additional power feed for the shutdown unit(s) from the 230 kV switchyard. Whenever there is inadequate power from the generating units, the 230 kV switching station and the hydro units, power is available to the standby power buses either directly from the 100 kV Central Tie Substation or from Lee Steam Station via a Transformer CT5. (ref. 2)

Upon loss of power from the Oconee generating unit and 230 kV switchyard, power is supplied from both Keowee Hydro Station units through two separate and independent routes. One route is an underground feeder to Transformer CT4 which supplies the two redundant Main Feeder Buses (MFB-1 and MFB-2). The other route is an overhead feeder to the 230 KV switching station which supplies each unit startup transformer. (ref. 3)

The Standby Shutdown Facility (SSF) consists of standby systems for use in an extreme emergency and is equipped with a manually started diesel generator that can supply power necessary to maintain hot shutdown of the reactors of each unit in the event of loss of power. However, the SSF cannot supply power to the essential buses and therefore not credited in this EAL (ref. 3).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. If the capability of a second source of emergency bus power is not restored within 15 minutes, an Alert is declared under this EAL.

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

An "AC power source" is a source recognized in APs and EOPs, and capable of supplying required power to an essential bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one essential power source (e.g., CT4, CT5, CT1, 2 ,3 (Keowee)).

- A loss of essential power sources (e.g., CT4, CT5, CT1, CT2, CT3 (Keowee)) with a single train of essential buses being back-fed from an offsite power source.

Escalation of the emergency classification level would be via IC SS1.

ONS Basis Reference(s):

1. UFSAR Figure 8.1 Single Line Diagram
2. UFSAR Section 8.2 Offsite Power System
3. UFSAR Section 8.3 Onsite Power Systems
4. NEI 99-01 SA1

Category: S – System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of **all** offsite power and **all** emergency AC power to essential buses for 15 minutes or longer

EAL:

SS1.1	**Site Area Emergency**
Loss of **all** offsite and **all** emergency AC power capability to essential 4160 V buses MFB-1 and MFB-2 for ≥ 15 min. (Note 1)	

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

For emergency classification purposes, "capability" means that an AC power source is available to and capable of powering the emergency bus(es) within 15 minutes, whether or not the buses are currently powered from it.

The condition indicated by this EAL is the degradation of the offsite and emergency power sources resulting in a loss of all AC power to the emergency buses. 4160 V buses MFB-1 and MFB-2 are the essential buses (ref. 1).

Each unit is provided with two physically independent circuits from the switching station. One is the circuit from the 230 kV switching station through the startup transformer, which is designed to be available within a few seconds following a loss of coolant accident. The second circuit is the path from the switchyard through the main step-up transformer, the main generator bus and the unit auxiliary transformer with the generator disconnected from the main bus. The second circuit is currently used during refueling as an additional power feed for the shutdown unit(s) from the 230 kV switchyard. Whenever there is inadequate power from the generating units, the 230 kV switching station and the hydro units, power is available to the standby power buses either directly from the 100 kV Central Tie Substation or from Lee Steam Station via a Transformer CT5. (ref. 2)

Upon loss of power from the Oconee generating unit and 230 kV switchyard, power is supplied from both Keowee Hydro Station units through two separate and independent routes. One route is an underground feeder to Transformer CT4 which supplies the two redundant Main Feeder Buses (MFB-1 and MFB-2). The other route is an overhead feeder to the 230 KV switching station which supplies each unit startup transformer. (ref. 3)

The Standby Shutdown Facility (SSF) consists of standby systems for use in an extreme emergency and is equipped with a manually started diesel generator that can supply power necessary to maintain hot shutdown of the reactors of each unit in the event of loss of power.

However, the SSF cannot supply power to the essential buses and therefore not credited in this EAL. (ref. 3).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. The interval begins when both offsite and onsite AC power capability are lost.

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

ONS Basis Reference(s):

1. UFSAR Figure 8.1 Single Line Diagram
2. UFSAR Section 8.2 Offsite Power System
3. UFSAR Section 8.3 Onsite Power Systems
4. NEI 99-01 SS1

Category: S –System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** emergency AC power to essential buses

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** emergency AC power capability to essential 4160 V buses MFB-1 and MFB-2

AND

Failure to power SSF equipment and PSW unavailable

AND EITHER:

- Restoration of at least one essential bus in < 4 hour is **not** likely (Note 1)
- CETC reading > 1200°F

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

For emergency classification purposes, "capability" means that an AC power source is available to and capable of powering the essential bus(es), whether or not the buses are currently powered from it. 4160 V buses MFB-1 and MFB-2 are the essential buses (ref. 1).

Each unit is provided with two physically independent circuits from the switching station. One is the circuit from the 230 kV switching station through the startup transformer, which is designed to be available within a few seconds following a loss of coolant accident. The second circuit is the path from the switchyard through the main step-up transformer, the main generator bus and the unit auxiliary transformer with the generator disconnected from the main bus. The second circuit is currently used during refueling as an additional power feed for the shutdown unit(s) from the 230 kV switchyard. Whenever there is inadequate power from the generating units, the 230 kV switching station and the hydro units, power is available to the standby power buses either directly from the 100 kV Central Tie Substation or from Lee Steam Station via Transformer CT5. (ref. 2)

Upon loss of power from the Oconee generating unit and 230 kV switchyard, power is supplied from both Keowee Hydro Station units through two separate and independent routes. One route is an underground feeder to Transformer CT4 which supplies the two redundant Main

Feeder Buses (MFB-1 and MFB-2). The other route is an overhead feeder to the 230 KV switching station which supplies each unit startup transformer. (ref. 3)

The Standby Shutdown Facility (SSF) consists of standby systems for use in an extreme emergency and is equipped with a manually started diesel generator and Protected Service Water (PSW) power supply that can supply power necessary to maintain hot shutdown of the reactors of each unit in the event of loss of power. Although the SSF requires manual initiation, it is considered in this EAL because it may be capable of powering the SSF load center (ref. 3, 4).

The station blackout coping period is four hours (ref. 5).

Core Exit Thermocouple readings of 1200°F are indicative of superheat conditions and inability to adequately remove heat from the core (ref. 6).

This IC addresses a prolonged loss of all power sources to AC essential buses. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A prolonged loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

The EAL should require declaration of a General Emergency prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC essential bus by the end of the analyzed station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of challenges to multiple fission product barriers.

The estimate for restoring at least one essential bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, protective actions for the public.

The EAL will also require a General Emergency declaration if the loss of AC power results in parameters that indicate an inability to adequately remove decay heat from the core.

ONS Basis Reference(s):

1. UFSAR Figure 8.1 Single Line Diagram
2. UFSAR Section 8.2 Offsite Power System
3. UFSAR Section 8.3 Onsite Power Systems
4. UFSAR Section 9.6 Standby Shutdown Facility
5. UFSAR Section 8.3.2.2.4 Station Blackout Analysis
6. RP/0/A/1000/18 Core Damage Assessment
7. NEI 99-01 SG1

Category: S –System Malfunction

Subcategory: 1 – Loss of Essential AC Power

Initiating Condition: Loss of **all** essential AC and vital DC power sources for 15 minutes or longer

EAL:

SG1.2 General Emergency

Loss of **all** offsite and **all** emergency AC power capability to essential 4160 V buses MFB-1 and MFB-2 for ≥ 15 min.

AND

Failure to power SSF equipment and PSW unavailable

AND

Loss of 125 VDC power based on battery bus voltage indications < 105 VDC on **both** vital DC Distribution Centers DCA and DCB for ≥ 15 min. (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

This EAL is indicated by the loss of all offsite and emergency AC power capability to 4160 V essential buses MFB-1 and MFB-2 for greater than 15 minutes in combination with degraded vital DC power voltage. This EAL addresses operating experience from the March 2011 accident at Fukushima Daiichi.

For emergency classification purposes, "capability" means that an AC power source is available to and capable of powering the emergency bus(es) within 15 minutes, whether or not the buses are currently powered from it.

Each unit is provided with two physically independent circuits from the switching station. One is the circuit from the 230 kV switching station through the startup transformer, which is designed to be available within a few seconds following a loss of coolant accident. The second circuit is the path from the switchyard through the main step-up transformer, the main generator bus and the unit auxiliary transformer with the generator disconnected from the main bus. The second circuit is currently used during refueling as an additional power feed for the shutdown unit(s) from the 230 kV switchyard. Whenever there is inadequate power from the generating units, the 230 kV switching station and the hydro units, power is available to the standby power buses either directly from the 100 kV Central Tie Substation or from Lee Steam Station via Transformer CT5. (ref. 2)

Upon loss of power from the Oconee generating unit and 230 kV switchyard, power is supplied from both Keowee Hydro Station units through two separate and independent routes. One route is an underground feeder to Transformer CT4 which supplies the two redundant Main Feeder Buses (MFB-1 and MFB-2). The other route is an overhead feeder to the 230 KV switching station which supplies each unit startup transformer. (ref. 3)

The Standby Shutdown Facility (SSF) consists of standby systems for use in an extreme emergency and is equipped with a manually started diesel generator and Protected Service Water (PSW) power supply that can supply power necessary to maintain hot shutdown of the reactors of each unit in the event of loss of power. Although the SSF requires manual initiation, it is considered in this EAL because it may be capable of powering the SSF load center (ref. 3).

For each unit, two independent and physically separated 125 volt DC batteries and DC buses are provided for the vital instrumentation and control power system. (ref. 4, 5). Minimum DC bus voltage is 105 VDC (ref. 6).

This IC addresses a concurrent and prolonged loss of both emergency AC and Vital DC power. A loss of all emergency AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both emergency AC and vital DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

ONS Basis Reference(s):

1. UFSAR Figure 8.1 Single Line Diagram
2. UFSAR Section 8.2 Offsite Power System
3. UFSAR Section 8.3 Onsite Power Systems
4. UFSAR Figure 8.5 Typical DC and AC Vital Power System - Single Line
5. UFSAR Section 8.3.2 DC Power Systems
6. EP/*/A/1800/001 Blackout Tab
7. NEI 99-01 SG8

Category: S – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

Initiating Condition: Loss of all vital DC power for 15 minutes or longer

EAL:

SS2.1	**Site Area Emergency**
Loss of 125 VDC power based on battery bus voltage indications < 105 VDC on **both** vital DC Distribution Centers DCA and DCB for ≥ 15 min. (Note 1)	

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

For each unit, two independent and physically separated 125 volt DC batteries and DC buses are provided for the vital instrumentation and control power system. (ref. 1, 2). Minimum DC bus voltage is 105 VDC (ref. 3).

This IC addresses a loss of vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

ONS Basis Reference(s):

1. UFSAR Figure 8.5 Typical DC and AC Vital Power System - Single Line
2. UFSAR Section 8.3.2 DC Power Systems
3. EP/*/A/1800/001 Blackout Tab
4. Technical Specifications 3.8.3 DC Sources – Operating
5. NEI 99-01 SS8

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer

EAL:

SU3.1	**Unusual Event**
An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)	

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • CETC temperature • Level in at least one S/G • EFW flow to at least one S/G

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The SPDS serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1).

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via IC SA3.

ONS Basis Reference(s):

1. UFSAR Section 7.5 Display Instrumentation
2. NEI 99-01 SU2

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress

EAL:

SA3.1 Alert

An UNPLANNED event results in the inability to monitor **one or more** Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

Any significant transient is in progress, Table S-3

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • CETC temperature • Level in at least one S/G • EFW flow to at least one S/G

Table S-3 Significant Transients
• Reactor trip • Runback > 25% thermal power • Electrical load rejection > 25% electrical load • ECCS actuation

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The SPDS serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1).

Significant transients are listed in Table S-3 and include response to automatic or manually initiated functions such as reactor trips, runbacks involving greater than 25% thermal power change, electrical load rejections of greater than 25% full electrical load, reactor power cutbacks or ECCS (SI) injection actuations.

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via ICs FS1 or IC RS1

ONS Basis Reference(s):

1. UFSAR Section 7.5 Display Instrumentation
2. NEI 99-01 SA2

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.1	**Unusual Event**
RCS activity > 50 μCi/gm Dose Equivalent I-131 for > 48 hr continuous period	
OR	
RCS activity > 280 μCi/gm Dose Equivalent Xe-133 for > 48 hr continuous period	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

The specific iodine activity is limited to ≤ 50 μCi/gm Dose Equivalent I-131 for > 48 hr continuous period. The specific Xe-133 activity is limited to ≤ 280 μCi/gm Dose Equivalent Xe-133 for > 48 hr continuous period. Entry into Condition C of LCO 3.4.11 meets the intent of this EAL (ref 1).

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

ONS Basis Reference(s):

1. ONS Technical Specifications LCO 3.4.11 RCS Specific Activity
2. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1 Unusual Event

RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min.

OR

RCS identified leakage > 25 gpm for ≥ 15 min.

OR

Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min. (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Manual or computer-based methods of performing an RCS inventory balance are normally used to determine RCS leakage (ref. 1).

Identified leakage includes (ref. 2):

- Leakage such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank, or
- Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage, or
- RCS leakage through a steam generator to the secondary system.

Unidentified leakage is all leakage (except RCP seal water injection or leakoff) that is not identified leakage (ref. 2).

Pressure Boundary leakage is leakage (except SG leakage) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall (ref. 2).

Reactor coolant leakage outside of the containment that is not considered identified or unidentified leakage per Technical Specifications includes leakage via interfacing systems.

Escalation of this EAL to the Alert level is via Category F, Fission Product Barrier Degradation, EAL FA1.1.

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

The first and second EAL conditions are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications). The third condition addresses an RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These conditions thus apply to leakage into the containment, a secondary-side system (e.g., steam generator tube leakage) or a location outside of containment.

The leak rate values for each condition were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). The first condition uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. An emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated).

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

Escalation of the emergency classification level would be via ICs of Recognition Category R or F.

ONS Basis Reference(s):

1. PT/1,2,3/A/0600/010 Reactor Coolant Leakage
2. ONS Technical Specifications Section 1.1 Definitions
2. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

SU6.1	**Unusual Event**
An automatic trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** RPS setpoint is exceeded **AND** A subsequent automatic trip or the manual trip pushbutton is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)	

Note 8: A manual trip action is **any** Control Room operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protective System (RPS) trip function. A reactor trip is automatically initiated by the RPS when certain continuously monitored parameters exceed predetermined setpoints (ref. 1).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from an automatic reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. For the purpose of emergency classification a successful trip has occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power below the Power Operation Mode threshold of 5% (ref. 2).

5% rated power is the Power Operation mode threshold. Below 5%, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than or equal to 5 % power (ref. 1, 2).

For the purposes of emergency classification, a successful manual trip action is that which can be quickly performed from the reactor control console (actuation of the manual trip pushbutton). There is a separate set of switch contacts in series with the output of each reactor trip component. All switch contacts are actuated through a mechanical linkage from a single pushbutton. Reactor shutdown achieved by use of other trip actions such as opening supply breakers, emergency boration, or manually driving control rods) do not constitute a successful manual trip (ref. 3).

Following any automatic RPS trip signal, insertion of redundant manual trip signals are performed to back up the automatic RPS trip function and ensure reactor shutdown is achieved. Even if the first subsequent manual trip signal inserts all control rods to the full-in position immediately after the initial failure of the automatic trip, the lowest level of classification that must be declared is an Unusual Event.

In the event that the operator identifies a reactor trip is imminent and initiates a successful manual reactor trip before the automatic RPS trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. However, if subsequent manual reactor trip actions fail to reduce reactor power below 5%, the event escalates to the Alert under EAL SA6.1.

If by procedure, operator actions include the initiation of an immediate manual trip following receipt of an automatic trip signal and there are no clear indications that the automatic trip failed (such as a time delay following indications that a trip setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic trip or manual actions. If a subsequent review of the trip actuation indications reveals that the automatic trip did not cause the reactor to be shut down, consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 10CFR50.72 should be considered for the transient event.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic trip is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor trip, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor. Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

Should a reactor trip signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the trip failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

ONS Basis Reference(s):

1. ONS Technical Specifications Section 3.3.1 Reactor Protective System (RPS) Instrumentation – Operating
2. ONS Technical Specifications Table 1.1-1 Modes
3. UFSAR Section 7.2.3.7 Manual Trip
4. OSC-02820 Emergency Procedure Setpoints, Setpoint No. 7.1.7.5
5. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

SU6.2	**Unusual Event**
A manual trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** manual trip action was initiated **AND** A subsequent automatic trip or the manual trip pushbutton is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)	

Note 8: A manual trip action is **any** Control Room operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

This EAL addresses a failure of a manually initiated trip in the absence of having exceeded an automatic RPS trip setpoint and a subsequent automatic or manual trip is successful in shutting down the reactor (ref. 1).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from an automatic reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. For the purpose of emergency classification a successful trip has occurred when there is sufficient rod insertion from the manual trip to bring the reactor power below the Power Operation Mode threshold level of 5% (ref. 2).

5% rated power is the Power Operation mode threshold. Below 5%, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than or equal to 5% power (ref. 1, 2).

For the purposes of emergency classification, a successful manual trip action is that which can be quickly performed from the reactor control console (actuation of the manual trip pushbutton). There is a separate set of switch contacts in series with the output of each reactor trip component. All switch contacts are actuated through a mechanical linkage from a single pushbutton. Reactor shutdown achieved by use of other trip actions such as opening supply breakers, emergency boration, or manually driving control rods) do not constitute a successful manual trip (ref. 3).

