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Director
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Southern Nuclear
Operating Company, Inc.
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Docket Nos.: 52-025
52-026

ND-16-0150
10 CFR 50.90
10 CFR 50, Appendix E

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Southern Nuclear Operating Company
Vogtle Electric Generating Plant Units 3 and 4
Request for License Amendment:
Vogtle 3 and 4 Proposed Emergency Action Levels (LAR-16-002)

Ladies and Gentlemen:

Pursuant to 10 CFR Part 50, Appendix E and in accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC), the licensee for Vogtle Electric Generating Plant (VEGP) Units 3 and 4, requests an amendment to Combined License (COL) Numbers NPF-91 and NPF-92, for VEGP Units 3 and 4, respectively. In accordance with 10 CFR 50, Appendix E, Section IV, Item B, this amendment proposes new plant-specific Emergency Action Levels (EALs) for VEGP Units 3 and 4 for review and approval.

In addition, this License Amendment Request (LAR) is being submitted to request a change to License Condition 2.D(12)(d) for VEGP Units 3 and 4. This license condition requires that no later than 180 days before initial fuel load for each unit, SNC submit in writing to the Director of NRO (or designee) a fully developed set of plant-specific EALs in accordance with Nuclear Energy Institute (NEI) 07-01, "Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors," Revision 0, with no deviations. This LAR proposes that the license conditions be modified to allow SNC to submit plant-specific EALs developed using the criteria as described in Enclosure 1.

The Description, Technical Evaluation, Regulatory Evaluation (including Significant Hazards Consideration Determination), and Environmental Considerations for the proposed VEGP Units 3 and 4 EALs and changes to License Condition 2.D(12)(d) for each unit are contained in Enclosure 1 of this letter. Enclosure 2 provides the Technical Bases Document, which includes the plant-specific EALs and an explanation and rationale for each EAL proposed in the LAR. When approved, the EALs will be incorporated into two EAL Classification Matrices. These matrices will be based on HOT operating conditions and COLD operating conditions. The proposed EAL Classification Matrices are provided for information only in Enclosure 3. Enclosure 4 contains the VEGP Units 3 and 4 License Condition 2.D(12)(d) change related to the proposed EALs. Enclosure 5 provides an information only matrix correlating the proposed VEGP Units 3 and 4 EALs to the guidance in NEI 07-01, Revision 0 and NEI 99-01, Revision 6.

Enclosure 6 contains the results of design reviews (based on AP1000 DCD Rev. 19) performed by Westinghouse on the EAL Technical Bases Document (Enclosure 2).

The changes proposed in this LAR are consistent with the technical content in the South Carolina Electric and Gas (SCE&G) December 1, 2015, submittal of License Amendment Request LAR-14-13R for the Virgil C. Summer Nuclear Station, Units 2 and 3 [ML15335A448].

This letter contains no regulatory commitments.

SNC requests staff approval of the license amendment by November 18, 2016 to support operator training, emergency plan implementing procedure development, and exercise development. SNC expects to implement the proposed amendment within 30 days of approval of the proposed changes.

In accordance with 10 CFR 50.91, SNC is notifying the State of Georgia of this LAR by transmitting a copy of this letter and enclosures to the designated State Official.

Should you have any questions, please contact Ms. Kelli A. Roberts at (205) 992-6831.

(Affirmation and signature are provided on the following page)

Mr. Brian H. Whitley states that: he is the Regulatory Affairs Director of Southern Nuclear Operating Company; he is authorized to execute this oath on behalf of Southern Nuclear Operating Company; and to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

B. H. Whitley

B. H. Whitley

BHW/TEA/kms

Sworn to and subscribed before me this 4th *day of* March, *2016*

Notary Public: [signature]

My commission expires: 12/1/2019

Enclosures:
1) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 – Request for License Amendment: Vogtle 3 and 4 Proposed Emergency Action Levels (LAR-16-002)
2) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 – Technical Bases Document for the Proposed Emergency Action Levels (LAR-16-002)
3) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 – EAL Classification Matrices for the Proposed Emergency Action Levels – For Information Only (LAR-16-002)
4) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 – Proposed License Condition Changes for Combined License Numbers NPF-91 and NPF-92 (LAR-16-002)
5) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 – Correlation Matrix for the Proposed EALs – For Information Only (LAR-16-002)
6) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 – Westinghouse Review Results for the Proposed EALs (LAR-16-002)

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Southern Nuclear Operating Company

ND-16-0150

Enclosure 1

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Request for License Amendment:

Vogtle 3 and 4 Proposed Emergency Action Levels

(LAR-16-002)

(Enclosure 1 consists of 11 pages, including this cover page)

Table of Contents

1. SUMMARY DESCRIPTION

In accordance with the provisions of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) requests an amendment to the Vogtle Electric Generating Plant (VEGP) Units 3 and 4 Combined Licenses (COLs) (License Nos. NPF-91 and NPF-92, respectively). In accordance with 10 CFR 50, Appendix E, Section IV, Item B, this amendment request proposes new emergency action levels (EALs) for VEGP Units 3 and 4 for review and approval. These proposed new EALs require NRC approval prior to implementation.

In addition, this License Amendment Request (LAR) is being submitted to request a change to License Conditions 2.D(12)(d) for VEGP Units 3 and 4. These license conditions require no later than 180 days before initial fuel load for each unit that SNC submit in writing to the Director of NRO (or designee) a fully developed set of plant-specific EALs in accordance with Nuclear Energy Institute (NEI) 07-01, "Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors," Revision 0, with no deviations. This LAR proposes that the license conditions be modified to allow SNC to develop plant-specific EALs using criteria from NEI 07-01 with consideration of design improvements as reflected in the AP1000 Design Control Document (DCD), Revision 19.

2. DETAILED DESCRIPTION AND TECHNICAL EVALUATION

EALs are the plant-specific indications, conditions, or instrument readings that are utilized to classify emergency conditions defined in the VEGP Units 3 and 4 Emergency Plan. The EAL Technical Bases Document (Enclosure 2) provides the plant-specific EALs and an explanation and rationale for each EAL included in this LAR. In the limited cases where design development information or plant-specific information/procedures were not available at the time this LAR was submitted, the bases document is highlighted accordingly and reference is made to developer notes that are included in Attachment 5 of Enclosure 2. These developer notes provide the methodology that will be utilized to concisely develop the remaining plant-specific information as necessary to support EAL evaluations. The new VEGP Units 3 and 4 plant-specific EALs proposed in this LAR are based on the following:

1. NEI 07-01, "Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors," Revision 0, July 2009 [ML092030210]

   NEI 07-01, Revision 0, states, "NEI 07-01, Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors, is based on the EAL work accomplished through the NUMARC/NESP 007, NEI 99-01 Revision 4 and Revision 5 development process."

   "Nuclear utilities must respond to a formal set of threshold conditions that require plant personnel to take specific actions with regard to notifying state and local governments and the public when certain off-normal indicators or events are recognized. Emergency classification levels are defined in 10 CFR 50. Levels of response and the conditions leading to those responses are defined in joint NRC/FEMA guidelines contained in Appendix 1 of NUREG-0654/FEMA-REP-1, Rev. 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," November 1980."

   "The EAL Task Force developed a systematic approach and supporting basis for EAL development. This methodology developed a set of generic EAL guidelines, together with the basis for each, such that they could be used and adapted by each utility on a consistent basis. The review of the industry's experiences with EALs, in conjunction with regulatory considerations, was applied directly to the development of this generic set of EAL guidelines. The generic guidelines were intended to clearly define conditions that represent increasing risk to the public and can give consistent classifications when applied at different sites. The guidance presented here is not intended to be applied to plants as-is. It is intended to give the user the logic for developing site specific EALs (i.e., instrument readings, etc.) using site specific EAL presentation methods (formats). Basis information is provided to aid station personnel in preparation of their own site specific EALs, to provide necessary information for training, and for explanation to state and local officials. In addition, state and local requirements have not been reflected in the generic guidance and should be considered on a case-by-case basis with appropriate state and local emergency response organizations."