Following any automatic RPS trip signal, insertion of redundant manual trip signals are performed to back up the automatic RPS trip function and ensure reactor shutdown is achieved. Even if a subsequent automatic trip signal or the first subsequent manual trip signal inserts all control rods to the full-in position immediately after the initial failure of the manual trip, the lowest level of classification that must be declared is an Unusual Event.

If both subsequent automatic and subsequent manual reactor trip actions in the Control Room fail to reduce reactor power < 5% following a failure of an initial manual trip, the event escalates to an Alert under EAL SA6.1.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic trip is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor trip, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor. Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

Should a reactor trip signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the trip failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

ONS Basis Reference(s):

1. ONS Technical Specifications Section 3.3.1 Reactor Protective System (RPS) Instrumentation – Operating
2. ONS Technical Specifications Table 1.1-1
3. UFSAR Section 7.2.3.7 Manual Trip
4. OSC-02820 Emergency Procedure Setpoints, Setpoint No. 7.1.7.5
5. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor

EAL:

SA6.1	**Alert**

An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%

AND

Manual trip pushbutton is **not** successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8)

Note 8: A manual trip action is **any** Control Room operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

This EAL addresses any automatic or manual reactor trip signal that fails to shut down the reactor followed by a subsequent manual trip that fails to shut down the reactor to an extent the reactor is producing significant power (ref. 1).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from an automatic reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. For the purpose of emergency classification a successful trip has occurred when there is sufficient rod insertion from the manual trip to bring the reactor power below 5% (ref. 2).

5% rated power is the Power Operation mode threshold. Below 5%, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than or equal to 5% power (1, 2).

For the purposes of emergency classification, a successful manual trip action is that which can be quickly performed from the reactor control console (actuation of the manual trip pushbutton). There is a separate set of switch contacts in series with the output of each reactor trip component. All switch contacts are actuated through a mechanical linkage from a single pushbutton. Reactor shutdown achieved by use of other trip actions such as opening supply breakers, emergency boration, or manually driving control rods) do not constitute a successful manual trip (ref. 3).

Escalation of this event to a Site Area Emergency would be under EAL SS6.1 or Emergency Coordinator judgment.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control console (e.g., locally opening breakers). Actions taken at back panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control console".

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shut down the reactor is prolonged enough to cause a challenge to the core cooling or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6 or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

ONS Basis Reference(s):

1. ONS Technical Specifications Section 3.3.1 Reactor Protective System (RPS) Instrumentation – Operating
2. ONS Technical Specifications Table 1.1-1
3. UFSAR Section 7.2.3.7 Manual Trip
4. OSC-02820 Emergency Procedure Setpoints, Setpoint No. 7.1.7.5
5. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

EAL:

SS6.1 Site Area Emergency
An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%
AND
All actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5%
AND EITHER:
• CETCs >1200°F on ICCM
• RCS subcooling < 0°F

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

This EAL addresses the following:

- Any automatic reactor trip signal (ref. 1) followed by a manual trip that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed (ref. 5), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS Barriers.

Reactor shutdown achieved by use of other trip actions such as opening supply breakers, emergency boration, or manually driving control rods are also credited as a successful manual trip provided reactor power can be reduced below 5% before indications of an extreme challenge to either core cooling or heat removal exist (ref. 2, 3).

5% rated power is the Power Operation mode threshold. Below 5%, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than or equal to 5% power.

Indication of continuing core cooling degradation is manifested by CETCs are reading greater than 1200°F. This setpoint is used as an indication of an extreme ICC condition and entry into the Oconee Severe Accident Guidelines (OSAG) is initiated for further mitigative actions (ref. 4).

Indication of inability to adequately remove heat from the RCS is manifested by subcooling less than 0°F (ref. 6).

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shut down the reactor. The inclusion of this IC and EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Escalation of the emergency classification level would be via IC RG1 or FG1.

ONS Basis Reference(s):

1. ONS Technical Specifications Section 3.3.1 Reactor Protective System (RPS) Instrumentation – Operating
2. ONS Technical Specifications Table 1.1-1
3. UFSAR Section 7.2.3.7 Manual Trip
4. OSC-02820 Emergency Procedure Setpoints, Setpoint No. 7.1.1.7
5. OSC-02820 Emergency Procedure Setpoints, Setpoint No. 7.1.7.5
6. EP/1,2,3/A/1800/001 Loss of Subcooling Margin
7. NEI 99-01 SS5

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

SU7.1 Unusual Event

Loss of **all** Table S-4 onsite communication methods

OR

Loss of **all** Table S-4 offsite communication methods

OR

Loss of **all** Table S-4 NRC communication methods

Table S-4 Communication Methods

System	Onsite	Offsite	NRC
Commercial phone service	X	X	X
ONS site phone system	X	X	X
EOF phone system		X	X
Public Address system	X		
Onsite radio system	X		
DEMNET		X	
Offsite radio system		X	
NRC Emergency Telephone System			X
Satellite Phone		X	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Onsite, offsite and NRC communications include one or more of the systems listed in Table S-4 (ref. 1).

1. Commercial phone service

   The Commercial phone service does not go through the site telephone system.

2. ONS site phone system

   The site phone system is generator and battery backed with:

   - Fiber-Optic to Charlotte GO (65 lines)
   - Telephone line to Easley (6 circuits)
   - Anderson (4 lines)
   - Six Mile (4 lines)
   - Site Telephone System: Inward and outward direct dial available from the Control Room, TSC, and OSC

3. EOF phone system

   The emergency communications systems at the Charlotte EOF are designed to ensure the reliable, timely flow of information between all parties having an emergency response role.

4. Public Address (Paging) system

   The paging system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature.

5. Onsite radio system

   The onsite radio system receives emergency backup power from Keowee Hydro Units supporting communications with: Control Room 1&2, 3, Fire Brigade, Chemistry, Safety, Radiation Protection, Maintenance, Medical Emergency Response Team, and Hazardous Materials Response Team.

6. DEMNET

   DEMNET is the primary means of offsite communication. This circuit allows intercommunication among the EOF, TSC, control room, counties, and states. DEMNET operates as an internet based (VoIP) communications system with a satellite back-up. Should the internet transfer rate become slow or unavailable, the DEMNET will automatically transfer to satellite mode.

7. Offsite radio system

   The offsite radio system is battery backed supporting communications with: Control Room Units 1&2, TSC, Field Monitoring Teams, EOF, counties and State of South Carolina.

8. NRC Emergency Telephone System (ETS)

   The NRC uses a Duke Energy dedicated telephone line which allows direct telephone communications from the plant to NRC regional and national offices. The Duke Energy communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the Oconee Control Rooms, TSC, and EOF and can be used to establish NRC Emergency Notification System (ENS) and Health Physics Network (HPN) capability.

9. Satellite Phone

Satellite Phones can be used for external communications

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

This IC addresses a significant loss of on-site or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the State EOC and FEO, Pickens County LEC and EOC, and Oconee County LEC and EOC.

The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

ONS Basis Reference(s):

1. ONS Emergency Plan, Section 7.2 Communications Systems
2. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 8 – Containment Failure

Initiating Condition: Failure to isolate containment or loss of containment pressure control.

EAL:

SU8.1 Unusual Event

Any penetration is **not** closed within 15 min. of a VALID ES actuation signal

OR

Containment pressure > 10 psig with < one full train of containment heat removal system (1 RBS with > 700 gpm spray flow **AND** 2 RBCUs) operating per design for ≥ 15 min. (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

VALID - An indication, report, or condition, is considered to be valid when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

Basis:

Reactor Building isolations are initiated by Engineered Safeguards Actuation Channels 5 and 6 in response to a high reactor building pressure signal (3.0 psig) (ref. 1, 2, 4).

Two engineered safeguards systems, the Reactor Building Spray System and the Reactor Building Cooling System, are provided to remove heat from the containment atmosphere following an accident. Both the Reactor Building Spray System and the Reactor Building Cooling System, with either at full capacity, are individually capable of maintaining the containment pressure below the design limit following a LOCA or MSLB. (ref. 1, 3)

The Reactor Building Spray (RBS) System consists of two separate trains of equal capacity. Spray flow greater or equal to 700 gpm satisfies the spray flow design requirement. The Reactor Building pressure setpoint (10 psig) is the pressure at which the Reactor Building Spray equipment should actuate and begin performing its function (ref. 1, 2, 3, 5).

Each of three Reactor Building Cooling Units (RBCUs) consists of a fan, cooling coils, and the required distribution duct work. The Reactor Building atmosphere is circulated past cooling coils by fans and returned to the building. Cooling water for the cooling units is supplied by the Low Pressure Service Water System. The Reactor Building Cooling System provides the design heat removal capacity with two of three coolers operating (ref. 1).

This EAL addresses a failure of one or more containment penetrations to automatically isolate (close) when required by an actuation signal. It also addresses an event that results in high containment pressure with a concurrent failure of containment pressure control systems. Absent challenges to another fission product barrier, either condition represents potential degradation of the level of safety of the plant.

For the first condition, the containment isolation signal must be generated as the result on an off-normal/accident condition (e.g., a safety injection or high containment pressure); a failure resulting from testing or maintenance does not warrant classification. The determination of containment and penetration status – isolated or not isolated – should be made in accordance with the appropriate criteria contained in the plant APs and EOPs. The 15-minute criterion is included to allow operators time to manually isolate the required penetrations, if possible.

The second condition addresses a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment sprays) are either lost or performing in a degraded manner.

This event would escalate to a Site Area Emergency in accordance with IC FS1 if there were a concurrent loss or potential loss of either the Fuel Clad or RCS fission product barriers.

ONS Basis Reference(s):

1. UFSAR Section 6.2.2 Containment Heat Removal Systems
2. UFSAR Table 7-2 Engineered Safeguards Actuation Conditions
3. UFSAR Table 6-25 Minimum Acceptable Combinations of Containment Heat Removal Equipment Performance
4. OSC-02820 Emergency Procedure Setpoints, Setpoint No. 7.1.1.1
5. OSC-02820 Emergency Procedure Setpoints, Setpoint No. 7.4.1.2
6. NEI 99-01 SU7

Category: S – System Malfunction

Subcategory: 9 – Hazardous Event Affecting Safety Systems

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

EAL:

SA9.1 Alert

The occurrence of **any** Table S-5 hazardous event

AND EITHER:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Table S-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

VISIBLE DAMAGE - Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

Basis:

- The significance of seismic events are discussed under EAL HU2.1 (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps.
- External flooding at ONS is highly unlikely since the probable maximum flood (PMF) would be contained by the Keowee Reservoir. Plant grade elevation is 796.0 ft MSL. The minimum external access elevation for the Auxiliary, Turbine, and Service Buildings is 796.5 ft MSL which provides a 6 inch water sill. (ref. 2)
- High winds in excess of design (95 mph) or tornado strikes can cause significant structural damage (ref. 3).
- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area (ref. 4, 5).
- An explosion that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

The first condition addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

The second condition addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC FS1 or RS1.

ONS Basis Reference(s):

1. AP/0/A/1700/005 Earthquake
2. UFSAR Section 3.4.1.1 Flood Protection Measures for Seismic Class 1 Structures
3. UFSAR Section 3.3.1.1 Design Wind Velocity
4. OSS-0254.00-00-4008 Design Bases Specification for Fire Protection
5. AP/1,2,3/A/1700/050 Challenging Plant Fire
6. NEI 99-01 SA9

Category F – Fission Product Barrier Degradation

EAL Group: Hot Conditions (RCS temperature > 200ºF); EALs in this category are applicable only in one or more hot operating modes.

EALs in this category represent threats to the defense in depth design concept that precludes the release of highly radioactive fission products to the environment. This concept relies on multiple physical barriers any one of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment. The primary fission product barriers are:

A. Fuel Clad (FC): The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.
B. Reactor Coolant System (RCS): 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.
C. Containment (CMT): The Containment (Reactor Building) Barrier includes the Reactor Building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the Reactor Building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 2). "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Alert:
Any loss or any potential loss of either Fuel Clad or RCS Barrier

Site Area Emergency:
Loss or potential loss of any two barriers

General Emergency:
Loss of any two barriers and loss or potential loss of third barrier

The logic used for emergency classification based on fission product barrier monitoring should reflect the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier.
- Unusual Event ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.

- For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC RG1 has been exceeded.
- The fission product barrier thresholds specified within a scheme reflect plant-specific ONS design and operating characteristics.
- As used in this category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside the primary containment, an interfacing system, or outside of the primary containment. The release of liquid or steam mass from the RCS due to the as designed/expected operation of a relief valve is not considered to be RCS leakage.
- At the Site Area Emergency level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the Emergency Coordinator would have more assurance that there was no immediate need to escalate to a General Emergency.

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **Any** loss or **any** potential loss of either Fuel Clad or RCS barrier

EAL:

FA1.1	**Alert**
Any loss or **any** potential loss of either Fuel Clad or RCS barrier (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Alert classification level, Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1

ONS Basis Reference(s):

1. NEI 99-01 FA1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss or potential loss of **any** two barriers

EAL:

FS1.1	**Site Area Emergency**
Loss or potential loss of **any** two barriers (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Site Area Emergency classification level, each barrier is weighted equally. A Site Area Emergency is therefore appropriate for any combination of the following conditions:

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

At the Site Area Emergency classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, the Emergency Coordinator would have greater assurance that escalation to a General Emergency is less imminent.

ONS Basis Reference(s):

1. NEI 99-01 FS1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss of **any** two barriers and loss or potential loss of third barrier

EAL:

FG1.1	**General Emergency**
Loss of **any** two barriers **AND** Loss or potential loss of third barrier (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the General Emergency classification level each barrier is weighted equally. A General Emergency is therefore appropriate for any combination of the following conditions:

- Loss of Fuel Clad, RCS and Containment Barriers
- Loss of Fuel Clad and RCS Barriers with potential loss of Containment Barrier
- Loss of RCS and Containment Barriers with potential loss of Fuel Clad Barrier
- Loss of Fuel Clad and Containment Barriers with potential loss of RCS Barrier

ONS Basis Reference(s):

1. NEI 99-01 FG1

ATTACHMENT 2

Fission Product Barrier Loss/Potential Loss Matrix and Bases

Introduction

Table F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Containment). The table is structured so that each of the three barriers occupies adjacent columns. Each fission product barrier column is further divided into two columns; one for Loss thresholds and one for Potential Loss thresholds.

The first column of the table (to the left of the Fuel Clad Loss column) lists the categories (types) of fission product barrier thresholds. The fission product barrier categories are:

A. RCS or SG Tube Leakage

B. Inadequate Heat removal

C. CMT Radiation / RCS Activity

D. CMT Integrity or Bypass

E. Emergency Coordinator Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word "None" is entered in the cell.

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad Barrier Loss in Category A would be assigned "FC Loss A.1," the third Containment Barrier Potential Loss in Category C would be assigned "CMT P-Loss C.3," etc.

If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of Table F-1, locates the likely category and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine if a threshold has been exceeded. If a threshold has not been exceeded, the EAL-user proceeds to the next likely category and continues review of the thresholds in the new category

If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded, only that one barrier is lost or potentially lost. The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if containment radiation is sufficiently high, a Loss of the Fuel Clad and RCS Barriers and a Potential Loss of the Containment Barrier can occur. Barrier

Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, and FA1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad Barrier threshold bases appear first, followed by the RCS Barrier and finally the Containment Barrier threshold bases. In each barrier, the bases are given according category Loss followed by category Potential Loss beginning with Category A, then B,..., E.

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad (FC) Barrier		Reactor Coolant System (RCS) Barrier		Containment (CMT) Barrier	
Category	Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
A RCS or SG Tube Leakage	None	1. RVLS ≤ 0 (Note 9)	1. An automatic or manual ES actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	1. RCS leakage > normal makeup capacity due to **EITHER:** • UNISOLABLE RCS leakage • SG tube leakage 2. RCS cooldown < 400°F at > 100°F/hr **OR** HPI has operated in the injection mode with **no** RCPs operating	1. A leaking SG is FAULTED outside of containment	None
B Inadequate Heat Removal	1. CETCs > 1200°F	1. CETCs > 700°F 2. RCS heat removal **cannot** be established **AND** RCS subcooling < 0 °F	None	1. RCS heat removal **cannot** be established **AND** RCS subcooling < 0 °F 2. HPI forced cooling initiated	None	1. CETCs > 1200°F **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C CMT Radiation / RCS Activity	1. 1/2/3RIA 57/58 > Table F-2 column "FC Loss" 2. Coolant activity > 300 µCi/ml DEI	None	1. Containment radiation: • 1,3 RIA 57/58 > 1.0 R/hr • 2 RIA 57 > 1.6 R/hr • 2 RIA 58 > 1.0 R/hr	None	None	1. 1/2/3RIA 57/58 > Table F-2 column "CMT Potential Loss"
D CMT Integrity or Bypass	None	None	None	None	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on Emergency Coordinator judgment • UNISOLABLE pathway from Containment to the environment exists 2. Indications of RCS leakage outside of Containment	1. Containment pressure > 59 psig 2. Containment hydrogen concentration > 4% 3. Containment pressure > 10 psig with < one full train of containment heat removal system (1 RBS with > 700 gpm spray flow **AND** 2 RBCUs) operating per design for ≥ 15 min. (Note 1)
E EC Judgment	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the fuel clad barrier	1. **Any** condition in the judgment of the Emergency Coordinator that indicates potential loss of the Fuel Clad Barrier	1. **Any** condition in the judgment of the Emergency Coordinator that indicates loss of the RCS Barrier	1. **Any** condition in the judgment of the Emergency Coordinator that indicates potential loss of the RCS Barrier	1. **Any** condition in the judgment of the Emergency Coordinator that indicates loss of the Containment Barrier	1. **Any** condition in the judgment of the Emergency Coordinator that indicates potential loss of the Containment Barrier

Barrier: Fuel Clad

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

1. RVLS ≤ 0" (Note 9)

Note 9: RVLS is **not** valid if **EITHER** of the following exists:

- One or more RCPs are running

OR

- LPI pump(s) are running **AND** taking suction from the LPI drop line

Definition(s):

None

Basis:

RVLS indicated level ≤ 0" with all RCPs not running and both LPI pumps taking suction from the drop line not running represents reactor vessel level below the bottom of the RCS hotleg (without instrument uncertainty considered). This is the lowest measurable reactor vessel level and is used in lieu of actual reactor vessel level indication of level at or below top of active fuel.