See Enclosure 5 for an information only matrix correlating the proposed VEGP Units 3 and 4 EALs to the guidance in NEI 07-01, Rev 0.

2. Updates related to AP1000 design improvements

By letter dated March 28, 2008 [ML081050133], SNC submitted its application to the NRC for COLs for two AP1000 advanced passive pressurized-water reactors. This initial application incorporated by reference 10 CFR Part 52, Appendix D, "Design Certification Rule for the AP1000 Design," and the Westinghouse application for amendment of the AP1000 design as described in Revision 16 of the Design Control Document (DCD) (submitted May 26, 2007) [ML071580630]. Subsequent to the initial application, SNC provided submittal 8 of the application in a letter dated June 24, 2011 [ML11180A086]. Submittal 8 included Revision 5 of the Final Safety Analysis Report (FSAR) that incorporated by reference AP1000 DCD Revision 19 (submitted by Westinghouse on June 13, 2011 [ML11171A315]). In NUREG-1793 Supplement 2 [ML11293A120] (published September 2011), the NRC concluded that the changes to the DCD (up to and including Revision 19 to the AP1000 DCD) were acceptable. Design improvements that have been included in the DCD as a result of the transition from revision 16 to revision 19 have been factored into the Technical Bases Document (Enclosure 2).

3. Technical reviews by Westinghouse conducted in October 2014 and November 2015

In September 2014, South Carolina Electric and Gas (SCE&G) requested a technical review of the VC Summer Units 2/3 Emergency Action Level Technical Bases be performed by Westinghouse to identify any information in the document that was not consistent with the design basis of the Westinghouse AP1000 (DCD Rev. 19).

In October 2014, WEC provided editorial and technical comments to SCE&G and these comments were captured in a table entitled "Consolidated Comment Matrix from VCS, WEC, and Vogtle." The comments were later resolved through agreements between Westinghouse, SCE&G, and SNC and the EAL Technical Bases Document was updated accordingly.

Since that initial Westinghouse review, SCE&G and SNC performed an extensive final review of the EAL Technical Bases Document and as a result, additional changes were incorporated into the document. This combined final review was performed in support of

SCE&G's and SNC's intent to submit the same EAL Technical Bases Document (allowing for site specific differences) in LARs for Virgil C. Summer Nuclear Station (VCSNS) Units 2&3 and VEGP Units 3&4, respectively.

In November 2015, SCE&G and SNC requested that Westinghouse perform another technical review of the EAL Technical Bases Document (Enclosure 2) to obtain concurrence that the document was still consistent with the design basis of the Westinghouse AP1000 (DCD Rev. 19). The resulting Westinghouse letter provided concurrence and included a matrix (Enclosure 6) which identified: comments originally supplied by Westinghouse in their 2014 review that affected Westinghouse design related information; agreed upon resolutions to those comments; changes that had been added based on the SCE&G and SNC November 2015 final review; and agreed upon resolutions to these new changes.

The new VEGP Units 3 and 4 plant-specific EALs proposed in this LAR also take into consideration guidance provided in NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors," Revision 6, November 2012 [ML12326A805].

NEI 99-01, Revision 6 states, "The purpose of Nuclear Energy Institute (NEI) 99-01 is to provide guidance to nuclear power plant operators for the development of a site-specific emergency classification scheme. The methodology described in this document is consistent with Federal regulations, and related US Nuclear Regulatory Commission (NRC) requirements and guidance. In particular, this methodology has been endorsed by the NRC as an acceptable approach to meeting the requirements of 10 CFR §50.47(b)(4), related sections of 10 CFR 50, Appendix E, and the associated planning standard evaluation elements of NUREG-0654/ FEMA-REP-1, Rev. 1, *Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants*, November 1980."

"NEI 99-01 contains a set of generic Initiating Conditions (ICs), Emergency Action Levels (EALs) and fission product barrier status thresholds. It also includes supporting technical basis information, developer notes and recommended classification instructions for users. Users should implement ICs, EALs and thresholds that are as close as possible to the generic material presented in this document with allowance for changes necessary to address site-specific considerations such as plant design, location, terminology, etc."

"Properly implemented, the guidance in NEI 99-01 will yield a site-specific emergency classification scheme with clearly defined and readily observable EALs and thresholds. Other benefits include the development of a sound basis document, the adoption of industry-standard instructions for emergency classification (e.g., transient events, classification of multiple events, upgrading, downgrading, etc.), and incorporation of features to improve human performance. An emergency classification using this scheme will be appropriate to the risk posed to plant workers and the public, and should be the same as that made by another NEI 99-01 user plant in response to a similar event."

See Enclosure 5 for an information only matrix correlating the proposed VEGP Units 3 and 4 EALs to the guidance in NEI 99-01, Revision 6.

This amendment request also proposes changes to the VEGP Units 3 and 4 License Conditions 2.D(12)(d). These license conditions are a result of Request for Additional Information (RAI) 13.03-3 and following is a brief synopsis of its history.

NRC letter dated March 6, 2009 [ML090690215] states in RAI 13.03-3:

"The initial EALs, which are required by 10 CFR 50.47(b)(4) and Section IV.B of Appendix E to 10 CFR Part 50, must be approved by the NRC. The Vogtle combined license (COL) application does not fully address certain aspects of the required EAL scheme. This is because various equipment set points and other information cannot be determined until the as-built information is available; e.g., head corrections, radiation shine, final technical specifications, and equipment calculations and tolerances. The NRC has been evaluating possible options to ensure applicants address the regulations and provide the following:

Option 1 – Submit an entire EAL scheme, which contains all site-specific information, including set points. Until this information is finalized, EALs would remain an open item.

Option 2 – Submit emergency plan Section D, "Emergency Classification System," which addresses the four critical elements of an EAL scheme (listed below). The NRC will determine the acceptability of the EAL scheme.

…

Please review the two options provided above, identify which option will be chosen, and provide the detailed EAL information in support of the chosen option."

SNC responded to RAI 13.03-03 in a letter dated April 3, 2009 [ML090990453]. Supplemental responses to RAI 13.03-3 were provided June 18, 2009 [ML091750106] and on April 28, 2010 [ML101200570].

SNC committed to Option 2 of the subject RAI. Option 2 identified four Critical Elements to be addressed if Option 2 was selected. SNC's response to the four critical Elements is shown below:

Critical Element 1 – "Emergency Plan Subsection D.1, Classification of Emergencies, provides an overview defining the four emergency classification levels: Notification of Unusual Event, Alert, Site Area Emergency, and General Emergency as defined in NEI 99-01, Rev. 5. Subsection D.1 also provides a general list of applicant actions at each emergency classification level."

Critical Element 2 – "SNC will develop the remainder of the site-specific EAL scheme using the NRC-endorsed version of NEI 07-01, Rev. 0. The fully developed site-specific EAL scheme will be included in the Emergency Plan. Accordingly, the current EAL scheme will be removed from Annex V2, Section D of the Emergency Plan in a future revision of the COLA. In addition, Section D of the Emergency Plan will be revised to clarify the basis for the EAL scheme."

Critical Element 3 – "SNC proposes the following License Condition related to the creation of a fully developed set of site-specific EALs in accordance with the guidance document discussed above:"

PROPOSED LICENSE CONDITION:

> "The licensee shall submit a fully developed set of site-specific Emergency Action Levels (EALs) to the NRC in accordance with the NRC-endorsed version of NEI 07-01, Revision 0 with no deviations. The EALs shall have been discussed and agreed upon with State and local officials. These fully developed EALs shall be submitted to the NRC for confirmation at least 180 days prior to initial fuel load."

Critical Element 4 – "As discussed in Critical Element 2, the fully developed site-specific EAL scheme will be incorporated into a future revision to the Emergency Plan. Accordingly, future changes to the EAL scheme will require an evaluation under 10 CFR 50.54(q) to determine if the changes will reduce the effectiveness of the Emergency Plan."

As a result of SNC's commitment to Option 2 in response to RAI 13.03-3 as supplemented, the following License Condition, 2.D(12)(d), was generated for each unit:

> "(d) No later than 180 days before initial fuel load, SNC shall submit to the Director of NRO, or the Director's designee, in writing, a fully developed set of plant-specific emergency action levels (EALs) for VEGP Unit 3 [4] in accordance with Nuclear Energy Institute (NEI) 07-01, "Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors," Revision 0, with no deviations. The EALs shall have been discussed and agreed upon with State and local officials."

Given that NEI 07-01, Revision 0, was issued in July 2009, the requirement to develop EALs in accordance with NEI 07-01, Revision 0, with no deviations (as is stipulated in the license condition) would inhibit SNC's ability to include pertinent updates that have been identified since 2009. In order to provide EALs that reflect the latest industry and regulatory guidance, SNC is requesting to be allowed to develop the proposed new plant-specific EALs for Vogtle Electric Generating Plant Units 3 and 4 using NEI 07-01, Rev. 0, as well as updates related to AP1000 design improvements and recommendations from EAL technical reviews performed by Westinghouse. Therefore, SNC is proposing that License Condition 2.D(12)(d) for VEGP Units 3 and 4 be modified to allow submittal of plant-specific EALs that are developed based on the criteria described above, as opposed to limiting those EALs to the guidance provided solely in NEI 07-01, Revision 0, with no deviations. Modifications to License Condition 2.D(12)(d) for Units 3 and 4 are proposed as follows:

> "(d) No later than 180 days before initial fuel load, SNC shall submit to the Director of NRO, or the Director's designee, in writing, a fully developed set of plant-specific emergency action levels (EALs) for VEGP Unit 3 [4] in accordance with the criteria defined in Amendment No. XX ~~Nuclear Energy Institute (NEI) 07-01, "Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors," Revision 0, with no deviations~~. The EALs shall have been discussed and agreed upon with State and local officials."

The proposed changes, including the modification of VEGP Units 3 and 4 License Conditions 2.D(12)(d) and submittal of the new plant-specific EALs for both units, affect the VEGP Units 3 and 4 Combined Licenses, but do not alter requirements of the Emergency Plan or Technical Specifications. These changes do not alter any of the assumptions used in the safety analyses, nor do they cause any safety system parameters to exceed their acceptance limit. Therefore, the proposed changes have no adverse effect on plant safety.

3. TECHNICAL EVALUATION (INCLUDED IN SECTION 2)

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.47(b)(4) states: "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."

10 CFR 50 Appendix E, Section IV "Content of Emergency Plans," item B, "Assessment Actions," states:

"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."

"2. A licensee desiring to change its entire emergency action level scheme shall submit an application for an amendment to its license and receive NRC approval before implementing the change. Licensees shall follow the change process in § 50.54(q) for all other emergency action level changes."

This LAR submits a change as described in Item 2 above. Therefore, the proposed new plant-specific EAL scheme requires NRC approval prior to implementation.

4.2 Precedent

No precedent is identified.

4.3 Significant Hazards Consideration Determination

Southern Nuclear Operating Company (SNC) has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

4.3.1 Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The requested amendment proposes changes to the Vogtle Electric Generating Plant (VEGP) Units 3 and 4 License Conditions 2.D(12)(d) and submits the new plant-specific Emergency Action Level (EAL) scheme for both units.

The proposed changes, including the modification of VEGP Units 3 and 4 License Condition 2.D(12)(d) and submittal of the new plant-specific EALs for both units, do not impact the physical function of plant structures, systems, or components (SSCs) or the manner in which SSCs perform their design function. The proposed changes neither adversely affect accident initiators or precursors, nor alter design assumptions. The proposed changes do not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an initiating event within assumed acceptance limits. No operating procedures or administrative controls that function to prevent or mitigate accidents are affected by the proposed changes.

Therefore, the requested amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

4.3.2 Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed changes, including the modification of VEGP Units 3 and 4 License Conditions 2.D(12)(d) and submittal of the new plant-specific EALs for both units, do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed or removed) or a change in the method of plant operation. The proposed changes will not introduce failure modes that could result in a new accident, and the changes do not alter assumptions made in the safety analysis. The proposed changes are not initiators of any accidents.

Therefore, the requested amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

4.3.3 Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

Margin of safety is associated with the ability of the fission product barriers (i.e., fuel cladding, reactor coolant system pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed changes to the plant-specific EALs and the modification of VEGP Units 3 and 4 License Conditions 2.D(12)(d) do not impact operation of the plant or its response to transients or accidents. The proposed changes do not affect the Technical Specifications. The proposed changes do not involve a change in the method of plant operation, and no accident analyses will be affected by the proposed changes.

Additionally, the proposed changes will not relax any criteria used to establish safety limits and will not relax any safety system settings. The safety analysis acceptance criteria are not affected by these proposed changes. The proposed changes will not result in plant operation in a configuration outside the design basis. The proposed changes do not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

4.4 Conclusion

Based on the above, SNC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92, and accordingly, a finding of "no significant hazards consideration" is justified.

5. ENVIRONMENTAL CONSIDERATIONS

This amendment request proposes modifications to the VEGP Units 3 and 4 License Conditions 2.D(12)(d) and submits the new plant-specific EAL scheme for both units. The proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.21 and 10 CFR 51.22(c)(9), in that:

(i) *There is no significant hazards consideration.*

As documented in Section 4.3, Significant Hazards Consideration Determination, of this amendment request, an evaluation was completed to determine whether or not a significant hazards consideration is involved by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment.” As noted above, the proposed changes will not affect how the plant is designed, constructed or operated. The Significant Hazards Consideration determined that; (1) the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) the proposed amendment does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of “no significant hazards consideration” is justified.

(ii) *There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.*

The requested amendment proposes changes to the VEGP Units 3 and 4 License Conditions 2.D(12)(d) and submits the new plant-specific EAL scheme for both units. As noted above, the proposed changes will not affect how the plant is designed, constructed, or operated. The VEGP Units 3 and 4 EALs are unrelated to any aspects of plant construction or operation that would introduce any changes to effluent types (e.g., effluents containing chemicals or biocides, sanitary system effluents, and other effluents) or affect any plant radiological or non-radiological effluent release quantities. Furthermore, these changes do not diminish the functionality of any design or operational features that are credited with controlling the release of effluents during plant operation.

Therefore, the proposed amendment does not involve a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite.

(iii) *There is no significant increase in individual or cumulative occupational radiation exposure.*

The requested amendment proposes changes to the VEGP Units 3 and 4 License Conditions 2.D(12)(d) and submits the new plant-specific EAL scheme for both units. As noted above, the proposed changes will not affect how the plant is designed, constructed, or operated. Consequently, the proposed changes have no effect on individual or cumulative occupational radiation exposure during plant operation. Therefore, the proposed amendment does not involve a significant increase in individual or cumulative occupational radiation exposure.

Based on the above review of the proposed amendment, it has been determined that anticipated construction and operational effects of the proposed amendment do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment of the proposed amendment is not required.

6. REFERENCES

None

Southern Nuclear Operating Company

ND-16-0150

Enclosure 2

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Technical Bases Document for the Proposed Emergency Action Levels

(LAR-16-002)

(Enclosure 2 consists of 219 pages, including this cover page)

VOGTLE ELECTRIC GENERATING PLANT
Units 3 & 4

EMERGENCY ACTION LEVELS

INITIATING CONDITIONS, THRESHOLD VALUES, AND BASIS

TABLE OF CONTENTS

TABLE OF CONTENTS

1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the Vogtle Electric Generating Plant Unit 3 and Unit 4 (VEGP3/4) EAL scheme. Decision-makers responsible for implementation of NMP-EP-110, Emergency Classification Determination and Initial Action, may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Director 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 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 Director 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). Additionally, changes to plant AOPs and EOPs that may impact EAL bases shall 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 Southern Nuclear Operating Company Standard Emergency Plan.

2.2 Fission Product Barrier Thresholds

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 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. Fuel Clad Barrier: The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.

B. RCS Barrier: The RCS Barrier includes the RCS primary side and its connections up to and including the reactor coolant pressure boundary isolation valves.

C. Containment Barrier: The Containment Barrier includes the containment 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 containment 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:

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 the third barrier*

2.4 EAL Organization

The VEGP3/4 EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
    - EALs applicable under all plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.