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

ONS Basis Reference(s):

1. OSC-2820 Emergency Procedure Setpoints, Setpoint No. 7.1.6.5
2. NEI 99-01 RCS or SG Tube Leakage Potential Loss 1.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

1. CETCs > 1200°F

Definition(s):

None

Basis:

CETCs > 1200°F indicates extreme ICC conditions that may result in at least 516°F of superheat.

This reading indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

ONS Basis Reference(s):

1. OSC-2820 Emergency Procedure Setpoints, Setpoint No. 7.1.1.7
2. NEI 99-01 Inadequate Heat Removal Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CETCs > 700°F

Definition(s):

None

Basis:

CETCs > 700°F indicates conditions that may result in at least ~16°F of superheat and that may indicate core uncovery.

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

ONS Basis Reference(s):

1. OSC-2820 Emergency Procedure Setpoints, Setpoint No. 7.1.1.6
2. NEI 99-01 Inadequate Heat Removal Potential Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. RCS heat removal cannot be established **AND** RCS subcooling < 0°F

Definition(s):

None

Basis:

In combination with RCS Potential Loss B.1, meeting this threshold results in a Site Area Emergency.

The combination of these conditions indicates the ultimate heat sink function is under extreme challenge (i.e., superheated). This threshold addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a potential loss of the Fuel Clad Barrier (ref. 1).

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the Fuel Clad Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

ONS Basis Reference(s):

1. OSC-2820 Emergency Procedure Setpoints, Setpoint No. 7.1.3.1
2. NEI 99-01 Inadequate Heat Removal Potential Loss 2.B

Barrier: Fuel Clad

Category: C. CMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. 1/2/3RIA 57/58 > Table F-2 column "FC Loss"

Table F-2 Containment Radiation – R/hr (1/2/3RIA 57/58)				
Time After S/D (Hrs)	**FC Loss**		**CMT Potential Loss**	
	RIA 57	**RIA 58**	**RIA 57**	**RIA 58**
0 - < 0.5	300	140	1500	700
0.5 - < 2.0	80	40	400	195
2.0 - < 8.0	32	15	160	75
≥ 8.0	10	5	50	25

Definition(s):

None

Basis:

The specified containment radiation monitor readings (ref. 1) indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the Containment. The readings are derived assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with approximately 4% fuel cladding failure into the Containment atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications. This value is higher than that specified for RCS barrier Loss #3.

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors RIA 57 and RIA 58 (ref. 1).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold C.1 since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the ECL to a Site Area Emergency.

ONS Basis Reference(s):

1. OSC-5283 ONS Core Damage Assessment Guidelines, Rev. 2, 2/27/12
2. NEI 99-01 CMT Radiation / RCS Activity FC Loss 3.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: C. CMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. Coolant activity > 300 μCi/ml DEI

Definition(s):

None

Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The threshold Dose Equivalent I-131 (DEI) concentration is well above that expected for iodine spikes and corresponds to about 2% to 5% fuel clad damage. When reactor coolant activity reaches this level the Fuel Clad Barrier is considered lost (ref. 1).

This threshold indicates that RCS radioactivity concentration is greater than 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications.

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

ONS Basis Reference(s):

1. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.B

Barrier: Fuel Clad

Category: C. CMT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 2
Fissien Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. Emergency Coordinator Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the judgment of the Emergency Coordinator that indicates loss of the Fuel Clad Barrier

Definition(s):

None

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Fuel Clad Barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety function acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that are to be used by the Emergency Coordinator in determining whether the Fuel Clad barrier is lost

ONS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Fuel Clad Loss 6.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad

Category: E. Emergency Coordinator Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the judgment of the Emergency Coordinator that indicates potential loss of the Fuel Clad Barrier

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Fuel Clad Barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety function acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that are to be used by the Emergency Coordinator in determining whether the Fuel Clad barrier is potentially lost. The Emergency Coordinator should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

ONS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Potential Fuel Clad Loss 6.A

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. An automatic or manual ES actuation required by **EITHER:**
    - UNISOLABLE RCS leakage
    - SG tube RUPTURE

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

RUPTURE - The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

Basis:

A RCS leak should be considered UNISOLABLE if the leak cannot be isolated within 15 minutes

This threshold is based on an UNISOLABLE RCS leak of sufficient size to require an automatic or manual actuation of the Emergency Core Cooling System (ECCS). This condition clearly represents a loss of the RCS Barrier.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED. If a RUPTURED steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.

ONS Basis Reference(s):

1. UFSAR Section 7.3 Engineered Safeguards Protective System
2. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Loss 1.A

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

1. RCS leakage > normal makeup capacity due to **EITHER**:
   - UNISOLABLE RCS leakage
   - SG tube leakage

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

A RCS leak should be considered UNISOLABLE if the leak cannot be isolated within 15 min.

This threshold is based on the inability to maintain liquid inventory within the RCS by normal operation of the High Pressure Injection System (HPI). The HPI includes three pumps. (ref. 1)

Any one HPI pump runout flow rate is 475 gpm (ref. 2).

This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ES actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs an action to restore and maintain pressurizer level due to exceeding NORMAL MAKEUP CAPABILITY.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

If a leaking steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.

ONS Basis Reference(s):

1. UFSAR Section 9.3.2 High Pressure Injection System
2. OSC-2820 Emergency Procedure Setpoints, Setpoint No. 7.3.1.2
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.A

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

2. RCS cooldown to < 400°F at > 100°F/hr

   OR

   HPI has operated in the injection mode with no RCPs operating

Definition(s):

None

Basis:

400°F is the temperature below which a cooldown greater than 100°F/hr requires implementation of Pressurized Thermal Shock (PTS) guidance (rule 8) (ref. 1, 2).

HPI operating in the injection mode with no RCPs operating also invokes Rule 8 (ref. 3).

This condition indicates an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized).

ONS Basis Reference(s):

1. OSC-2820 Emergency Procedure Setpoints, Setpoint No. 7.1.2.7
2. OSC-2820 Emergency Procedure Setpoints, Setpoint No. 7.1.8.7
3. EP/*/A/1800/001 Rule 8 Pressurized Thermal Shock (PTS)
4. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.B

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. RCS heat removal cannot be established

AND

RCS subcooling < 0°F

Definition(s):

None

Basis:

In combination with FC Potential Loss B.1, meeting this threshold results in a Site Area Emergency.

The combination of these conditions indicates the ultimate heat sink function is under extreme challenge (i.e., superheated). This threshold addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a potential loss of the RCS Barrier.

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the RCS Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

Meeting this threshold results in a Site Area Emergency because this threshold is identical to Fuel Clad Barrier Potential Loss threshold B.2; both will be met. This condition warrants a Site Area Emergency declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase RCS pressure to the point where mass will be lost from the system.

ONS Basis Reference(s):

1. OSC-2820 Emergency Procedure Setpoints, Setpoint No. 7.1.3.6
2. NEI 99-01 Inadequate Heat Removal RCS Loss 2.B

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. HPI forced cooling initiated

Definition(s):

None

Basis:

HPI Forced Cooling (Rule 4) is used when the SGs are not capable of heat removal and RCS pressure is greater than 2300 psig. A Pressurizer PORV is opened to relieve pressure until HPI cools the reactor (feed and bleed). (ref. 1)

ONS Basis Reference(s):

1. OSC-2820 Emergency Procedure Setpoints, Setpoint No. 7.1.4.16
2. NEI 99-01 Other Indications Potential Loss 5.A

Barrier: Reactor Coolant System

Category: C. CMT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment radiation:
   - 1,3 RIA 57/58 > 1.0 R/hr
   - 2 RIA 57 > 1.6 R/hr
   - 2 RIA 58 > 1.0 R/hr

Definition(s):

N/A

Basis:

Containment radiation monitor readings greater than the specified values (ref. 1) indicate the release of reactor coolant to the Containment. The readings assume the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the Containment atmosphere. Because of the very high fuel clad integrity, only small amounts of noble gases would be dissolved in the primary coolant.

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors RIA-57 and RIA-58. The difference in the threshold 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 other detectors). (ref. 1)

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold C.1 since it indicates a loss of the RCS Barrier only.

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

ONS Basis Reference(s):

1. OSC-4244 ONS High Range Containment Monitor Correlation Factors for RIA-57 and RIA-58
2. NEI 99-01 CMT Radiation / RCS Activity RCS Loss 3.A

Barrier: Reactor Coolant System

Category: B. CMT Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. Emergency Coordinator Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the judgment of the Emergency Coordinator that indicates loss of the RCS Barrier

Definition(s):

None

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the RCS Barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety function acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Emergency Coordinator in determining whether the RCS Barrier is lost.

ONS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Loss 6.A

Barrier: Reactor Coolant System

Category: E. Emergency Coordinator Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the judgment of the Emergency Coordinator that indicates potential loss of the RCS Barrier

Definition(s):

None

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the RCS Barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety function acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Emergency Coordinator in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

ONS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Potential Loss 6.A

Barrier: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. A leaking SG is FAULTED outside of containment

Definition(s):

FAULTED - The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.

Basis:

This threshold addresses a leaking Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG leakage, is determined in accordance with the thresholds for RCS Barrier Potential Loss A.1 and Loss A.1, respectively. This condition represents a bypass of the containment barrier.

FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably (part of the FAULTED definition) and the FAULTED steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition, the steam generator is still considered FAULTED for emergency classification purposes.

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU4 for the fuel clad barrier (i.e., RCS activity values) and IC SU5 for the RCS barrier (i.e., RCS leak rate values).

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking steam generator directly to atmosphere to cooldown the plant. These type of condition will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG Atmospheric Dump Valve(s) do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown.

Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Following an SG tube leak, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, gland seal exhausters, valve packing, steam traps, terry turbine exhaust, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

The ECLs resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

	Affected SG is FAULTED Outside of Containment?	
P-to-S Leak Rate	**Yes**	**No**
Less than or equal to 25 gpm	No classification	No classification
Greater than 25 gpm	Unusual Event per SU5.1	Unusual Event per SU5.1
Greater than normal makeup pump capacity (*RCS Barrier Potential Loss*)	Site Area Emergency per FS1.1	Alert per FA1.1
Requires an automatic or manual ECCS (ES) actuation (*RCS Barrier Loss*)	Site Area Emergency per FS1.1	Alert per FA1.1

There is no Potential Loss threshold associated with RCS or SG Tube Leakage.

ONS Basis Reference(s):

1. NEI 99-01 RCS or SG Tube Leakage Containment Loss 1.A

Barrier: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CETCs > 1200°F **AND** Restoration procedures **not** effective within 15 min. (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Definition(s):

None

Basis:

Core Exit Thermocouples (CETCs) are a component of Inadequate Core Cooling Instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. Although clad rupture due to high temperature is not expected for CETC readings less than the threshold, temperatures of this magnitude signal significant superheating of the reactor coolant and core uncover (ref. 1).

The restoration procedures are those emergency operating procedures that address the recovery of the RCS and core heat removal acceptance criteria. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing (ref. 1).The 15 minute threshold starts when operator action begins taking procedurally directed functional recovery actions.

If CETC readings are greater than 1,200°F, Fuel Clad barrier is also lost.

This threshold addresses any other factors that may be used by the Emergency Coordinator in determining whether the Containment Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

ONS Basis Reference(s):

1. EP/1,2,3/A/1800/001 Inadequate Core Cooling
2. NEI 99-01 Inadequate Heat Removal Containment Potential Loss 2.A

Barrier: Containment

Category: C. CMT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: C. CMT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

1. 1/2/3RIA 57/58 > Table F-2 column "CMT Potential Loss"

Table F-2 Containment Radiation – R/hr (1/2/3RIA 57/58)				
Time After S/D (Hrs)	**FC Loss**		**CMT Potential Loss**	
	RIA 57	**RIA 58**	**RIA 57**	**RIA 58**
0 - < 0.5	300	140	1500	700
0.5 - < 2.0	80	40	400	195
2.0 - < 8.0	32	15	160	75
≥ 8.0	10	5	50	25

Definition(s):

None

Basis:

Containment radiation monitor readings greater than the values shown (ref. 1) indicate significant fuel damage well in excess of that required for loss of the RCS Barrier and the Fuel Clad Barrier.

The specified containment radiation monitor readings (ref. 1) indicate the release of reactor coolant, with significant fuel damage well in excess of that required for loss of the RCS Barrier and the Fuel Clad Barrier, into the Containment. The readings are derived assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with 20% clad failure into the Containment atmosphere.

Containment radiation readings at or above the Containment Barrier Potential Loss threshold signify a loss of two fission product barriers and Potential Loss of a third, indicating the need to upgrade the emergency classification to a General Emergency.

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors RIA-57 and RIA-58 (ref. 1).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the ECL to a General Emergency.

ONS Basis Reference(s):

1. OSC-5283 ONS Core Damage Assessment Guidelines, Rev. 2, 2/27/12
2. NEI 99-01 CMT Radiation / RCS Activity Containment Potential Loss 3.A

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. Containment isolation is required

   AND EITHER:

   - Containment integrity has been lost based on Emergency Coordinator judgment
   - UNISOLABLE pathway from Containment to the environment exists

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

The pathway should be considered UNISOLABLE if the Containment cannot be isolated within 15 min.

Reactor Building Essential and Non-essential Isolation occurs on an Engineered Safeguards signal of 3 psig (ref. 1).

These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both bulleted thresholds.

First Threshold – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the Emergency Coordinator will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure 1. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

Another example would be a loss or potential loss of the RCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

Second Threshold – Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment atmosphere to the environment. As used here, the term "environment" includes the atmosphere of a room or area, outside the containment, that may, in turn, communicate with the outside-the-plant atmosphere (e.g., through discharge of a ventilation system or atmospheric leakage). Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.

Refer to the top piping run of Figure 1. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Leakage between two interfacing liquid systems, by itself, does not meet this threshold.

Refer to the bottom piping run of Figure 1. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump developed a leak that allowed steam/water to enter the Auxiliary Building, then second threshold would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause the first threshold to be met as well.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable containment leakage through various penetrations or system components. Minor releases may also occur if a containment isolation valve(s) fails to close but the containment atmosphere escapes to an enclosed system. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold A.1.

ONS Basis Reference(s):

1. UFSAR Section 6.2.3 Containment Isolation System
2. NEI 99-01 CMT Integrity or Bypass Containment Loss 4.A

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

2. Indications of RCS leakage outside of Containment

Definition(s):

None

Basis:

Containment sump, temperature, pressure and/or radiation levels will increase if reactor coolant mass is leaking into the containment. If these parameters have not increased, then the reactor coolant mass may be leaking outside of containment (i.e., a containment bypass sequence). Increases in sump, temperature, pressure, flow and/or radiation level readings outside of the containment may indicate that the RCS mass is being lost outside of containment.

Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of RCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside of containment may not increase significantly; however, other unexpected changes in sump levels, area temperatures or pressures, flow rates, etc. should be sufficient to determine if RCS mass is being lost outside of the containment.