- EALs applicable only under hot operating modes – This group would only be reviewed by the EAL-user when the plant is in Safe Shutdown, Hot Standby, Startup, or Power Operations mode.
- EALs applicable only under cold operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling or Defueled 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/subcategories.
- Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. Subcategories are used in the VEGP3/4 scheme as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The VEGP3/4 EAL categories/subcategories are listed below.

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
ALL Conditions:	
R – Abnormal **R**ad Levels / Rad Effluent	1 – Radiological Effluent 2 – **I**rradiated 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 Gases 6 – Main Control Room Evacuation 7 – Judgment
Hot Conditions:	
S – **S**ystem Malfunction	1 – Loss of Electrical Power 2 – Loss of Monitoring and Control Functions 3 – RCS Activity 4 – RCS Leakage 5 – RTS Failure 6 – Loss of Communications 7 – Containment Isolation or Pressure Control Failure 8 – Hazardous Event Effecting Safety Systems
F – **F**ission Product Barrier Degradation	None
Cold Conditions:	
C – **C**old Shutdown / Refueling System Malfunction	1 – RCS Level 2 – RCS Temperature 3 – Loss of Class 1E DC or UPS Power 4 – Loss of AC Power 5 – Loss of Communications 6 – Loss of Monitoring and Control Functions 7 – Hazardous Event Effecting Safety Systems

The primary tool for determining the emergency classification level (ECL) is the EAL Classification Matrix. The user of the EAL Classification Matrix may (but is not required to) consult the EAL Technical Bases Document 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 (All, Hot, Cold), EAL category (R, C, H, S 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>Initiating Condition (IC) Identifier:</u>

Three characters define each IC identifier, e.g., FG1:

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

   G = General Emergency

   S = Site Area Emergency

   A = Alert

   U = Unusual Event
3. Third character (number): Subcategory (IC) 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).

<u>Emergency Classification Level (ECL):</u>

Unusual Event, Alert, Site Area Emergency or General Emergency.

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

<u>Operating Mode Applicability:</u>

One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Power Operations, 2 - Startup, 3 - Hot Standby, 4 - Safe Shutdown, 5 - Cold Shutdown, 6 - Refueling, Def - Defueled, or ALL. (See Section 2.6 for operating mode definitions)

Notes (where applicable)

Emergency Action Level(s) (EAL)

Each EAL is assigned a number corresponding to the number of EAL thresholds associated with the IC.

Basis:

A basis section that provides VEGP3/4-relevant information concerning the EAL. If the EAL wording contains a defined term, the definition of the term is included in this section. Defined terms used in an IC or EAL are identified by upper case letters. These definitions can also be found in Section 5.1.

VEGP3/4 Basis Reference(s):

Plant-specific source documentation from which the EAL is derived

2.6 Operating Mode Applicability (ref. 4.1.6)

1 Power Operations

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

2 Startup

$K_{eff} \geq 0.99$ and rated thermal power ≤ 5%.

3 Hot Standby

$K_{eff} < 0.99$ and average coolant temperature $T_{avg} > 420°F$.

4 Safe Shutdown

$K_{eff} < 0.99$, average coolant temperature $420°F \geq T_{avg} > 200°F$ and all reactor vessel head closure bolts fully tensioned.

5 Cold Shutdown

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

6 Refueling

Fuel in the reactor vessel with one or more vessel head closure bolts less than fully tensioned.

D Defueled

All reactor fuel removed from reactor pressure vessel (full core off load during refueling or extended outage).

The mode in effect at the time an event or condition occurs, 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 Safe 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 Safe Shutdown mode or higher.

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Emergency Director 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 ECL. 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. 4.1.8).

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.”

3.1.2 Valid Indications

All emergency classification assessments shall be based upon VALID indications, reports or conditions. 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 Director 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 10CFR 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.); the EAL and/or the associated basis discussion will identify the necessary analysis. In these cases, the 15-minute declaration period starts with the availability and report of the analysis results to the Main Control Room 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 Director 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. This EAL scheme provides the Emergency Director 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 Director 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 into 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(s) must be consistent with the related Operating Mode Applicability and Notes. If an EAL has been met or exceeded, then the IC is considered met and the associated ECL is declared in accordance with plant procedures.

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. 4.1.8).

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 ECL 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 Safe 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 Safe Shutdown mode or higher.

3.2.3 Classification of Imminent Conditions

Although EALs provide specific thresholds, the Emergency Director 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 Director, meeting an EAL is IMMINENT, the emergency classification should be made as if the EAL has been met. While applicable to all ECLs, this approach is particularly important at the higher ECLs since it provides additional time for implementation of protective measures.

3.2.4 Emergency Classification Level Upgrading and Downgrading

By generic industry classification guidance, an ECL may be downgraded when the event or condition that meets the highest IC and EAL no longer exists, and other plant-specific downgrading requirements are met. However, per VEGP3/4 Emergency Plan and implementing procedure guidance, down-grading of the ECL is not performed, rather the ECL is simply terminated, when the termination conditions are all met.

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 a failure of the reactor protection system to automatically scram/trip the reactor followed by a successful manual scram/trip or an event, such as an earthquake.

3.2.6 Classification of Transient Conditions

Many of the ICs and/or EALs contained in this document 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.

<u>EAL momentarily met during expected plant response</u> - 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.

<u>EAL momentarily met but the condition is corrected prior to an emergency declaration</u> – 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 PRHR fails to automatically actuate. Steam generator levels rapidly decrease and the plant enters an inadequate RCS heat removal condition (a potential loss of both the fuel clad and RCS barriers). If an operator manually actuates the PRHR (either by PMS or DAS) in accordance with an EOP step and clears the inadequate RCS heat removal 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 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 where an operator is able to take a successful corrective action prior to the Emergency Director 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 RIS 2007-02 Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events, February 2, 2007.

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

4.1.3 10 § CFR 50.72 Immediate Notification Requirements for Operating Nuclear Power Reactors

4.1.4 10 § CFR 50.73 License Event Report System

4.1.5 [*site specific- VEGP3/4 Offsite Dose Calculation Manual (ODCM)*]

4.1.6 UFSAR Chapter 16 Technical Specifications Modes Table

4.1.7 SNC Standard Emergency Plan

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

4.2 Implementing

4.2.1 NMP-EP-110 Emergency Classification Determination and Initial Action

4.2.2 VEGP Units 3 & 4 EAL Matrix

5.0 DEFINITIONS, ACRONYMS, & ABBREVIATIONS

5.1 Definitions

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

CONTAINMENT CLOSURE

Technical Specification Section 3.6 contains the meaning of containment closure for Modes 5 and 6. Containment closure means that all potential escape paths are closed or capable of being closed. Since there are no requirements for containment leak tightness, compliance with Appendix J leakage criteria and test are **not** required.

DEFUELED (mode)

All reactor fuel removed from reactor pressure vessel (full core off load during refueling or extended outage).

EMERGENCY ACTION LEVEL (EAL)

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

EMERGENCY CLASSIFICATION LEVEL (ECL)

One of a set of names or titles (Unusual Event, Alert, Site Area Emergency, and General Emergency) 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.

EMERGENCY DIRECTOR (ED)

A senior VEGP3/4 employee with overall responsibility for coordinating emergency response actions of the station, and the ERO with the affected state(s) and county agencies.

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 VEGP3/4 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. Observation of flame is preferred but is **not** required if large quantities of smoke and heat are observed. Sources of smoke such as slipping drive belts or overheated electrical equipment do **not** constitute fires.

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 process 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 VEGP 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 VEGP. 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.

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

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.

OWNER CONTROLLED AREA (OCA)

The area owned by the licensee and located within the confines of the SITE BOUNDARY (ref. 4.1.7).

PLANT OPERATOR

Any member of the plant staff who, by virtue of training and experience, is qualified to assess the indications or reports for validity and to compare the same to the EAL Matrix in Attachment I. The Plant Operator has the authority to declare the appropriate EAL and activate the Emergency Plan. A Plant Operator does **not** encompass plant personnel such as chemists, radiation protection technicians, security personnel, and others whose position require they report rather than assess abnormal conditions to the Main Control Room or Technical Support Center (TSC). The Plant Operator is the duty Shift Manager or the Emergency Director responsible for managing the emergency response (ref. 4.2.1).

PROJECTILE

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

PROTECTED AREA (PA)

An area, located within the Site Exclusion Area Boundary, encompassed by physical barriers and to which access is controlled per 10 CFR 73.55. The SNC Power Block Protected Area and the ISFSI Protected Area are two Protected Areas located within the Site Owner Controlled Area. (ref. 4.1.7).

RCS INTACT

The RCS can act as an effective pressure boundary.

REDUCED INVENTORY

RCS level 3 feet below the reactor vessel flange (8% wide range pressurizer level).

REFUELING PATHWAY

The reactor refueling cavity, spent fuel pool, and fuel transfer canal comprise the refueling pathway.

RUPTURED

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

SAFETY SYSTEM

A system required for safe plant operation, cooling down the plant, and/or placing it in the safe 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 process 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

The boundary of a reactor site beyond which the land or property is not owned, leased, or otherwise controlled by the licensee (ref. 4.1.7).

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 process 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 equipment 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 equipment or structure.

5.2 Acronyms & Abbreviations

°F .... Degrees Fahrenheit
' .... Feet or Minutes
" .... Inches or Seconds
ϒ .... Gamma

η .... Neutron

µCi/gm .... micro Curie per gram

AC .... Alternating Current

ALWR .... Advanced Light Water Reactor

ATWS .... Anticipated Transient Without Scram

CDE .... Committed Dose Equivalent

CFR .... Code of Federal Regulations

CTMT .... Containment

CSFST .... Critical Safety Function Status Tree

DAS .... Diverse Actuation System

DBA .... Design Basis Accident

DC .... Direct Current

DCD .... Design Control Document

DDS .... Data Display and Processing System

DG .... Non-Class 1-E Diesel Generator

EAL .... Emergency Action Level

ECCS .... Emergency Core Cooling System

ECL .... Emergency Classification Level

ECS .... Main AC Power System

EDS .... Non 1-E DC and UPS

EOP .... Emergency Operating Procedure

EPA .... Environmental Protection Agency

FSAR .... Final Safety Analysis Report

IC ....Initiating Condition

IDS.. Class 1-E DC and UPS Power System

IRWST ....In-containment Refueling Water Storage Tank

K_{eff}... Effective Neutron Multiplication Factor

LCO ....Limiting Condition for Operation

LOCA..Loss of Coolant Accident

MCR .... Main Control Room

MSSV ....Main Steam Safety Valve

mR ....milli-Roentgen

mRem, mrem, mREM ....milli-Roentgen Equivalent Man

NEI...Nuclear Energy Institute

NRC...Nuclear Regulatory Commission

NSSS..Nuclear Steam Supply System

(NO)UE...(Notification Of) Unusual Event

OBE ....Operating Basis Earthquake

OCA..Owner Controlled Area

ODCM..Offsite Dose Calculation Manual

ORO ....Off-site Response Organization

PA..Protected Area

PAG ....Protective Action Guideline

PCS ....Passive Containment Cooling System

PDSP...Primary Dedicated Safety Panel

PMS...Protection and Safety Monitoring System

PORV ....Power Operated Relief Valve

PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment

PRHR .... Passive Residual Heat Removal

PSIG .... Pounds per Square Inch Gauge

PXS .... Passive Core Cooling System

R .... Roentgen

RCS .... Reactor Coolant System

Rem, rem, REM .... Roentgen Equivalent Man

RNS .... Normal Residual Heat Removal

RPV .... Reactor Pressure Vessel

RTS .... Reactor Trip System

SFP .... Spent Fuel Pool

SG .... Steam Generator

TEDE .... Total Effective Dose Equivalent

TOAF .... Top of Active Fuel

TSC .... Technical Support Center

UFSAR .... Updated Final Safety Analysis Report

VBS .... Nuclear Island Non-radioactive Ventilation System

6.0 ATTACHMENTS

6.1 Attachment 1, Emergency Action Level Technical Bases

6.2 Attachment 2, Fission Product Barrier Matrix and Bases

6.3 Attachment 3, Safe Operation & Shutdown Areas/Rooms Tables R-2 & H-2 Bases

6.4 Attachment 4, Figures

6.5 Attachment 5, Developer Notes

ATTACHMENT 1

EMERGENCY ACTION LEVEL

TECHNICAL BASES

Category R – Abnormal Rad Levels / Radiological Effluent

EAL Group: ALL (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 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.

RU1

ECL: Unusual Event

Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the [*ODCM*] limits for 60 minutes or longer.

Operating Mode Applicability: All

Emergency Action Levels: (1 or 2)

Note 1:	The Emergency Director 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.

(1) Reading on **any** of the following effluent radiation monitors for ≥ 60 min. (Notes 1, 2, 3)

Plant Vent VFS-RY103	2 X Hi-Rad alarm
Turbine Island Vent TDS-RY001A	2 X Hi-Rad alarm
Liquid Radwaste WLS-RY229	2 X Hi-Rad alarm
Waste Water WWS-RY021	2 X Hi-Rad alarm

(2) Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x [*ODCM*] limits for ≥ 60 min. (Notes 1, 2)

Basis:

[Developer notes: *See Attachment 5*]

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 as well as planned batch radioactivity releases from monitored gaseous or liquid effluent pathways.

EAL (1)

Gaseous Releases

The Plant Vent Gas radiation monitors, VFS-RY103 (normal range - 1.0E-7 to 1.0E-2 μCi/cc), VFS-RY104A (mid range - 1.0E-4 to 1.0E+2 μCi/cc) and VFS-RY104B (hi range - 1.0E-1 to 1.0E+5 μCi/cc) measures the concentration of gaseous radioactive airborne contamination being released through the plant vent, which is the only design pathway for the release of radioactive materials to the atmosphere (ref. 2).

The Turbine Island Vent discharge radiation monitor (TDS-RY001A/B) measures the concentration of radioactive gases in the steam and non-condensable gases that are discharged by the condenser vacuum pumps and the gland seal steam condenser. This measurement provides early indication of leakage between the primary and secondary sides of the steam generators. The monitor provides an alarm in the Main Control Room if concentrations exceed a predetermined setpoint. Turbine island vent radiation monitor includes two G-M tubes with nominal ranges of 1.0E-6 to 1.0E+0 μCi/cc (normal range) and 1.0E-1 to1.0E+5 μCi/cc (hi range) (ref. 2).

Liquid Releases

The liquid radwaste discharge radiation monitor (WLS-RY229 - 1.0E-6 to 1.0E-1 μCi/cc) measures the concentration of radioactive materials in liquids released to the environment. The liquid releases are made in batches that are mixed thoroughly and sampled. The samples are analyzed on site before discharge to determine that the discharge is within allowable concentration limits and within allowable totals. The liquid radwaste discharge radiation monitor is an inline monitor that provides signals to isolate the discharge of liquid radwaste, stop the liquid radwaste system discharge pumps and alarms in the Main Control Room if the concentrations exceed a predetermined setpoint (ref. 2).

The waste water discharge radiation monitor (WWS-RY021 - 1.0E-7 to 1.0E-2 µCi/cc) measures the concentration of radioactive materials in the discharge from the waste water system. The waste water discharge radiation monitor provides data for reports of liquid releases of radioactive materials. The waste water discharge radiation monitor is an inline monitor. Upon reaching the high radiation setpoint the turbine building sump pumps stop and an alarm is initiated in the Main Control Room. Following an alarm, the operator can manually realign the discharge to the liquid radwaste system for processing (ref. 2, 3).

EAL (2)

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 ECL would be via IC RA1.

VEGP3/4 Basis Reference(s):

1. [*plant-specific - VEGP3/4 Offsite Dose Calculation Manual*]
2. UFSAR Section 11.5.2.3.3 Liquid and Gaseous Effluent Monitors
3. APP-WWS-M3C-101 section 5.1.3.6.