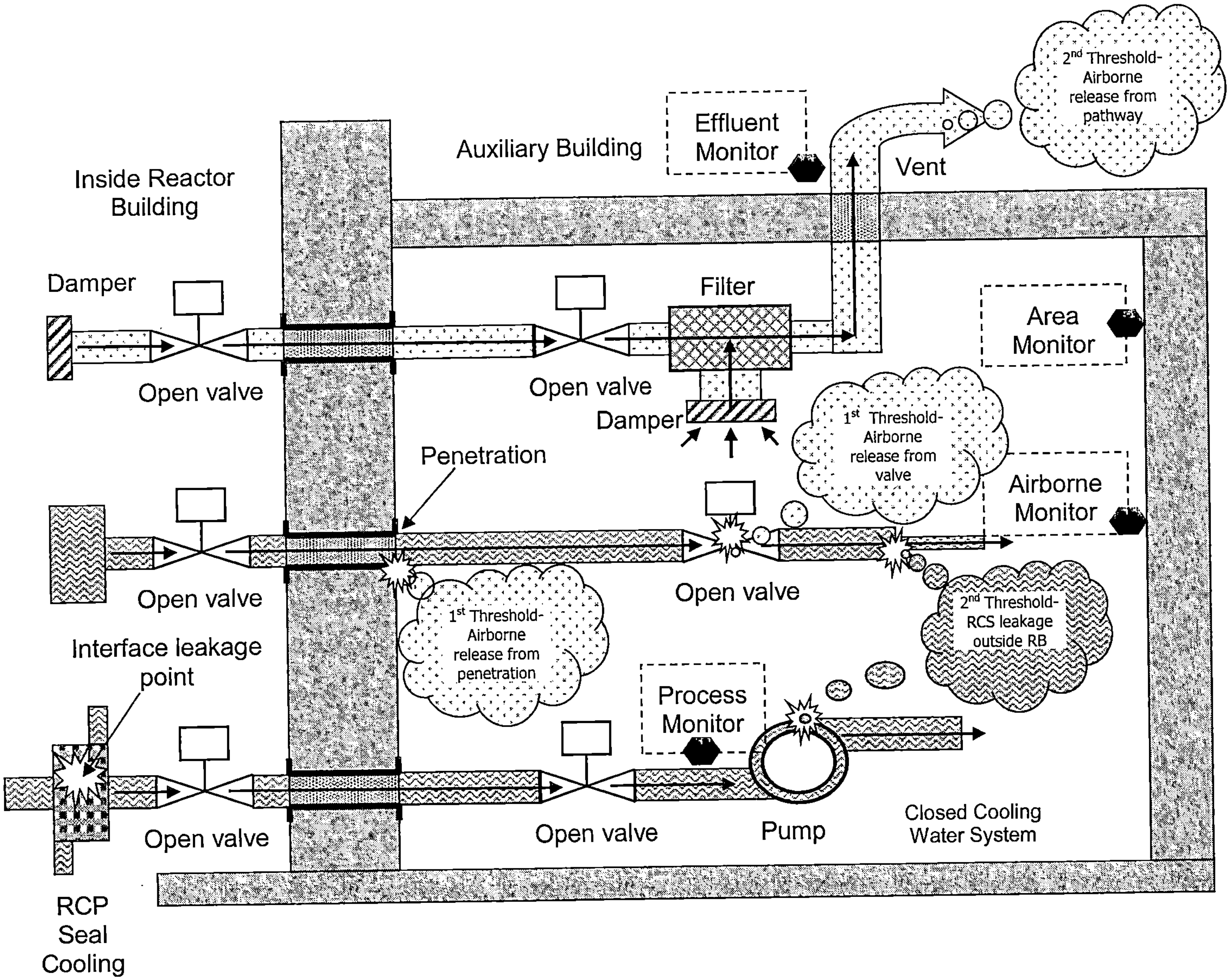
Refer to the middle piping run of Figure 1. In this simplified example, a leak has occurred at a reducer on a pipe carrying reactor coolant in the Auxiliary Building. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure and cause threshold D.1 to be met as well.

To ensure proper escalation of the emergency classification, the RCS leakage outside of containment must be related to the mass loss that is causing the RCS Loss and/or Potential Loss threshold A.1 to be met.

ONS Basis Reference(s):

1. NEI 99-01 CMT Integrity or Bypass Containment Loss

Figure 1: Containment Integrity or Bypass Examples



Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

1. Containment pressure > 59 psig

Definition(s):

None

Basis:

The Reactor Building is designed for an internal pressure of 59 psig (ref. 1).

If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the RCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a Site Area Emergency and General Emergency since there is now a potential to lose the third barrier.

ONS Basis Reference(s):

1. UFSAR Section 6.2.1 Containment Functional Design
2. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.A

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

2. Containment hydrogen concentration ≥ 4%

Definition(s):

None

Basis:

Following a design basis accident, hydrogen gas may be generated inside the containment by reactions such as zirconium metal with water, corrosion of materials of construction and radiolysis of aqueous solution in the core and sump.

The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations. ONS is equipped with a Containment Hydrogen Monitoring System (CHMS) that provides continuous indication of hydrogen concentration in the containment atmosphere. The measurement capability is provided over the range of 0% to 10%. A continuous indication of the hydrogen concentration is not required in the control room at all times during normal operation. If continuous indication of the hydrogen concentration is not available at all times, continuous indication and recording shall be functioning within 90 minutes of the initiation of the safety injection. (ref. 1, 2)

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a potential loss of the Containment Barrier.

ONS Basis Reference(s):

1. UFSAR Section 9.3.7 Containment Hydrogen Monitoring System
2. UFSAR Section 15.16.3 Evaluation of Hydrogen Concentrations
3. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.B

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

3. Containment pressure > 10 psig with < one full train of containment heat removal system (1 RBS with > 700 gpm spray flow **AND** 2 RBCUs) operating per design for ≥ 15 min. (Note 1)

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Definition(s):

None

Basis:

Two engineered safeguards systems, the Reactor Building Spray System and the Reactor Building Cooling System, are provided to remove heat from the containment atmosphere following an accident. Both the Reactor Building Spray System and the Reactor Building Cooling System, with either at full capacity, are individually capable of maintaining the containment pressure below the design limit following a LOCA or MSLB. (ref. 1, 3)

- The Reactor Building Spray (RBS) System consists of two separate trains of equal capacity. Spray flow greater or equal to 700 gpm satisfies the spray flow design requirement. The Reactor Building pressure setpoint (10 psig) is the pressure at which the Reactor Building Spray equipment should actuate and begin performing its function (ref. 1, 2, 3, 4).

- Each of three Reactor Building Cooling Units (RBCUs) consists of a fan, cooling coils, and the required distribution duct work. The Reactor Building atmosphere is circulated past cooling coils by fans and returned to the building. Cooling water for the cooling units is supplied by the Low Pressure Service Water System. The Reactor Building Cooling System provides the design heat removal capacity with two of three coolers operating (ref. 1).

This threshold describes a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. This threshold represents a potential loss of containment in that containment heat removal/depressurization systems (e.g., containment sprays, ice condenser fans, etc., but not including containment venting strategies) are either lost or performing in a degraded manner.

ONS Basis Reference(s):

1. UFSAR Section 6.2.2 Containment Heat Removal Systems
2. UFSAR Table 7-2 Engineered Safeguards Actuation Conditions
3. UFSAR Table 6-25 Minimum Acceptable Combinations of Containment Heat Removal Equipment Performance
4. OSC-02820 Emergency Procedure Setpoints, Setpoint No. 7.4.1.2
5. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.C

Barrier: Containment

Category: E. Emergency Coordinator Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the judgment of the Emergency Coordinator that indicates loss of the Containment Barrier

Definition(s):

None

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Containment Barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety function acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Emergency Coordinator in determining whether the Containment Barrier is lost.

ONS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

Barrier: Containment

Category: E. Emergency Coordinator Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the judgment of the Emergency Coordinator that indicates potential loss of the Containment Barrier

Definition(s):

None

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Containment Barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- <u>Imminent barrier degradation</u> exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term “imminent” refers to recognition of the inability to reach safety function acceptance criteria before completion of all checks.
- <u>Barrier monitoring</u> capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- <u>Dominant accident sequences</u> lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Emergency Coordinator in determining whether the Containment Barrier is lost.

ONS Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Potential Loss 6.A

Background

NEI 99-01 Revision 6 ICs AA3 and HA5 prescribe declaration of an Alert based on impeded access to rooms or areas (due to either area radiation levels or hazardous gas concentrations) where equipment necessary for normal plant operations, cooldown or shutdown is located. These areas are intended to be plant operating mode dependent. Specifically the Developers Notes for AA3 and HA5 states:

The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.

The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).

Further, as specified in IC HA5:

The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.

ATTACHMENT 3

Safe Operation & Shutdown Rooms/Areas Tables R-2 & H-2 Bases

ONS Table R-2 and H-2 Bases

NEI 99-01 Rev 06 addresses elevated radiation levels and hazardous gases in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant shutdown and cool down.

Power Operation was reviewed to determine if any actions are "necessary" to maintain power operations. Over reasonable periods (several days), there are some actions outside the Control Room that are required to be performed to maintain normal operations. The following table lists the locations into which an operator may be dispatched in order perform a normal plant operation, shutdown and cool down.

The review was completed using the following procedures as the controlling documents:

- OP/*/A/1102/010 (Controlling Procedure for Unit Shutdown)
- OP/*/A/1106/001 (Turbine Generator)
- OP/*/A/1106/015 (EHC System)
- OP/*/A/1103/004A (RCS Boration)
- OP/*/A/1104/027 (Bleed Transfer Pump Recirculation)
- PT/*/A/0600/001 B (Surveillance to go to Mode 3)
- OP/*/A/1102/010 (Unit SD Mode 1 to Mode 3)
- IP/*/A/0200/047 (LTOP Calibration)
- OP/*/A/1103/006 (RCP Operations)
- OP/*/A/1104/012 (CCW Pump Operations)
- CP/1/A/2002/014 (RCS Sampling)
- OP/*/A/1104/049 (LTOP Operation)
- OP/1/A/1104/001 (Core Flood Operations)
- OP/0/A/1104/048 (TBS Operations)
- OP/*/A/1104/004 (Low Pressure Injection System)
- OP/*/A/1103/008 (RCS Crud Burst)

Travel paths to the locations where the equipment is operated were considered as part of the determination of affected rooms. ONS Reactor and Auxiliary Building design consist of mostly single entry rooms located off of a common hallway, therefore access to the hallway is required to access a given room. Some equipment is located within the hallway itself.

Room	Mode	Procedure	Enclosure	Steps
TB	1	OP/1/A/1102/010	4.1	Unit SD
TB	1	OP/1/A/1106/001	4.2	TG
TB	1	OP/1/A/1106/014	4.3	MSRH
TB	1	OP/1/A/1106/015	4.2	EHC
A-2 LDST Hatch area	1,2,3	OP/1/A/1103/004 A	4.1	RCS Boration
A-1 hallway 8' S/ col 65	1,2,3	OP/1/A/1103/004 A	4.2	RCS Boration
A-1 hallway 8' S/ col 65	1,2,3	OP/1/A/1103/004 A	4.3	RCS Boration
Unit 1 BTP Rm	1,2,3	OP/1/A/1103/004 A	4.3	RCS Boration

Unit 1 BTP Rm	1,2,3	OP/1/A/1103/004 A	4.4	RCS Boration
A-1 hallway 8' S/ col 65	1,2,3	OP/1/A/1103/004 A	4.4	RCS Boration
A-2-Unit 2 LDST Hatch area	1,2,3	OP/1/A/1103/004 A	4.4	RCS Boration
Unit 2 BTP Rm	1,2,3	OP/1/A/1103/004 A	4.4	RCS Boration
A-1-hallway N of Col 82	1,2,3	OP/1/A/1103/004 A	4.4	RCS Boration
A-1 hallway 5' S/ col 67	1,2,3	OP/1/A/1103/004 A	4.4	RCS Boration
A-1 hallway col 82	1,2,3	OP/1/A/1103/004 A	4.4	RCS Boration
A-2 Unit 3 LDST Hatch area	1,2,3	OP/1/A/1103/004 A	4.5	RCS Boration
Unit 3 BTP Rm	1,2,3	OP/1/A/1103/004 A	4.5	RCS Boration
A-1 hallway 10' S/col 96)	1,2,3	OP/1/A/1103/004 A	4.5	RCS Boration
A-1 hallway 5' S/ col 67	1,2,3	OP/1/A/1103/004 A	4.5	RCS Boration
LPI Cooler Rm 1' W/ North door	1,2,3	OP/1/A/1103/004 A	4.5	RCS Boration
A-1- BAMT Rm	1	OP/1/A/1103/004 A	4.6	RCS Boration
A-1 Unit 1 & 2 BAMT Rm	1	OP/1/A/1103/004 A	4.6	RCS Boration
Rm 111	1	OP/1/A/1103/004 A	4.6	RCS Boration
A-2 LDST Hatch area	1	OP/1/A/1103/004 A	4.6	RCS Boration
CTT Rm	1	OP/1/A/1103/004 A	4.6	RCS Boration
A-2-1&2 Chem. Add Panel	1	OP/1/A/1103/004 A	4.6	RCS Boration
A-1-Col Q70	1	OP/1/A/1103/004 A	4.7	RCS Boration
A-2-LDST Hatch area	1	OP/1/A/1103/004 A	4.7	RCS Boration
A-1-Unit 1 CBAST Rm	1	OP/1/A/1103/004 A	4.7	RCS Boration
Unit 1 BTP Rm	1	OP/1/A/1104/027	4.19	BTP Recirc
Unit 1 BTP Rm	1	OP/1/A/1104/027	4.20.	BTP Recirc
	1	OP/1/A/1102/010	4.2	Unit SD
	1	PT/1/A/0600/001 B	13.2	Surv. Mode3
Unit 1-BTP Rm	1,2	OP/1/A/1103/004	4.5	Makeup
Unit 1-BTP Rm	1,2	OP/1/A/1103/004	4.6	Makeup
	1,2,3	OP/1/A/1102/010	4.3	SD Mode 1 to 3
	3	OP/1/A/1102/010	4.4	
RB 779', Cable Room, 1UB2	3	IP/1/A/0200/047		LTOP Calibration
RB 779'	3	IP/1/A/0200/047		LTOP Calibration
1UB2	3	IP/1/A/0200/047		LTOP Calibration
1AT7	3	IP/1/A/0200/047		LTOP Calibration
1MTC-4	3	IP/1/A/0200/047		LTOP Calibration
1AT5	3	IP/1/A/0200/047		LTOP Calibration
LPI Cooler Room	3	OP/1/A/1103/006	4.12	
	3	OP/1/A/1102/010	4.7	
		OP/1/A/1104/012 A	4.2	CCW Pump
	3	OP/1/A/1102/010	4.7	
Unit 1 Primary Sample Hood		CP/1/A/2002/014	4.2	
AB SAMPLE RM.308		CP/1/A/2002/014	4.2	
A-4-402 PZR Heaters	3	OP/1/A/1102/010	4.7	

Equip Rm XO/XP	3	OP/1/A/1102/010	4.7	
Equip Rm XO/XP	3	OP/1/A/1104/049	4.2	
A-4-402 PZR Heaters	3	OP/1/A/1104/049	4.2	
	3	OP/1/A/1104/001	4.14	
A-4-409	3	OP/1/A/1104/001	4.4	
A-3-308	3	OP/1/A/1104/001	4.4	
A-2 Hallway	3	OP/1/A/1104/001	4.4	
R-1G-W	3	OP/1/A/1104/001	4.4	
R-1-around "A" CFT	3	OP/1/A/1104/001	4.4	
R-B above Emer Sump	3	OP/1/A/1104/001	4.4	
R-B above RBNS	3	OP/1/A/1104/001	4.4	
R-1-around "B" CFT	3	OP/1/A/1104/001	4.4	
R-B-20' above LD Clr RM	3	OP/1/A/1104/001	4.4	
Equip Rm XO/XP	3	OP/1/A/1104/001	4.14	
A-4-W Pent Rm	3	OP/1/A/1104/001	4.14	
A-4-E Pent	3	OP/1/A/1104/049	4.2	
R-3G East Side	3	OP/1/A/1104/049	4.2	
A-4-402	3	OP/1/A/1104/049	4.2	LTOP Alignment
A-2-Col. P-63)	3	OP/1/A/1104/049	4.2	LTOP Alignment
T-3-Equip Rm)	3	OP/1/A/1104/049	4.2	LTOP Alignment
	3	OP/1/A/1102/010	4.7	
	3	OP/1/A/1102/010	4.15	
A-2-Unit 1 BAMT, in hallway	3	OP/1/A/1104/002	4.17	
Turbine Building	3	OP/1/A/1106/002 A	4.14	
Turbine Building	3	OP/0/A/1104/048	4.4	Step 3.7
		OP/1/A/1102/010	4.7	
		Next actions---LPI		
LPI System Start-up (CR & SSF-CR)	3	OP/1/A/1104/004	4.2	LPI Fill & S/U
AB 1st Floor	3	OP/1/A/1104/004	4.5	Valve lineup for LPI
AB Pent. Rooms	3	OP/1/A/1102/010	4.1	Breaker line up S/D
TB-3 & CR	3	OP/1/A/1102/010		Secondary Steam SD
TB All Levels	3	OP/1/A/1102/010	4.1	Align FDW clean-up
AB-2	4 & 5	OP/1/A/1102/010	4.11	RCS H2 Sampling
RB, AB-1, 2 & 3rd	5	OP/1/A/1103/008		RCS Crud Burst

ATTACHMENT 3

Safe Operation & Shutdown Rooms/Areas Tables R-2 & H-2 Bases

Unit Shutdown Room List	Mode
Turbine Building	1,2,3
A-1 hallway 8' S/ col 65	1,2,3
A-1-hallway N of Col 82	1,2,3
A-1 hallway 5' S/ col 67	1,2,3
A-1 hallway col 82	1,2,3
A-1 hallway 10' S/col 96)	1,2,3
A-1- BAMT Rm	1
A-1 Unit 1 & 2 BAMT Rm	1
A-1-Col Q70	1
A-2 LDST Hatch area	1,2,3
A-2-Unit 2 LDST Hatch area	1,2,3
A-2 Unit 3 LDST Hatch area	1,2,3
A-2-1&2 Chem. Add Panel	1
A-2-Col. P-63	3
A-2-Unit 1 BAMT, in hallway	3
A-2 Hallway	3
A-3-308	3
A-4-402	3
A-4-409	3
A-4-W Pent Rm	3
A-4-E Pent	3
Unit 1 BTP Rm	1,2,3
Unit 2 BTP Rm	1,2,3
Unit 3 BTP Rm	1,2,3
U1 LPI Cooler Rm	1,2,3
RB 779', Cable Room, 1UB2	3
RB 779'	3
R-1G-W	3
R-1-around "A" CFT	3
R-1-around "B" CFT	3
R-B above Emer. Sump	3
R-B above RBNS	3
R-B-20' above LD Cooler RM	3
R-3G East Side	3
1UB2	3
1AT7	3
1MTC-4	3
1AT5	3
Unit 1 Primary Sample Hood	3
AB SAMPLE RM.308	3
RB, AB	4 & 5

Table R-2 & H-2 Results

Table R-2 & H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Turbine Building	1, 2, 3
Equipment and Cable Rooms	1, 2, 3
Auxiliary Building	1, 2, 3, 4, 5
Reactor Buildings	3, 4, 5

Emergency Facilities And Equipment

H.1.a Technical Support Center (TSC)

A Technical Support Center has been designated for the Oconee Nuclear Station in the area known as the Operations Center, together with the nearby offices adjacent to the Control Rooms 1&2 on the fifth floor of the Auxiliary Building. This area has the same ventilation and shielding as the Control Room enabling plant management and supporting technical and engineering personnel to evaluate plant status and support operations in conjunction with the Operational Support Center.

The Technical Support Center has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of the reactor operations in the event of an accident, and those persons who are responsible for the management of the accident. Upon activation, this facility will provide the main communication link between the Plant, Operational Support Center, the Nuclear Regulatory Commission Regional Headquarters, and the Emergency Operations Facility. The Technical Support Center is staffed by plant management and technical personnel.

The Technical Support Center has access to the following capabilities and characteristics: (Figure H-1).

1. Redundant two-way communication with the Control Room, the Emergency Operations Facility and the Nuclear Regulatory Commission Operations Center.

2. Monitoring for direct radiation and airborne radioactive contaminants, with local readout of radiation level and alarms if preset levels are exceeded. Laboratory analysis is required if it becomes necessary to detect radioiodines at concentrations as low as 1.0 E-7 microcurie/cc.

3. Display, printout or trending of comprehensive data necessary to monitor reactor systems status and to evaluate plant system abnormalities; in-plant radiological parameters and meteorological parameters are also available. This capability is provided via each unit's Operator Aid Computer.

   Offsite radiological conditions are provided via radio from the field monitoring teams.

4. Ready access to as-built plant drawings such as general arrangement, flow diagrams, electrical one-lines, instrument details, etc.

5. Habitability during postulated radiological accidents to the same degree as the Control Room.

H.1.b Operational Support Center (OSC)(Figure H-2)

An Operational Support Center has been established in the Operations Center located in the Unit 3 Control Room. Personnel assigned to this support center will include the following:

- Work Control
- Chemistry
- Radiation Protection
- Maintenance
- Operations
- Engineering
- Nuclear Supply Chain
- Security

The Operational Support Center has shielding and ventilation to the same degree as the Control Room. Breathing equipment and protective clothing are available in the Operational Support Center should any craftsman/technician be required to perform a task or function in an area that would require protective clothing and breathing apparatus.

H.1.c Alternate Emergency Response Facility (ERF) (Figure H-14 and H-2A)

An Alternate Technical Support Center has been established at the Oconee Office Building, Room 316. Radio and telephone communications are available to offsite agencies and the NRC to the same extent as the designated TSC.

An Alternate Operational Support Center has been established in the Oconee Office Building, Room 316 A. Communication links are provided for information flow both to the Control Room and Technical Support Center.

H. EMERGENCY FACILITIES AND EQUIPMENT

The Issaqueena Trail Facility (JIC) serves as an alternate response facility that would be accessible even if the site is under threat of or experiencing hostile action, to function as a staging area for augmentation of emergency response staff and having the following characteristics required collectively of the alternate facilities for use when onsite emergency facilities cannot be safely accessed during hostile action:

- The capability for communication with the emergency operations facility, control room, and plant security.
- The capability to perform offsite notifications.
- The capability for engineering assessment activities, including damage control team planning and preparation.

H.2 Emergency Operations Facility (EOF) (Figures H3-A and H3-B)

The Emergency Operations Facility is located at the Charlotte General Office in North Carolina. The facility is located approximately 120 miles from the Oconee Nuclear Station.

The EOF has the following capabilities:

a. The capability for obtaining and displaying plant data and radiological information for each reactor at a nuclear power reactor site and for each nuclear power reactor site that the facility serves.

b. The capability to analyze plant technical information and provide technical briefings on event conditions and prognosis to licensee and offsite response organizations for each reactor at a nuclear power reactor site and for each nuclear power reactor site that the facility serves.

c. The capability to support response to events occurring simultaneously at more than one nuclear power reactor site if the emergency operations facility serves more than one site.

H. EMERGENCY FACILITIES AND EQUIPMENT

Two utility circuits feed Energy Center Phase II where the EOF is located. Primary power to the Energy Center is provided by commercial power. All electrical outlets, as well as lighting fixtures and the wiring closet that supports both the voice and data communications in the Energy Center EOF are on generator backed up power. A loss of commercial power should not impact any of the voice or data communications equipment located in the EOF. All common Duke Energy telecom infrastructures that support EOF functions, including, but not limited to, fiber optic transmission equipment, telephone switching equipment and data network routers, is configured to operate from at least one and usually multiple backup power sources in the event of a loss of commercial power. These backup sources include generator, DC battery and UPS systems. EOF HVAC loads are not backed up.

H.3 County, State Emergency Operations Center

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

H.4 Activation and Staffing of the Emergency Response Organization

Activation and staffing of the Emergency Response Organization will be in accordance with the emergency action levels and the procedures developed for determining emergency response.

Division/Section Directives describe the Emergency Response Organization. Figures A-2A, A-2B depicts the procedure for recall of the Emergency Organization.

H.5 Monitoring Systems

On Site - If an emergency situation occurs at the plant, plant personnel continually monitor plant parameters with regard to limits and surveillance requirements specified in the appropriate Technical Specifications, Operating Procedures and Emergency Procedures. These parameters will affect the emergency classification and therefore affect decisions implementing specific emergency measures. In addition to monitoring plant parameters, radiological surveys may be used to verify, augment and/or delineate the assessment of the emergency. (Figure H-20).

H. EMERGENCY FACILITIES AND EQUIPMENT

H.5.a Natural Phenomena Monitors

Natural phenomena instrumentation to monitor wind speed and direction, temperature and vertical temperature gradient (Figure H-4); and seismic activity (Figure H-18).

H.5.b Radiological Monitors (H-5)

Area Radiation Monitoring System

The area radiation monitoring system detectors are located throughout the plant in locations where significant radiation levels may exist, which may change with time and with the operation being performed. They are designed primarily for the protection of personnel performing such operations as routine coolant sampling, refueling, reactor building entry, radioactive waste disposal operations and for certain other operating and maintenance work. The system has sufficient range and flexibility to permit readout during routine operations and during any transient or emergency conditions that may exist. The equipment is self checking for proper operation and alarms both in the local area and in the respective control room. Where necessary or desirable, readout is also provided locally.

Process Radiation Monitoring System

Radiation monitoring of process systems provides early warning of equipment, component, or system malfunctions or potential radiological hazards. The Process Radiation Monitoring System includes alarms, indications, and recording of data in the control rooms. In some cases automatic action is taken upon an alarm condition; in others the alarm serves as a warning to the operator so that manual corrective action can be taken.

Radioactive liquid and gaseous waste effluent are monitored and coordinated by Operations and controlled to assure that radioactivity released does not exceed 10 CFR 20 limits for the plant as a whole.

H. EMERGENCY FACILITIES AND EQUIPMENT

Personnel Monitoring System

Personnel monitoring equipment consisting of film badges and/or their equivalent (thermo-luminescent dosimeters, TLD's), are assigned by the Radiation Protection Section and worn by all personnel at Oconee whose job involves significant levels of radiation exposure as defined in 10 CFR 20. In addition, pocket chambers, electronic dosimeters, self-reading dosimeters, pocket high radiation alarms, wrist badges, and/or finger tabs are readily available for use by those persons who ordinarily work in the Controlled Area or whose job requires frequent access to this area.

Portable Monitors - sufficient numbers are available for use in assessing radiological conditions. (Figure H-6).

Sampling Equipment - sufficient numbers are available for use in assessing radiological conditions. (Figure H-7).

H.5.c Process Monitors - Non-radiological Monitoring

Non-radiological monitoring capabilities include reactor coolant system pressure, temperatures, flows, and water level for detection of inadequate core cooling. Containment pressure, temperature, liquid levels, flow rates, and status of equipment components are monitored to assess containment integrity.

H.5.d Fire and Combustion products detectors - (Figure H-8).

H.6 Offsite Monitoring and Analysis for Emergency Response

H.6.a Natural-Phenomena Monitors

Facilities and equipment include two onsite meteorological towers. Also, an agreement has been established with the Greenville-Spartanburg National Weather service to provide meteorological information should our system become inoperable.

H.6.b Radiological monitors for emergency environmental monitoring are provided in emergency kits. The established environmental monitoring network and sampling equipment in the surrounding area are also available to provide emergency assessment data.

The existing radiological monitoring program will provide base line information as well as in-place monitoring for early assessment data. (Figure A) (H-9 and H-10).

H. EMERGENCY FACILITIES AND EQUIPMENT

Normal environmental monitoring equipment includes radioiodine and particulate continuous air samplers and thermo-luminescent dosimeters, located and collected according to pre-established criteria. Environmental monitoring will be expanded as necessary during an emergency situation in accordance with offsite monitoring procedures.

H.6.c Laboratory Facilities - Include mobile emergency monitoring capabilities available through the S.C. Department of Health and Environmental Control, Bureau of Solid and Hazardous Waste Management and the DOE Radiological Assistance Team. In addition, Oconee Nuclear Station (ONS) has emergency vehicles for mobile assessment purposes. Fixed facilities are available for gross counting and spectral analysis in the ONS counting laboratory (Figure H-11) and at the Duke Energy Environmental Laboratory near the McGuire Nuclear Station, Charlotte, North Carolina.

Should the plant lose the capability to use the count room onsite, samples can be counted at the backup count room or in one of the mobile assessment field monitoring vans. Portable equipment would be relocated to this area. (Figure H-3)

H.7 Offsite radiological monitoring equipment is located in the storage area outside the protected area. Emergency kits are available for off-site monitoring teams who would be monitoring for radiation offsite. (Figure H-12).

H.8 Meteorological Instrumentation

A primary and one auxiliary meteorological tower provides the basic parameters on display in the Control Room. (Figure H-4 shows the meteorological equipment.)

Meteorological measurement equipment meets the criteria of the milestones addressed in Appendix 2 of NUREG 0654 and Proposed Revision 1 to Regulatory Guide 1.23.

An operable dose calculation methodology is in use in the Control Room, Technical Support Center and the Emergency Operations Facility.

The dose assessment methodology for the Oconee Nuclear Station consists of calculations for three separate source terms. The first source term is based on the activity that has been or is actually being released through the unit vent; the second source term is based on a potential release using the reactor building dose rate and design basis assumptions for containment leakage; the third source term is based on the activity that has been or is actually being released through the steam relief valves.

The release rate is calculated for each source term using relative atmospheric dispersion factors calculated by the meteorological model and either actual sample data or radiation monitor readings. These release rates are then added together and used to calculate the dose rate or a projected dose over the duration of the release or over 4 hours if release duration is unknown at 1, 2, 5 and 10 miles downwind from the plant.

These dose assessment methods provide the capability to calculate the dose from actual or potential releases following an accident. A fifty-year committed dose equivalent (CDE) to the thyroid and a total effective dose equivalent (TEDE) from exposure to a semi-infinite cloud and a four-day ground shine as applicable are determined. The dose conversion factors are derived from EPA-400. Near real time radiation monitor readings, sample data, and meteorological data are combined to provide timely, realistic dose calculations. This model will provide the capability to assess and monitor actual or potential offsite consequences of a radiological emergency condition.

Direct telephone access to the person responsible for making offsite dose calculations is available to the Nuclear Regulatory Commission through the use of the NRC Health Physics Network line. The physical location of this person is in the Emergency Operations Facility.

H.9 Operational Support Center - Emergency Supplies

The Operational Support Center will have the same shielding, and ventilation as the Control Room. Protective clothing and breathing equipment are available to the personnel assembled in these areas. (See Figures H-13, H-14, H-17)

H.10 Inspection and Inventory of Emergency Equipment and Supplies

All emergency equipment designated by the Oconee Nuclear Station Emergency Plan shall be inventoried and inspected on a quarterly basis or in agreement with established procedures. Supplies will be inventoried/replaced after each drill and/or exercise or actual emergency where supplies might have been used.

Calibration of any/all emergency equipment shall be at the intervals recommended by the supplier of the equipment.

H. EMERGENCY FACILITIES AND EQUIPMENT

H.11 Identification of Emergency Kits

Emergency kits are located in various locations. See figures below and procedure for specific locations.

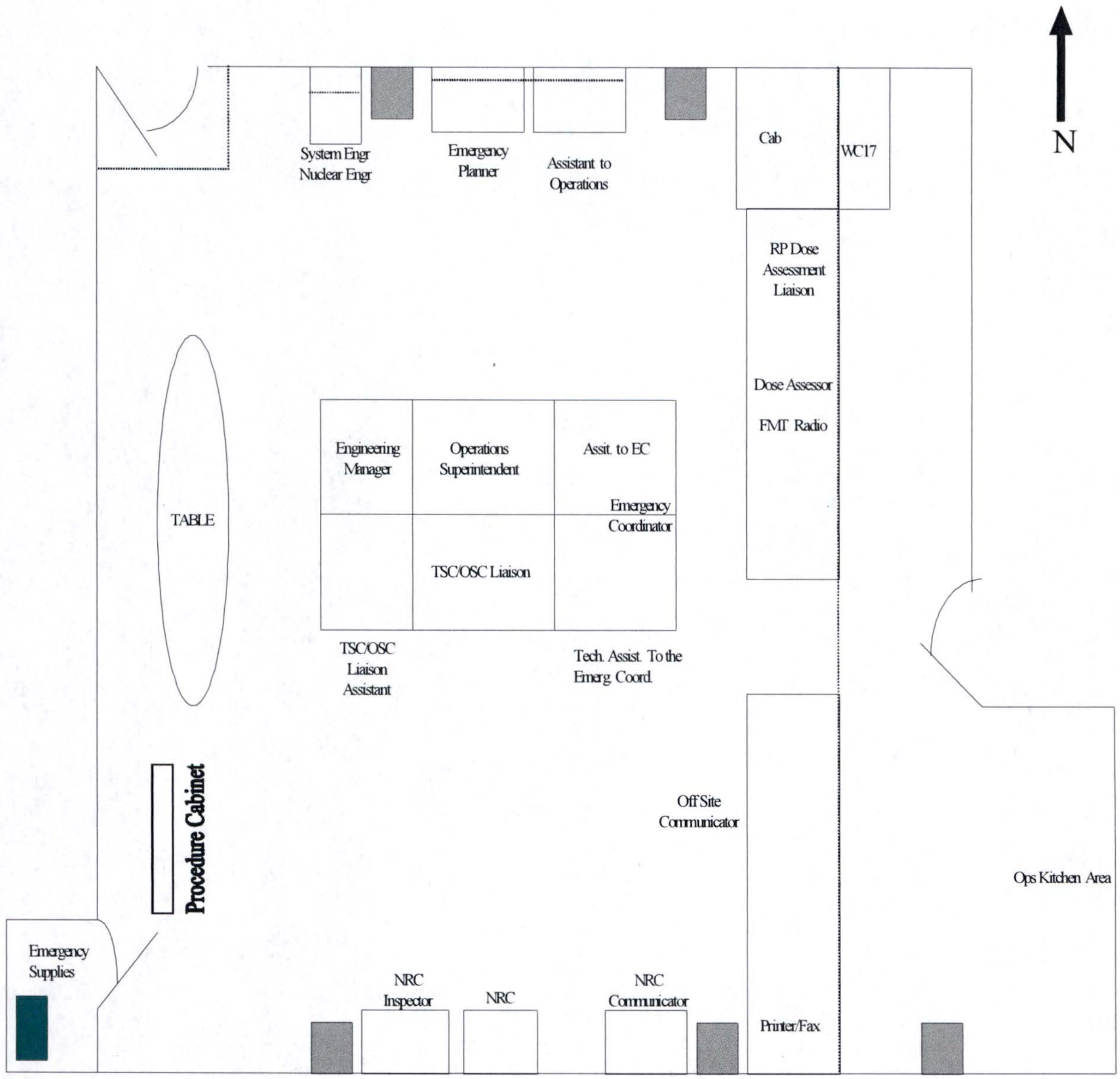
Protective Equipment Kits - Figures H-13, H-14, H-17
Communications Equipment - Figures H-12, H-16
Radiological Monitoring Equipment - Figures H-12, H-16, H-17
Emergency Supplies - Figures H-16, Figure H-17
Emergency Medical Supplies - L-1, L-2, L-3
Decontamination Supplies - K-3
Spill Cleanup Equipment/Supplies - H-19

H.12 Field Monitoring Data Collection

The Emergency Operations Facility has been designated as the central point for the receipt and analysis of all field monitoring data and coordination of sample media. The Radiological Assessment Manager at the Emergency Operations Facility will be responsible for the coordination efforts.