RA1

ECL: Alert

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

Operating Mode Applicability: All

Emergency Action Levels: (1 or 2 or 3)

Note 1:	The Emergency Director 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, RS1 and RG1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

(1) Reading on **any** of the following effluent radiation monitors for ≥ 15 min. (Notes 1, 2, 3, 4)

Plant Vent VFS-RY103	[plant-specific]
Plant Vent VFS-RY104A	[plant-specific]
Plant Vent VFS-RY104B	[plant-specific]
Turbine Island Vent TDS-RY001A	[plant-specific]
Turbine Island Vent TDS-RY001B	[plant-specific]

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

(3) 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)

Basis:

SITE BOUNDARY - The boundary of a reactor site beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

This IC addresses a release of gaseous 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.

<u>EAL (1)</u>

[Developer notes: *See Attachment 5*]

The Plant Vent Gas radiation monitors, VFS-RY103 (normal range - 1.0E-7 to 1.0E-2 μCi/cc), VFS-RY104A (mid range - 1.0E-4 to 1.0E+2 μCi/cc), and VFS-RY104B (hi range - 1.0E-1 to 1.0E+5 μCi/cc) measures the concentration of gaseous radioactive airborne contamination being released through the plant vent, which is the only design pathway for the release of radioactive materials to the atmosphere (ref. 1).

The Turbine Island Vent discharge radiation monitor (TDS-RY001A/B) measures the concentration of radioactive gases in the steam and non-condensable gases that are discharged by the condenser vacuum pumps and the gland seal steam condenser. This measurement provides early indication of leakage between the primary and secondary sides of the steam generators. The monitor provides an alarm in the Main Control Room if concentrations exceed a predetermined setpoint. Turbine island vent radiation monitor includes two G-M tubes with nominal ranges of 1.0E-6 to 1.0E+0 μCi/cc (normal range) and 1.0E-1 to1.0E+5 μCi/cc (hi range) (ref. 1).

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.

<u>EAL (2)</u>

Dose assessment may be performed computer based methods (ref. 3).

<u>EAL (3)</u>

91303-C Field Sampling and Surveys, provides guidance for emergency or post-accident radiological environmental monitoring (ref. 4).

Escalation of the ECL would be via IC RS1.

VEGP3/4 Basis Reference(s):

1. UFSAR Section 11.5.2.3.3 Liquid and Gaseous Effluent Monitors
2. [*Plant-specific calculation supporting effluent release values*]
3. NMP-EP-104 Dose Assessment
4. 91303-C Field Sampling and Surveys

RS1

ECL: Site Area Emergency

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

Operating Mode Applicability: All

Emergency Action Levels: (1 or 2 or 3)

Note 1:	The Emergency Director 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, RS1 and RG1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

(1) Reading on **any** of the following effluent radiation monitors for ≥ 15 min. (Notes 1, 2, 3, 4)

Plant Vent VFS-RY103	[plant-specific]
Plant Vent VFS-RY104A	[plant-specific]
Plant Vent VFS-RY104B	[plant-specific]
Turbine Island Vent TDS-RY001A	[plant-specific]
Turbine Island Vent TDS-RY001B	[plant-specific]

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

(3) 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)

Basis:

SITE BOUNDARY - The boundary of a reactor site beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

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.

EAL (1)

[Developer notes: *See Attachment 5*]

The Plant Vent Gas radiation monitors, VFS-RY103 (normal range - 1.0E-7 to 1.0E-2 μCi/cc), VFS-RY104A (mid range - 1.0E-4 to 1.0E+2 μCi/cc) and VFS-RY104B (hi range - 1.0E-1 to 1.0E+5 μCi/cc) measures the concentration of gaseous radioactive airborne contamination being released through the plant vent, which is the only design pathway for the release of radioactive materials to the atmosphere (ref. 1).

The Turbine Island Vent discharge radiation monitor (TDS-RY001A/B) measures the concentration of radioactive gases in the steam and non-condensable gases that are discharged by the condenser vacuum pumps and the gland seal steam condenser. This measurement provides early indication of leakage between the primary and secondary sides of the steam generators. The monitor provides an alarm in the Main Control Room if concentrations exceed a predetermined setpoint. Turbine island vent radiation monitor includes two G-M tubes with nominal ranges of 1.0E-6 to 1.0E+0 μCi/cc (normal range) and 1.0E-1 to1.0E+5 μCi/cc (hi range) (ref. 1).

EAL (2)

Dose assessment may be performed computer based methods (ref. 3).

EAL (3)

91303-C Field Sampling and Surveys, provides guidance for emergency or post-accident radiological environmental monitoring (ref. 4).

Escalation of the ECL would be via IC RG1.

VEGP3/4 Basis Reference(s):

1. UFSAR Section 11.5.2.3.3 Liquid and Gaseous Effluent Monitors
2. [*Plant-specific calculation supporting effluent release values*]
3. NMP-EP-104 Dose Assessment
4. 91303-C Field Sampling and Surveys

RG1

ECL: General Emergency

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1000 mrem TEDE or 5000 mrem thyroid CDE.

Operating Mode Applicability: All

Emergency Action Levels: (1 or 2 or 3)

Note 1:	The Emergency Director 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, RS1 and RG1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

(1) Reading on **any** of the following effluent radiation monitors for ≥ 15 min. (Notes 1, 2, 3, 4)

Plant Vent VFS-RY103	[plant-specific]
Plant Vent VFS-RY104A	[plant-specific]
Plant Vent VFS-RY104B	[plant-specific]
Turbine Island Vent TDS-RY001A	[plant-specific]
Turbine Island Vent TDS-RY001B	[plant-specific]

(2) Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

(3) Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation.

(Notes 1, 2)

Basis:

SITE BOUNDARY - The boundary of a reactor site beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses

greater than or equal to 100% 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 100% of the EPA PAG of 1,000 mrem while the 5000 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.

<u>EAL (1)</u>

[Developer notes: *See Attachment 5*]

The Plant Vent Gas radiation monitors, VFS-RY103 (normal range - 1.0E-7 to 1.0E-2 μCi/cc), VFS-RY104A (mid range - 1.0E-4 to 1.0E+2 μCi/cc) and VFS-RY104B (hi range - 1.0E-1 to 1.0E+5 μCi/cc) measures the concentration of gaseous radioactive airborne contamination being released through the plant vent, which is the only design pathway for the release of radioactive materials to the atmosphere (ref. 1).

The Turbine Island Vent discharge radiation monitor (TDS-RY001A/B) measures the concentration of radioactive gases in the steam and non-condensable gases that are discharged by the condenser vacuum pumps and the gland seal steam condenser. This measurement provides early indication of leakage between the primary and secondary sides of the steam generators. The monitor provides an alarm in the Main Control Room if concentrations exceed a predetermined setpoint. Turbine island vent radiation monitor includes two G-M tubes with nominal ranges of 1.0E-6 to 1.0E+0 μCi/cc (normal range) and 1.0E-1 to1.0E+5 μCi/cc (hi range) (ref. 1).

<u>EAL (2)</u>

Dose assessment may be performed computer based methods (ref. 3).

<u>EAL (3)</u>

91303-C Field Sampling and Surveys, provides guidance for emergency or post-accident radiological environmental monitoring (ref. 4).

Escalation of the ECL would be via IC RG1.

VEGP3/4 Basis Reference(s):

1. UFSAR Section 11.5.2.3.3 Liquid and Gaseous Effluent Monitors
2. [*Plant-specific calculation supporting effluent release values*]
3. NMP-EP-104 Dose Assessment
4. 91303-C Field Sampling and Surveys

RU2

ECL: Unusual Event

Initiating Condition: UNPLANNED loss of water level above irradiated fuel.

Operating Mode Applicability: All

Emergency Action Levels: (1)

(1) UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **any** of the following:

- Spent Fuel Pool Low Alarm (SFS-LT019A/B/C)
- Level indication trend
- Visual observation

AND

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

- RMS-RY012 Fuel Handling Area Monitor
- RMS-RY020 Fuel Handling Area Monitor
- Refueling Bridge Portable Monitor (Refueling Mode)

Basis:

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 reactor refueling cavity, Spent Fuel Pool and fuel transfer canal comprise the refueling pathway.