FIGURE H-1

OCONEE NUCLEAR STATION
TYPICAL TECHNICAL SUPPORT CENTER (TSC)
PRIMARY LOCATION
UNIT 1&2 OPS CENTER



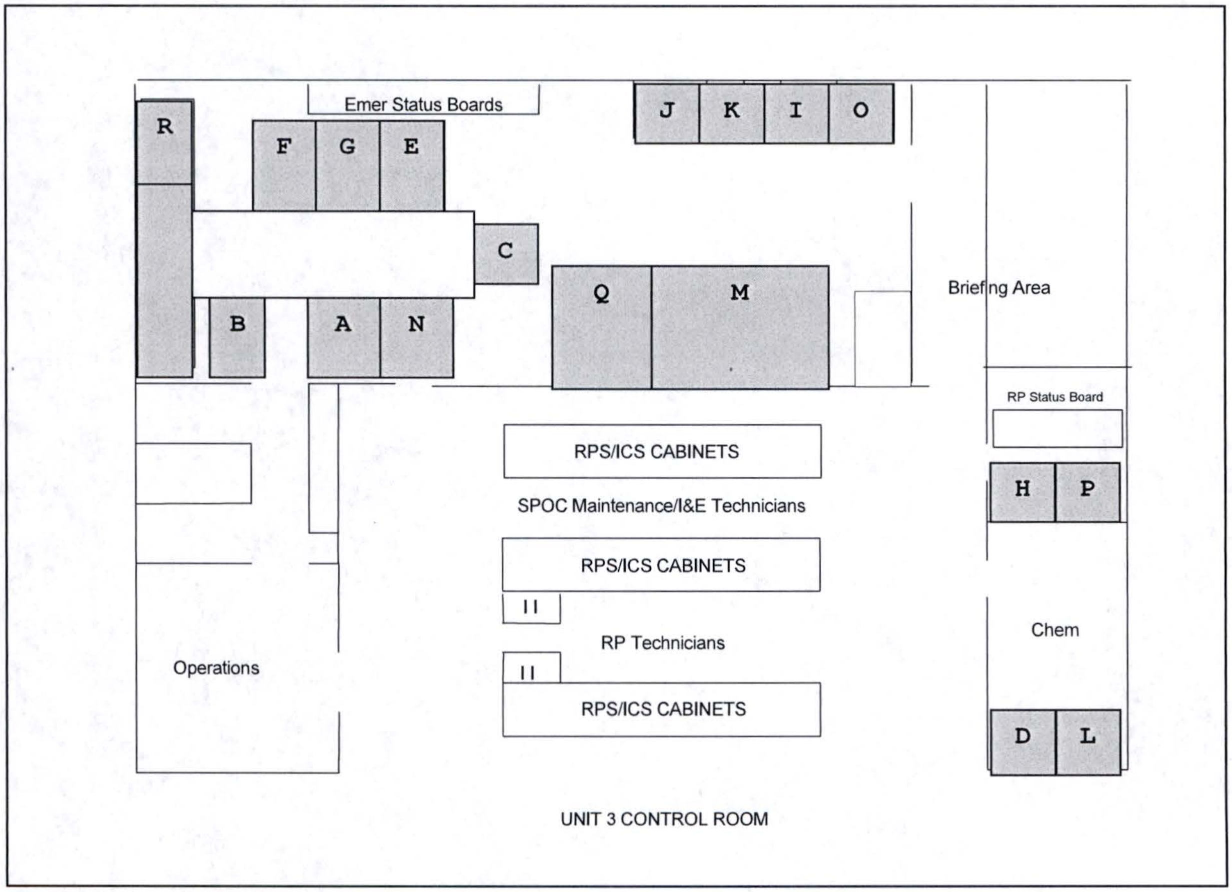
Rev. 2017-002
March 2017

FIGURE H-1A

NO LONGER USED

FIGURE H-2

OCONEE NUCLEAR STATION
TYPICAL OPERATIONAL SUPPORT CENTER (OSC)
PRIMARY LOCATION
UNIT 3 OPERATONS CENTER



A.	OSC Manager	J.	Electrical Engineering
B.	Ops Liaison	K.	Maintenance Supervisor (SPOC)
C.	RP Manager	L.	Chemistry Supervisor
D.	Chemistry Shift	M.	FIN24
E.	Technical Assistant I	N.	Technical Assistant II
F.	Chemistry Manager	O.	Nuclear Supply Chain Liaison
G.	Maintenance Manager	P.	RP Shift
H.	RP Supv	Q.	Assistant to RP Mgr.
I.	SPOC/FIN24	R.	Security Liaison

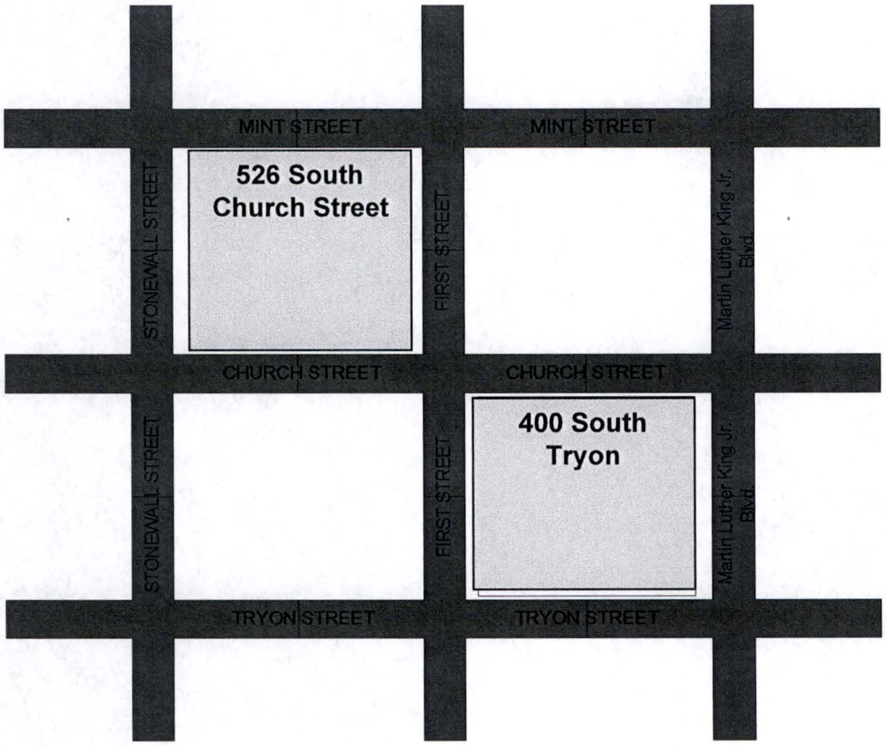
FIGURE H-2A

NO LONGER USED

FIGURE H-3A

DUKE ENERGY
OCONEE NUCLEAR STATION

CHARLOTTE EOF
GENERAL OFFICE BUILDING LAYOUT – CHARLOTTE, NC



The EOF is on the 3rd Floor of the Energy Center.

FIGURE H-3B
DUKE ENERGY
OCONEE NUCLEAR STATION
CHARLOTTE EMERGENCY OPERATIONS FACILITY LAYOUT

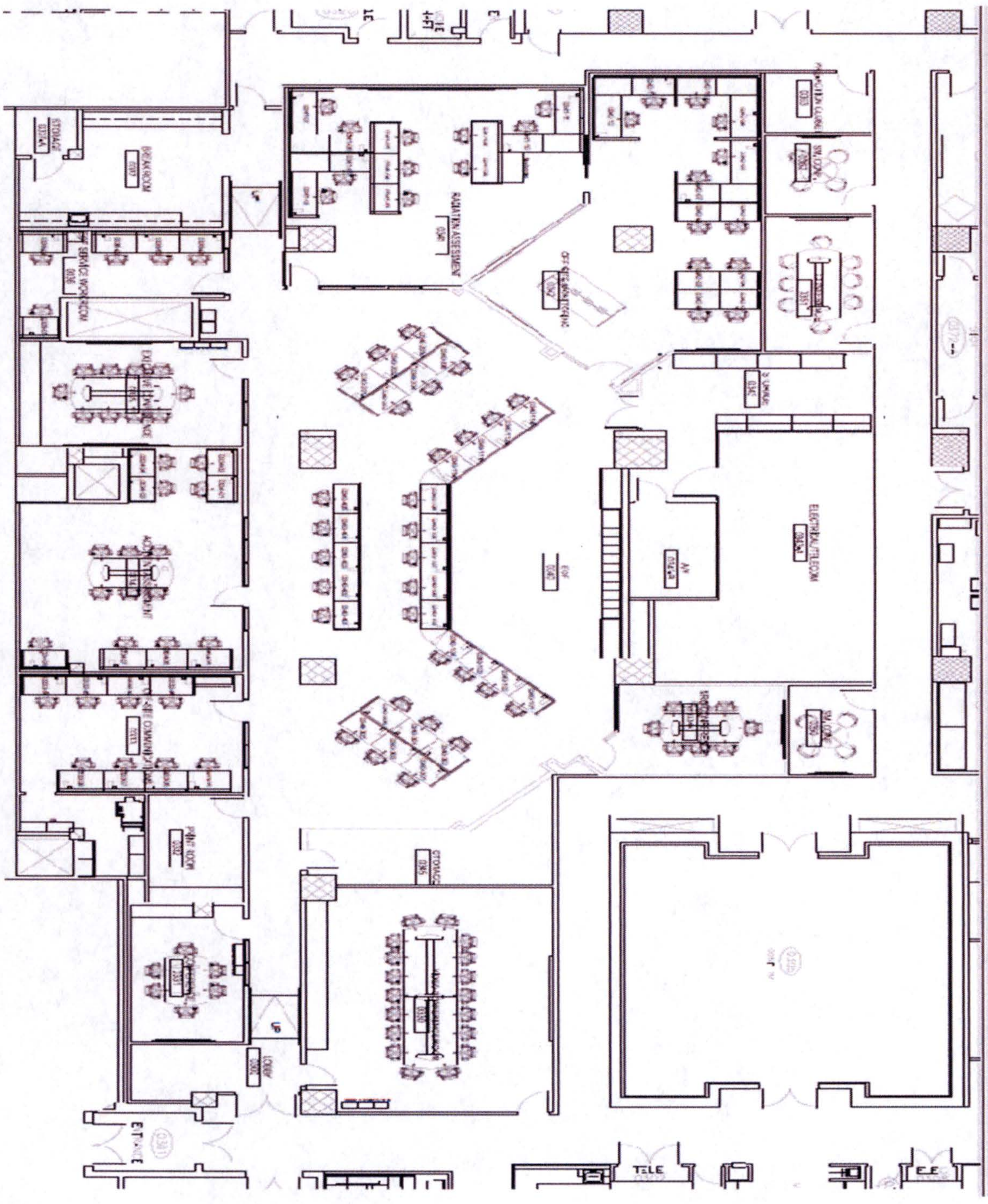


FIGURE H-3C

DUKE ENERGY
OCONEE NUCLEAR STATION

TYPICAL OCONEE JIC SET UP
(Alternate Emergency Response Facility)