Allowing level to decrease in the REFUELING PATHWAY could result in spent fuel being uncovered, reducing spent fuel decay heat removal, and creating an extremely hazardous radiation environment. Technical Specification LCO 3.7.5 requires at least 23 ft of water above the Spent Fuel Pool storage racks. Technical Specification LCO 3.9.4 requires at least 23 ft of water above the reactor vessel flange in the refueling cavity during refueling operations. This maintains sufficient water level in the fuel transfer canal, refueling cavity, and Spent Fuel Pool to retain iodine fission product activity in the water in the event of a fuel handling accident. (ref. 4, 5)

In the unlikely event of an extended loss of normal Spent Fuel Pool cooling, the water level will drop. Low Spent Fuel Pool level alarms and decreasing level indication trend in the Main Control Room will indicate to the operator the need to initiate makeup water to the pool. These alarms and indications are provided from safety-related and non-safety-related level instrumentation in the Spent Fuel Pool (ref 2).

The SFP level instrumentation is at an elevation of 229.5', which is 0.25' above the top of the spent fuel racks. It is capable of measuring the SFP level from the spent fuel racks up to the operating deck, which is at 255.25'. The instrument range is 0’ to 25.75’ as a result. The Low-2 level alarm indicates the possibility that a pipe break or leak exists in the system and that water is being lost through that break. The Low-2 water level alarm setpoint sets the alarm at an indicated water level of 23.5' above top of the racks (ref. 3).

Radiation monitors that may indicate a loss of shielding above irradiated fuel include (ref. 1):

- RMS-RY012 Fuel Handling Area Monitor
- RMS-RY020 Fuel Handling Area Monitor
- Refueling Bridge Portable Monitor (installed only in Refueling Mode)

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 ECL would be via IC RA2.

VEGP3/4 Reference(s):

1. UFSAR Section 11.5.6.4 Fuel Handling Area Criticality Monitors
2. UFSAR Section 9.1.3.4.3 Abnormal Conditions
3. APP-SFS-M3C-101 SFS Instrumentation and Packaged Mechanical System Interface Requirements
4. Technical Specifications LCO 3.7.5
5. Technical Specifications LCO 3.9.4

RA2

ECL: Alert

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

Operating Mode Applicability: All

Emergency Action Levels: (1 or 2 or 3)

(1) Uncovery of irradiated fuel in the REFUELING PATHWAY

(2) Damage to irradiated fuel resulting in a release of radioactivity as indicated by a Hi-Rad alarm on **any** of the following radiation monitors:

- VAS-RY001 Fuel Handling Area Exhaust
- VFS-RY001 Containment Air Filtration Exhaust
- RMS-RY012 Fuel Handling Area Monitor
- RMS-RY020 Fuel Handling Area Monitor
- Refueling Bridge Portable Monitor (Refueling Mode)

(3) Lowering of Spent Fuel Pool level to Level 2 (10 ft. on SFS-LT019A/B/C)

Basis:

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:

REFUELING PATHWAY- The reactor refueling cavity, Spent Fuel Pool and fuel transfer canal comprise the refueling pathway.

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.

<u>EAL (1)</u>

Allowing level to decrease in the REFUELING PATHWAY could result in spent fuel being uncovered, reducing spent fuel decay heat removal, and creating an extremely hazardous radiation environment. Technical Specification LCO 3.7.5 requires at least 23 ft of water above the Spent Fuel Pool storage racks. Technical Specification LCO 3.9.4 requires at least 23 ft of water above the reactor vessel flange in the refueling cavity during refueling operations. This maintains sufficient water level in the fuel transfer canal, refueling cavity, and Spent Fuel Pool to retain iodine fission product activity in the water in the event of a fuel handling accident. (ref. 2, 3)

In the unlikely event of an extended loss of normal Spent Fuel Pool cooling, the water level will drop. Low Spent Fuel Pool level alarms and decreasing level indication trend in the Main Control Room will indicate to the operator the need to initiate makeup water to the pool. These alarms and indications are provided from safety-related and non-safety-related level instrumentation in the Spent Fuel Pool (ref 1).

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.

<u>EAL (2)</u>

When considering escalation, information may come from:

- Radiation monitor readings
- Sampling and surveys
- Dose projections/calculations
- Reports from the scene regarding the extent of damage (e.g., refueling crew, heath physics specialists)

Radiation monitors listed in this EAL are (ref. 4, 5, 6):

- VAS-RY001 Fuel Handling Area Exhaust: The fuel handling area exhaust radiation monitor measures the concentration of radioactive materials in the exhaust air from the fuel handling area.
- VFS-RY001 Containment Air Filtration Exhaust: The containment air filtration exhaust radiation monitor measures the concentration of radioactive materials in the containment purge exhaust air.
- RMS-RY012 Fuel Handling Area Monitor
- RMS-RY020 Fuel Handling Area Monitor
- Refueling Bridge Portable Monitor (installed only in Refueling Mode)

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).

<u>EAL (3)</u>

The VEGP3/4 spent fuel storage facility is located within the seismic Category I Auxiliary Building fuel handling area. (ref. 8).

The SFP level instrumentation is at an elevation of 229.5', which is 0.25' above the top of the spent fuel racks. It is capable of measuring the SFP level from the spent fuel racks up to the operating deck, which is at 255.25'. The instrument range is 0’ to 25.75’ as a result (ref. 8, 9).

Level 2 is 10 ft. above the top of the fuel racks.

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 would be based on either Recognition Category R or C ICs.

VEGP3/4 Reference(s):

1. UFSAR Section 9.1.3.4.3 Abnormal Conditions
2. Technical Specifications LCO 3.7.5
3. Technical Specifications LCO 3.9.4
4. UFSAR Section 11.5.2.3.2 Airborne Monitors
5. UFSAR Section 11.5.2.3.1 Fluid Process Monitors
6. UFSAR Section 11.5.6.4 Fuel Handling Area Criticality Monitors
7. UFSAR Section 9.1.2.2 Facilities Description
8. UFSAR Table 7.3-4 Engineered Safety Features Actuation, Variables, Limits, Ranges and Accuracies (Nominal)
9. APP-SFS-M3C-101 SFS Instrumentation and Packaged Mechanical System Interface Requirements
10. NL-12-2200 Vogtle Electric Generating Plant Units 3 and 4 Southern Nuclear Operating Company's Initial Status and Full Compliance Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)

RS2

ECL: Site Area Emergency

Initiating Condition: Spent fuel pool (containing irradiated fuel) level at the top of the fuel racks

Operating Mode Applicability: All

Emergency Action Levels: (1)

(1) Lowering of Spent Fuel Pool level to Level 3 (0 ft. on SFS-LT019A/B/C)

Basis:

The VEGP3/4 spent fuel storage facility is located within the seismic Category I Auxiliary Building fuel handling area. (ref. 1).

The SFP level instrumentation is at an elevation of 229.5', which is 0.25' above the top of the spent fuel racks. It is capable of measuring the SFP level from the spent fuel racks up to the operating deck, which is at 255.25'. The instrument range is 0' to 25.75' as a result (ref. 2, 3).

Level 3 is 0 ft. above the top of the fuel racks.

This IC 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 RG1 or RG2.

VEGP3/4 Reference(s):

1. UFSAR Section 9.1.2.2 Facilities Description
2. UFSAR Table 7.3-4 Engineered Safety Features Actuation, Variables, Limits, Ranges and Accuracies (Nominal)
3. APP-SFS-M3C-101 SFS Instrumentation and Packaged Mechanical System Interface Requirements
4. NL-12-2200 Vogtle Electric Generating Plant Units 3 and 4 Southern Nuclear Operating Company's Initial Status and Full Compliance Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)

RG2

ECL: General Emergency

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

Operating Mode Applicability: All

Emergency Action Levels: (1)

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

(1) Spent Fuel Pool level **cannot** be restored to at least Level 3 (0 ft. on SFS-LT019A/B/C) for $\geq$ 60 min. (Note 1)

Basis:

The VEGP3/4 spent fuel storage facility is located within the seismic Category I Auxiliary Building fuel handling area. (ref. 1).