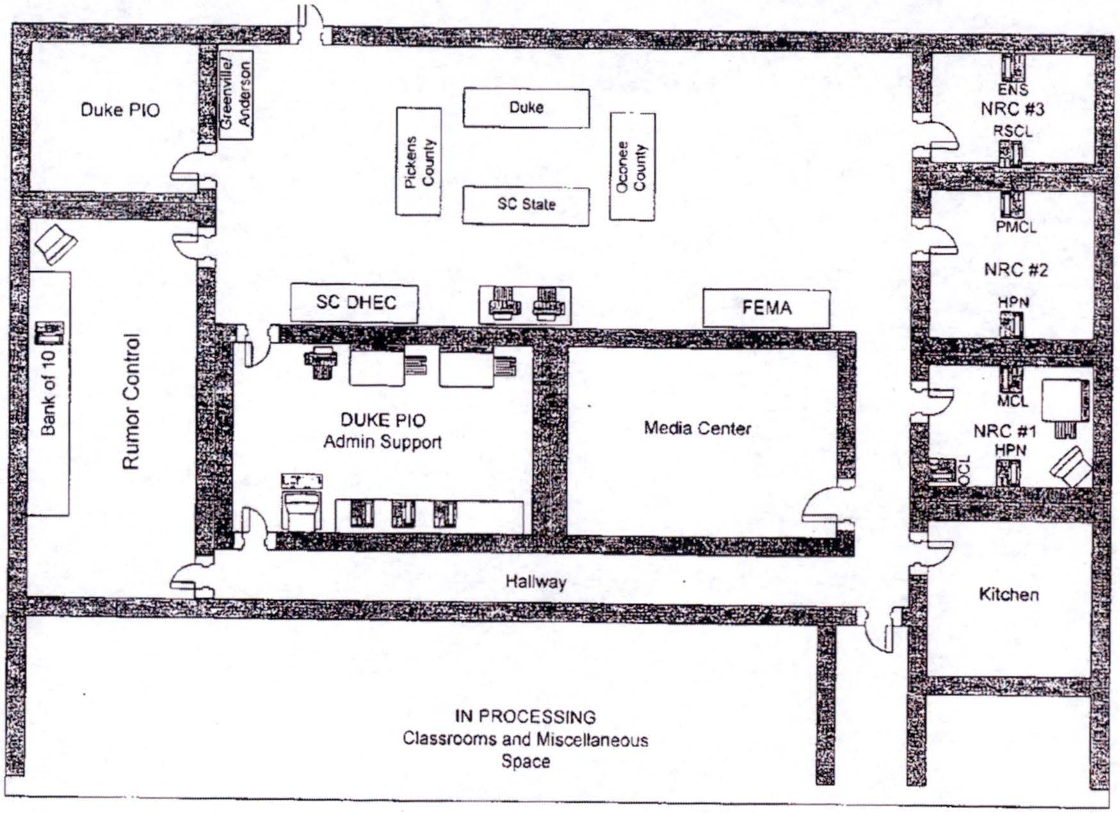


FIGURE H-3D

DUKE ENERGY
OCONEE NUCLEAR STATION

OCONEE MEDIA CENTER

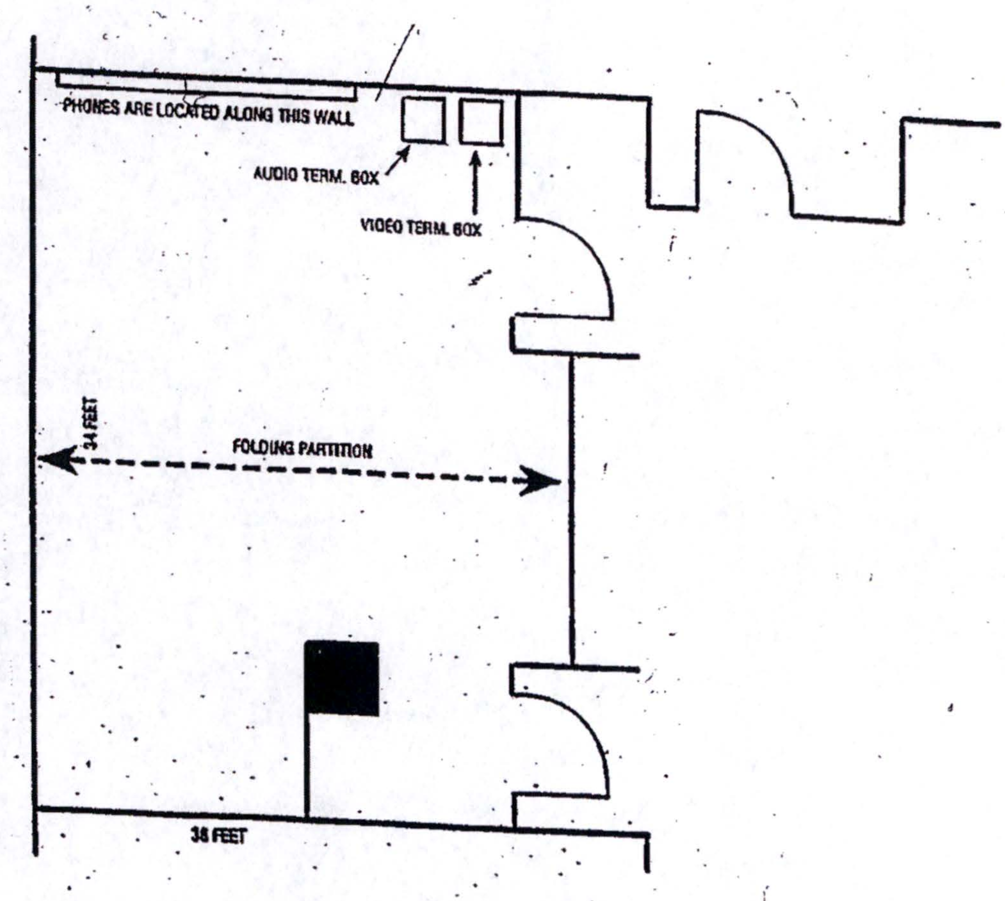


FIGURE H-3E

DUKE ENERGY
OCONEE NUCLEAR STATION

OCONEE JIC GENERAL LAYOUT

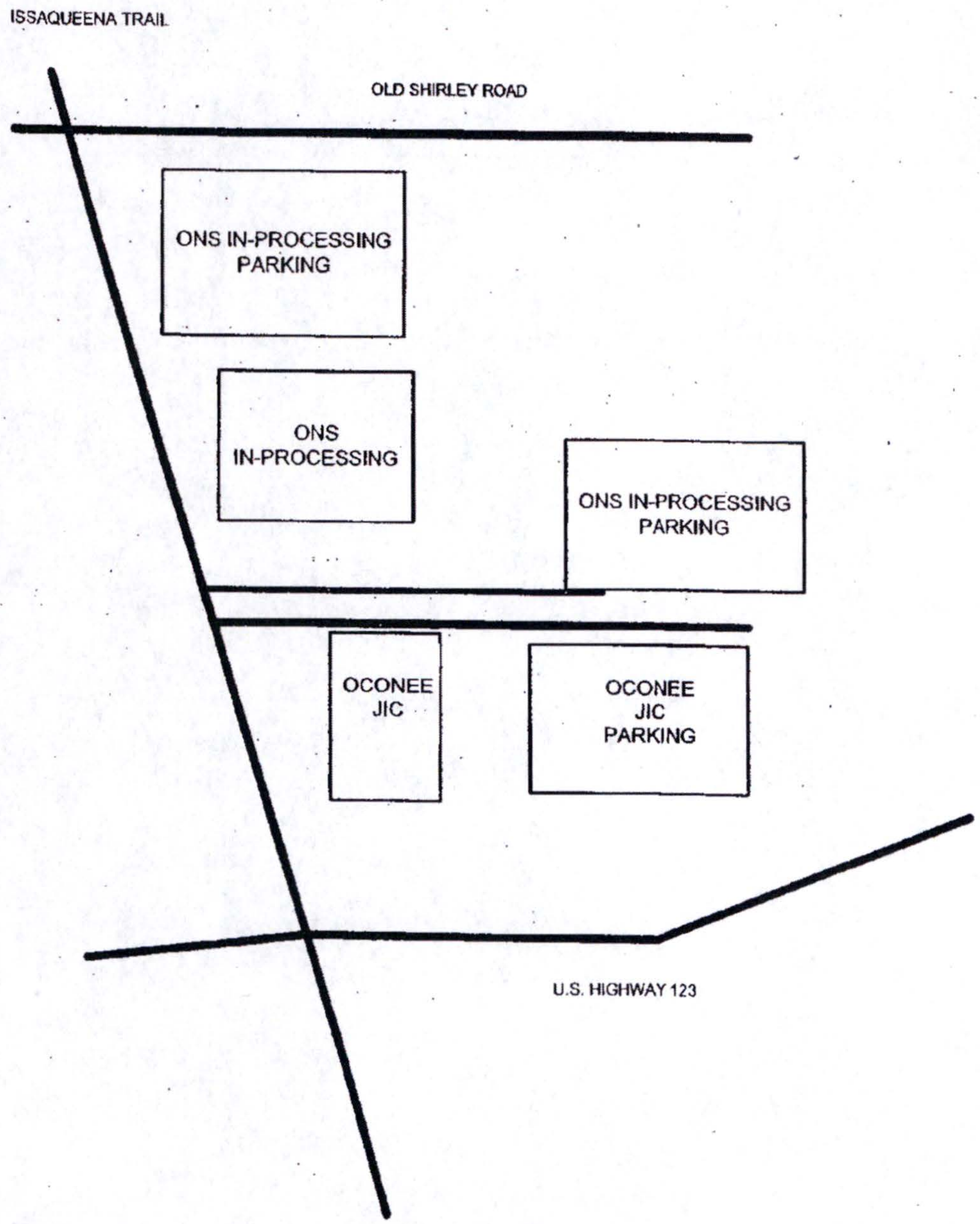


FIGURE H-3F

DUKE ENERGY
OCONEE NUCLEAR STATION

OCONEE BACKUP COUNT ROOM LOCATION
ONS ADMIN BUILDING

FIGURE H-4

DUKE ENERGY
OCONEE NUCLEAR STATION

METEOROLOGY EQUIPMENT

Wind Speed Monitoring Systems

Wind Direction Monitoring Systems

Platinum (RTD) T Delta or T/ΔT Monitoring System

Precipitation Monitoring System

NOTE: The Meteorological Monitoring System monitors and records continuous data for upper and lower levels of wind speed and direction, ambient air temperature and temperature differential at Site #1 (Northwest Met Tower). Wind speed, wind direction and precipitation is recorded at Site #2 (Keowee River Tower). All data points are included on each of the Units OAC computers where the data is averaged over a 15 minute period of time, except for precipitation.

IP/0/B/1601/003 (Meteorological Equipment Checks) gives range, accuracy and location.

FIGURE H-5

DUKE ENERGY
OCONEE NUCLEAR STATION

RADIATION INDICATING ALARMS (RIA)

RIA#	UNIT#	TYPE	RANGE	FUNCTION	LOCATION	CLASS
1	1,3	GM	.1 -1E4mRad/hr	Control Room	Control Room	Area
PAM*	1,2,3	GM	.1 -1E4mRad/hr	Main Bridge	Reactor Building	Area
PAM*	1,2,3	GM	.1 -1E4mRad/hr	Aux. Bridge	Reactor Building	Area
3	1,2,3	GM, IC	.1 -1E7mRad/hr	Refuel Canal	Transfer Canal	Area
4	1,2,3	GM, IC	.1 -1E7mRad/hr	RB Entrance	Personnel Hatch	Area
5	1,2,3	GM	.1 -1E4mRad/hr	Incore Tank	Outside Incore Tk Hatch	Area
PAM*	1,3	GM	.1 -1E4mRad/hr	Spent Fuel	SF Bridge	Area
6	1,3	GM, IC	.1 -1E7mRad/hr	Spent Fuel Area/Pool	SF Pool Area	Area
7	1	GM	.1 -1E4mRad/hr	Hot Machine Shop	East Wall	Area
8	1	GM	.1 -1E4mRad/hr	Hot Lab/ Chemistry	Hot Chem. Lab	Area
10	1,3	GM	.1 -1E4mRad/hr	Sample Hood/Primary	Primary Sample hood	Area
11	1,3	GM	.1 -1E4mRad/hr	Corridor 796'(3rd Level)	Unit 1/2 Change Room, Unit 3 Change Room	Area
12	1,3	GM	.1 -1E4mRad/hr	Chem Addition	Unit 1/2/3 Mix Tank	Area
13	1,3	GM	.1 -1E4mRad/hr	Waste Disposal Sink	Waste disposal Tk	Area
15	1,3	GM, IC	.1 -1E7mRad/hr	HPI	HPI Rooms	Area
16	1,3	GM, IC	.01 -1E7mRad/hr	"A" Main Steam Line	"A" Main Steam Lines	Area
17	1,3	GM, IC	.01 -1E7mRad/hr	"B" Main Steam Line	"B" Main Steam Lines	Area
31	1	NaI	10 -1E7cpm	LPI cooler LPSW Discharge	Turbine Building Basement	Effl

Figure H-5
DUKE ENERGY
OCONEE NUCLEAR STATION

RADIATION INDICATING ALARMS (RIA)

RIA#	UNIT#	TYPE	RANGE	FUNCTION	LOCATION	CLASS
32	1	P.Beta	10 - 1E7cpm	Aux. Bldg. Gas	AB-1 SF Resin Tank	Area
32	3	P.Beta	10 - 1E7cpm	Aux. Bldg. Gas	AB-2 Elevator Lobby	Area
33		NaI	10 - 1E7cpm	Normal LWD	Radwaste Facility	Effl
35	1,2,3	NaI	10 - 1E7cpm	LPSW Disch. Aux Building	Turbine Building Basement	Effl
37	1,3	P.Beta	10 - 1E7cpm	Normal GWD	Purge Equipment or Pen Room near elevator	Effl
38	1,3	GM	10 - 1E7cpm	High GWD	Purge Equipment or Pen Room near elevator	Effl
39	1,3	P.Beta	10 - 1E7cpm	CR-Gas	6th Fl. behind Em. Air Booster Pumps	Area
40	1,2,3	P.Beta	10 - 1E7cpm	Air ejector off gas	Purge Equip. room	Effl
41	1,3	P.Beta	10 - 1E7cpm	SF Bldg. Gas	Purge Equip. room	Area
42	1,3	NaI	10 - 1E7cpm	RCW return	Behind backwash pumps	Sys
43	1,2,3	P.Beta	10 - 1E7cpm	Unit vent particulates	Purge Equip. room	Effl
44	1,2,3	NaI	10 - 1E7cpm	Unit vent iodine	Purge Equip. room	Effl
45	1,2,3	P.Beta	10 - 1E7cpm	Unit vent gas normal	Purge Equip. room	Effl
46	1,2,3	CdTe	10 - 1E7cpm	Unit vent gas high	Purge Equip. room	Effl
47	1,2,3	P.Beta	10 - 1E7cpm	RB particulate	Purge Equip. room	Effl
48	1,2,3	NaI	10 - 1E7cpm	RB iodine	Purge Equip. room	Effl
49	1,2,3	P.Beta	10 - 1E7cpm	RB gas normal	Purge Equip. room	Effl
49A	1,2,3	CdTe	10 - 1E7cpm	RB gas high	Purge Equip. room	Effl
50	1,2,3	NaI	10 - 1E7cpm	Component Cooling	AB-1	Sys

Figure H-5
DUKE ENERGY
OCONEE NUCLEAR STATION

RADIATION INDICATING ALARMS (RIA)

RIA#	UNIT#	TYPE	RANGE	FUNCTION	LOCATION	CLASS
53	IB	P.Beta	10 - 1E7cpm	Interim Bldg. Gas	Interim Bldg.	Effl
54	1,3	NaI	10-1E7cpm	TB Sump	TB Basement	Effl
56	1,2,3	IC	1-1E8Rad/hr	Vent Stack Effluent	Vent Stack (Midway)	Effl
57	1,2,3	IC	1 -1E8Rad/hr	Containment High range monitor	Reactor Bldg. Penetration	Area
58	1,2,3	IC	1 -1E8Rad/hr	Containment High range monitor	Reactor Bldg. Penetration	Area

GM = Geiger Mueller

IC = Ion Chamber

PAM = Portable Area Monitor
* Portable area monitors do not have assigned RIA numbers and are local readout only.

IB = Interim Building

FIGURE H-6

DUKE ENERGY
OCONEE NUCLEAR STATION

PORTABLE SURVEY INSTRUMENTS

INSTRUMENT TYPE	RESPONSE TIME	DETECTOR TYPE	RANGES	RADIATION DETECTED	TUBE SATURATION	ADDITIONAL INFORMATION
Ludlum 3	4-22 seconds	Halogen quenched GM	X0.1 = 0-0.2 mR/hr X1.0 = 0-2.0 mR/hr X10 = 0-20 mR/hr X100 = 0-200 mR/hr	Beta & Gamma	Indicates offscale	Typically 1200 cpm per mR/hr. Speaker indication. Contains battery check position.
Eberline RM14 or Ludlum 177	2.2 - 22 seconds variable	Halogen quenched GM	X1=0-500 cpm X10=0-5000 cpm X100=0-50000 cpm	Beta & Gamma	Indicates offscale	Has alarm setting. Speaker indication. 50 hr operation on fully charged battery. Ludlum 177 has additional scale, x 1000 = 0-500 kcpm.
MGPI Telepole	2-30 seconds variable	Two GM tubes 1 low range 1 high range	0.05 mR/hr - 1000 R/hr	Gamma	Indicates over range	Automatic switching between GM tubes. 11' extension probe. Battery self check.
Eberline RO20 or Ludlum 9-3	5 seconds	Ion-chamber Air filled. Vented to atmosphere	0-50Rad/hr.	Beta & Gamma	Indicates offscale	Has battery check information
Eberline RO7 or Ludlum 9-7	Variable	Air filled ion chamber	Med range: 0.1-199.9 Rad/hr Hi range: 0-.01 - 19,900 Rad/hr	Beta & Gamma	Indicates over range	Digital ion chamber with cables to extend detection up to 60' away or under water.

FIGURE H-6

DUKE ENERGY
OCONEE NUCLEAR STATION

PORTABLE SURVEY INSTRUMENTS

Instrument Type	Response Time	Detector Type	Ranges	Radiation Detected	Tube Saturation	Additional Information
Ludlum-12	4-22 seconds	Cadmium loaded polyethylene sphere with He tube in center. Tube operates in proportional region	0 - 100,000 mRad/hr	Neutron	Rejects Gamma up to 10 Rad/hr.	Detector can be attached or moved from meter.
AMP-100	Variable	Energy Compensated GM tube	0 - 1000 R/hr	Gamma	Over range alarm	Can be used with variable length of cable.
AMP-200	Variable	Energy Compensated GM tube	1 - 10,000 R/hr	Gamma	Over range alarm	Can be used with variable length of cable.
ESP 2	Variable	Sodium Iodide Scintillator	Variable	Gamma	Over range alarm	Single channel analyzer w/pulse height analysis Nal detectors

FIGURE H-7

DUKE ENERGY
OCONEE NUCLEAR STATION

AIR SAMPLERS

INSTRUMENT NAME	EXPECTED FLOW RATE	AIR PUMP TYPE	MAXIMUM LENGTH OF OPERATION
HD29A	2 CFM	Centrifugal Carbon Vane Pump air-cooled motor	Continuous, constant flow
H-809V	2 CFM	Two-stage turbine blower air-cooled motor	15 minutes
RAP-1	2 CFM	Oil Free, Carbon Vane	Continuous, constant flow

FIGURE H-8

DUKE ENERGY
OCONEE NUCLEAR STATION

FIRE AND COMBUSTION PRODUCTS AND DETECTORS

FIRE DETECTION SYSTEM - Inaccessible Detectors

The purpose of this fire detection system is to detect visible and/or invisible smoke or other products of combustion in any space covered by detectors.

The principal parts of this system; Fire indicating unit, zone indicating units and detectors, with up to 8 zone indicating units-for each fire Indicating unit. Up to 4 detectors circuits (zones) on each zone indicating unit. Each detector circuit (zone) has up to 12 detectors.

When products of combustion are detected a flashing lamp on the detector base is turned on. The zone lamp for the zone covering that detector will come on. The Red "Alarm" lamp on the fire indicating unit will come on. The statalarm in the control room will come on.

In the event of a failure in the system which makes the system inoperative, an amber "Trouble" lamp will come on, a buzzer will sound and the statalarm will come on.

FIRE DETECTION SYSTEM - Accessible Detectors

The purpose of this fire detection system is to detect visible and/or invisible smoke or other products of combustion in any space covered by detectors.

The principal parts of this system include; Fire indicating unit, Zone indication units and detectors, with up to 8 zone indicating units for each fire indicating unit. Up to 4 detector circuits (zones) are on each zone indicating unit. Each detector circuit (zone) has up to 99 detectors.

When products of combustion are detected a red "LED" on the Honeywell detector will come on. The zone lamp for that detector will come on. The Red "Alarm" lamp on the fire indicating unit will come on. The statalarm in the control room will come on.

In the event of a failure in the system which makes the system inoperative, an amber "Trouble" lamp on the Honeywell will come on, a buzzer will sound and the statalarm will come on.

FIGURE H-9

DUKE ENERGY
OCONEE NUCLEAR STATION

NORMAL ENVIRONMENTAL MONITORING PROGRAM

ONSITE/OFFSITE TLD LOCATIONS

See: Oconee Offsite Dose Calculation Manual

FIGURE H-10

DUKE ENERGY
OCONEE NUCLEAR STATION

NORMAL ENVIRONMENTAL MONITORING PROGRAM

AIR SAMPLE LOCATIONS

OFFSITE LOCATIONS

See: Oconee Offsite Dose Calculation Manual

FIGURE H-11

DUKE ENERGY
OCONEE NUCLEAR STATION

COUNT ROOM EQUIPMENT (ONSITE)

INSTRUMENT TYPE	DESCRIPTION
Gamma Spectroscopy System	Computer based gamma spectroscopy system with solid state germanium detectors for analysis of various sample media.
Body Burden Analyzer and Stand-up Total Body Analyzer	Computer based gamma spectroscopy system with three sodium detectors mounted in a shielded chair which can analyze the thyroid, lungs, and lower torso simultaneously, along with a stand-up total body analyzer using large sodium iodine detectors.
Automatic Smear Counter	An automatic smear counter using a GM detector which performs beta only analyses on up to 50 smears.
Liquid Scintillator	Multiple sample liquid scintillation analysis systems that detect and quantify H-3 and gross beta using a computer to correct for quench and activity.
Alpha Scintillator	An automatic smear counter using a zinc sulfide scintillator detector to detect alpha only. Analyzes up to 50 smears/air samples at a time.

FIGURE H-12

DUKE ENERGY
OCONEE NUCLEAR STATION

CONTENTS OF EMERGENCY KITS FOR FIELD MONITORING TEAMS

(Location World of Energy)

SEE HP/0/B/1009/001

FIGURE H-13

DUKE ENERGY
OCONEE NUCLEAR STATION

EMERGENCY KIT INVENTORY SHEET

Control Room Locations

See HP/0/B/1009/001

FIGURE H-14

DUKE ENERGY
OCONEE NUCLEAR STATION

EMERGENCY KIT INVENTORY SHEET

Respiratory Equipment

See HP/0/B/1009/001

FIGURE H-15

DUKE ENERGY
OCONEE NUCLEAR STATION

EMERGENCY SUPPLIES INVENTORY LIST

Technical Support Center

Operational Support Center

Emergency Operation Facility

See PT/0/A/2000/008 and ST/0/A/4600/086

FIGURE H-16

DUKE ENERGY
OCONEE NUCLEAR STATION

EMERGENCY CABINET INVENTORY SHEET

INPLANT SURVEILLANCE EQUIPMENT

(WORLD OF ENERGY)

SEE HP/0/B/1009/001

FIGURE H-17

DUKE ENERGY
OCONEE NUCLEAR STATION

INVENTORY LIST FOR OPERATIONAL SUPPORT CENTER

EMERGENCY CABINET

See HP/0/B/1009/001

FIGURE H-18

DUKE ENERGY
OCONEE NUCLEAR STATION

SEISMIC INSTRUMENTATION PROGRAM

SEISMIC EQUIPMENT	UNIT 1 CABLE ROOM	UNIT 1 TENDON ACCESS GALLERY	UNIT 1 REACTOR BUILDING
Seismic Trigger (1) (Setpoint .05g and actuates a statalarm and computer alarm in Control Room 1 & 3. Also actuates Unit 1 & 2 Events Recorder.		x	
STRONG-MOTION ACCELEROGRAPH SYSTEM. Starter (1) Setpoint .01g for 1 sec will actuate accelerometers and recorders on Control Panel. Also actuates a computer alarm in Control Room 1.		x	
Accelerometers (2) Actuates recorders on Control Panel at .01g for 1 sec		x	x
Recorders (2) Records for 10 additional sec following completion of seismic events up to 30 minutes	x		
Control Panel (1) Event alarm-alarm light turns yellow to indicate system is recording approximately 10 sec. Event Indicator-normally black but after an event is recorded, it is white	x		
PEAK ACCELERATION RECORDER (6) Records the peak acceleration experienced. Capability to measure up to 2g. Uses no power supply.		x	x

FIGURE H-19

DUKE ENERGY
OCONEE NUCLEAR STATION

SPILL CONTROL EQUIPMENT/SUPPLIES

SEE THE FOLLOWING PROCEDURES/DOCUMENTS:

Emergency Planning:

PT/0/B/0250/030
PT/0/B/0250/045
ONS Prefire Plan

Chemistry:

CP/0/B/2001/008

Safety Assurance:

Spill Prevention Control Countermeasures Plan (SPCC)

FIGURE H-20

DUKE ENERGY
OCONEE NUCLEAR STATION

SURVEYS

Emergency	Control Room Instrumentation	In Station Radiological	Site and Site Boundary	Environs
Unusual	X	X	*	*
Alert	X	X	X	*
Site Area	X	X	X	X
General	X	X	X	X

* Conducted in the event effluent technical specifications are exceeded.

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/A/1000/001) identifies plant parameters that can be used to determine emergency situations that require activation of the station emergency plan. Revision 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 (ADAMS Accession Number ML12326A805). ONS conducted an EAL implementation upgrade project that produced the EALs, 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 the EAL Wallchart of RP/0/A/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 AD-EP-ALL-0202. AD-EP-ALL-0202 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 AD-EP-ALL-0202 determines the magnitude of the release of radioactive materials based on plant system parameters and effluent monitors (vent release).

I.4 Dose Calculation Methodology

AD-EP-ALL-0202 establishes the relationship between effluent monitor readings or reactor building dose rate readings and onsite/offsite doses for various meteorological conditions.

AD-EP-ALL-0202 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 ENS 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

AD-EP-ALL-0202 is the procedure 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 AD-EP-ALL-0203 and SH/0/B/2005/003 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:

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

$$\frac{dN_t^f}{dt} = \dot{F}\bar{Y} - N_t^f \lambda - N_t^f \upsilon \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots(1)$$

Where N_t^f 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)

$\bar{y}$ 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:

$$N_t^f = {}^f_o e^{-(\lambda+\upsilon)t} + \frac{\dot{F}\bar{y}}{(\lambda+\upsilon)} = (1 - e^{-(\lambda+\upsilon)t})$$

II. <u>In-Coolant Concentration</u>

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

$$\frac{dN_t^c}{dt} = N_t^f(\upsilon) - N_t^c(\lambda) - N_t^c K = \frac{dN_t^c}{dt} = \text{(Release Rate) - (Decay Rate) - (Purification Rate)} \quad (3)$$

where N_t^c 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:

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

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

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

or $N_t^c = N_t^{c1} + N_t^{c2} + N_t^{c3}$ (4)

In the above:

1) N_t^{c1} 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_t^{c2} 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:

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

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:

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

$\frac{\upsilon 133}{\upsilon 133} = a$

Substituting and solving for υ_{131} gives:

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

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:

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

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

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 $\upsilon_{131} / \upsilon_{133} = a$) was assumed to be function of the normalized iodine ratio. A curve was fit to the data resulting in:

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

<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:

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

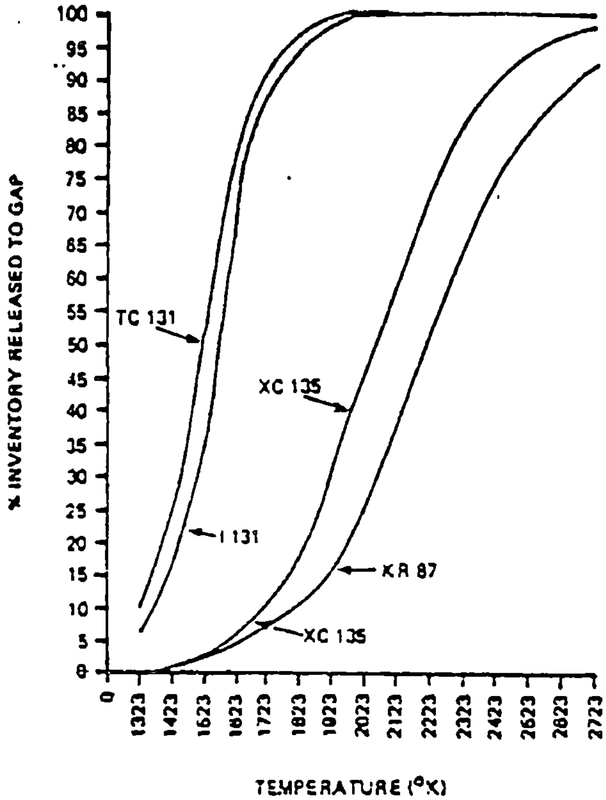
OSC-200 - Water Depth in Reactor Building

Information derived from the above calculations are used in RP/0/A/1000/018 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

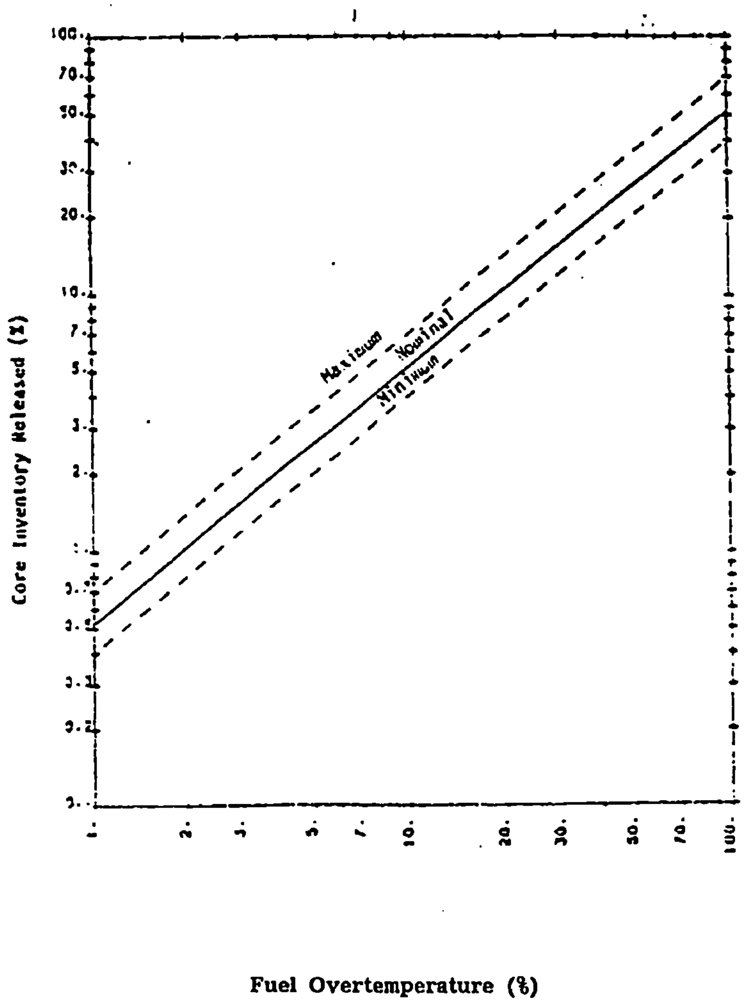


FIGURE I-4

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

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

C. General Equations for Barium

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

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

D. General Equations for Praseodymium

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

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

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

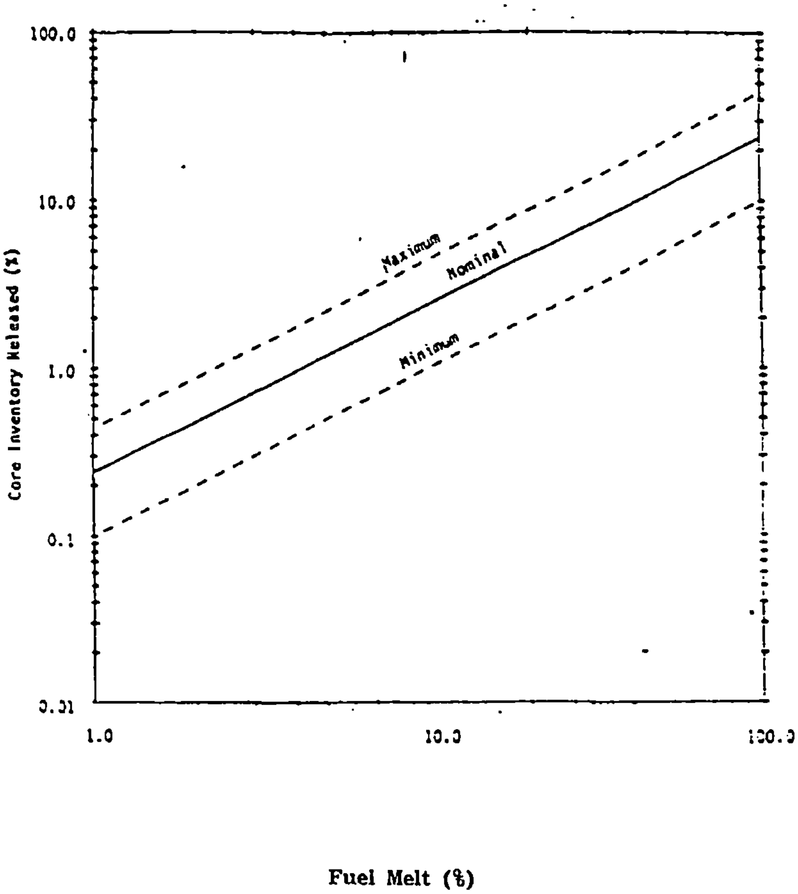


FIGURE I-4C

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

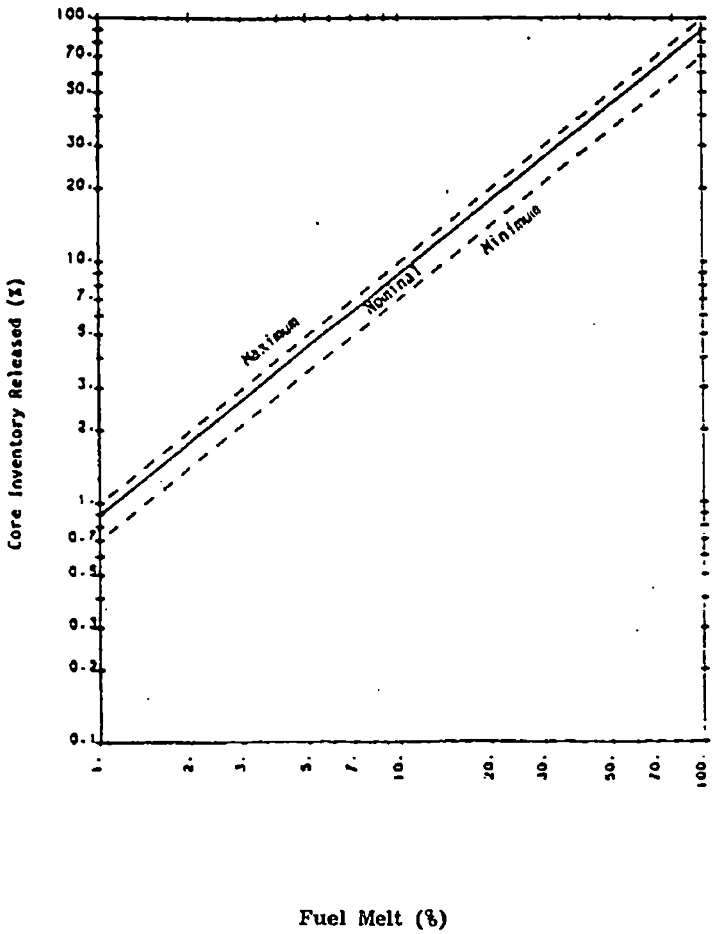


FIGURE I-5

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)		

$$*\frac{LOR2 - FSAR}{LOR2} \text{ x } 100$$

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

$E\gamma$ = 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:

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

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 x 10^5

FIGURE I-6C

DOSE RATE VS TIME FOR FUEL OVERHEATING WITHOUT FUEL MELT

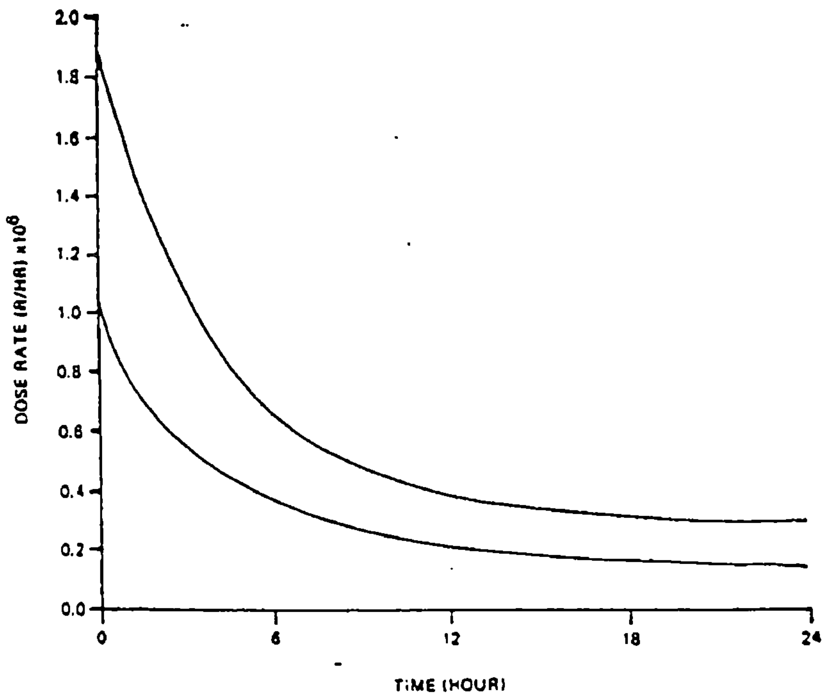


FIGURE I-7

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:

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

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

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

Y = Power correction factor= $\frac{Average\ Power\ for\ Prior\ 30\ days}{Rated\ power\ level}$

F m = Fuel failure percent

FIGURE I-7A

DOSE RATE VS. TIME FOR FUEL MELT

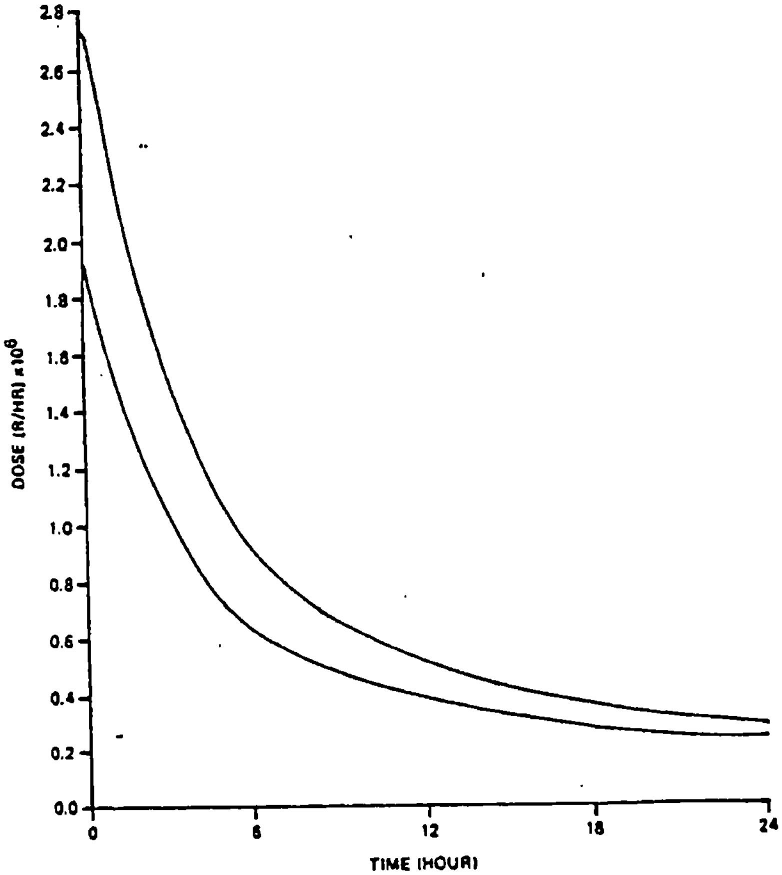


FIGURE I-8

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 = $\frac{V_{STP}}{Volume\ of\ one}\ mole$

$$M_H = \frac{P_C}{T_C} \quad X_{H_2} \quad \frac{6.317x10^7}{0.79} = \frac{P_C}{T_C} \quad X_{H_2} \quad (7.7616 \text{ x } 10^7)$$

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