The SFP level instrumentation is at an elevation of 229.5', which is 0.25' above the top of the spent fuel racks. It is capable of measuring the SFP level from the spent fuel racks up to the operating deck, which is at 255.25'. The instrument range is 0’ to 25.75’ as a result (ref. 2, 3).

Level 3 is 0 ft. above the top of the fuel racks.

This IC 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.

VEGP3/4 Reference(s):

1. UFSAR Section 9.1.2.2 Facilities Description
2. UFSAR Table 7.3-4 Engineered Safety Features Actuation, Variables, Limits, Ranges and Accuracies (Nominal)
3. APP-SFS-M3C-101 SFS Instrumentation and Packaged Mechanical System Interface Requirements
4. NL-12-2200 Vogtle Electric Generating Plant Units 3 and 4 Southern Nuclear Operating Company's Initial Status and Full Compliance Report in Response to March 12, 2012 Commission Order Modifying Licenses with Regard to Requirements for Reliable Spent Fuel Pool Instrumentation (Order Number EA-12-051)

RA3

ECL: Alert

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

Operating Mode Applicability: All

Emergency Action Levels: (1 or 2)

Note 5:	If the equipment in the listed area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.
Note 9:	Classification is not required if either train of RNS can be placed in service for Shutdown Cooling.

(1) Dose rate > 15 mR/hr in EITHER of the following areas:
- Main Control Room (RMS-RY010)
- Central Alarm Station (by survey)

(2) An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-1 area/room (Note 5)

Table R-1 Safe Operation & Shutdown Areas/Rooms

Building	Room Number	Room/Area Name	Applicable Mode
Auxiliary	12152	Primary Sample Room	3
	12252	Rad Chem Lab	3
	12321	Non-1E Equipment / Penetration Room	4
	12305	Division D I&C Penetration Room	4
	12304	Division B I&C Penetration Room	4
	12561	CCS Valve Room (Note 9)	4
Annex	4000	Annex Building (Note 9)	4
Containment	11209	CVS Room	4

Basis:

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.

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 Director should consider the cause of the increased radiation levels and determine if another IC may be applicable.

<u>EAL (1)</u>

RMS-RY010 (0.1 – 10,000 mR/hr) is the permanently installed Main Control Room area radiation monitor and, along with local radiation surveys, may be used to assess this EAL threshold. (ref. 1, 2). The Central Alarm Station (CAS) does not have a permanently installed area radiation monitor and therefore local survey must be performed to assess this criteria.

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 Director should consider the cause of the increased radiation levels and determine if another IC may be applicable.

<u>EAL (2)</u>

The Table R-1 safe operation and shutdown areas/rooms (with entry-related mode applicability) are those plant areas that contain equipment which require a manual/local action as specified in general operating procedures (and procedures referenced by them) used for normal plant operation, cooldown and shutdown. The list specifies the plant operating modes during which entry would be required for each area and thus specifying when a loss of access or IMPEDED access is applicable to this EAL (ref. 3).

Plant areas where actions of a contingent or emergency nature might be needed to be performed. (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) were not considered for inclusion. Additionally, areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections) were not considered for inclusion. Refer to Attachment 3 "Safe Operation & Shutdown Areas/Rooms Tables R-1 & H-3 Bases."

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.

For this EAL, 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 ECL would be via Recognition Category R, C or F ICs.

VEGP3/4 Basis Reference(s):

1. APP-RMS-J7-001 AP1000 Radiation Monitoring System - System Specification Document
2. UFSAR Table 11.5-2 Area Radiation Monitor Detector Parameters
3. NMP-EP-110-GL04 VEGP 3/4 EAL – ICs, Threshold Values and Basis, Attachment 3 "Safe Operation & Shutdown Areas/Rooms Tables R-1 & H-2 Bases”

Category C – Cold Shutdown / Refueling System Malfunction

EAL Group: COLD Conditions (RCS temperature ≤ 200ºF);
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, Def – Defueled).

The events of this category pertain to the following subcategories:

1. RCS Level

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

2. RCS Temperature

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

3. Loss of Class 1E DC or UPS Electrical Power

   Loss of Class 1E DC plant or UPS electrical power can compromise plant SAFETY SYSTEM operability including decay heat removal and core cooling systems which may be necessary to ensure fission product barrier integrity.

4. Loss of AC Power

Extended loss of onsite and offsite AC 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.

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. Loss of Monitoring and Control Functions

Loss of monitoring and control functions can challenge the Main Control Room staff ability to maintain SAFETY SYSTEM operability.

7. Hazardous Event Effecting Safety Systems

Certain hazardous natural and technological events may result in VISIBLE DAMAGE to or degraded performance of SAFETY SYSTEMS warranting classification.

CU1

ECL: Unusual Event

Initiating Condition: Inability to restore and maintain required RCS inventory level

Operating Mode Applicability: 5 - Cold Shutdown, 6 - Refueling

Emergency Action Levels: (1 or 2)

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

(1) Inability to restore and maintain RCS > a required lower limit ≥ 15 min. (Note 1)

(2) Reactor vessel/RCS level **cannot** be monitored

AND

UNPLANNED increase in **any** Table C-1 sump or tank levels due to a loss of reactor vessel/RCS inventory

Table C-1 Sumps & Tanks
• Visual observation in containment • Containment Sump • IRWST • RCDT • Auxiliary Building (WRS) Sump

Basis:

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.

EAL (1)

This EAL recognizes that the minimum required reactor vessel/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.

For example, with the plant in Cold Shutdown, RCS water level is normally maintained above the pressurizer Low-2 level setpoint of 10% (ref. 1). With the plant in Refueling mode, RCS water level is normally maintained above the reactor vessel flange.

This IC/EAL addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band) concurrent with indications of coolant leakage. This condition 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.

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.

<u>EAL (2)</u>

In this EAL, all water level indication is unavailable and the reactor vessel inventory loss must be detected by sump or tank level changes (Table C-1). Plant design and procedures provide the capability to detect and assess primary system leakage (ref. 2, 3).

This IC/EAL addresses the loss of the ability to monitor reactor vessel/RCS level concurrent with indications of coolant leakage. This condition 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 reactor vessel/RCS level have been lost. All means include local and remote instrumentation, cameras and direct observation. In this condition, 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 reactor vessel/RCS.

Continued loss of RCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA2.

VEGP3/4 Basis Reference(s):

1. APP-RCS-M3C-101 Section 5.1.3.5 Level Instrumentation and Packaged Mechanical Systems Interface Range and Setpoint Bases
2. APP-PXS-M3-001 Passive Core Cooling System SSD
3. APP-WLS-M3-001 Liquid Waste System SSD

CA1

ECL: Alert

Initiating Condition: Loss of reactor vessel/RCS inventory

Operating Mode Applicability: 5 - Cold Shutdown, 6 - Refueling

Emergency Action Levels: (1 or 2)

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

(1) RCS hot leg level < 64.5% for ≥ 15 min. (Note 1)

(2) Reactor vessel/RCS level **cannot** be monitored for ≥ 15 min. (Note 1)

AND

UNPLANNED increase in **any** Table C-1 sump or tank levels due to a loss of reactor vessel/RCS inventory

Table C-1 Sumps & Tanks
• Visual observation in containment • Containment Sump • IRWST • RCDT • Auxiliary Building (WRS) Sump

Basis:

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.

EAL (1)

An RCS Hot Leg water level indication of 64.5% corresponds to the Low-1 setpoint or approximately 20" above the inside bottom of the hot leg (ref. 1).

This IC/EAL addresses a condition that is a precursor 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 hot leg level below 64.5% for 15 minutes or longer indicates that operator actions have not been successful in restoring and maintaining reactor vessel/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 residual heat removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA2.

EAL (2)

In this condition, all RCS water level indication would be unavailable, and the reactor vessel inventory loss must be detected by sump or tank level changes (Table C-1). Plant design and procedures provide the capability to detect and assess primary system leakage (ref. 2, 3).

This IC/EAL addresses a condition that is a precursor 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 reactor vessel/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 reactor vessel/RCS.

The 15-minute duration for the loss of level indication was chosen to allow CA1.2 to be an effective precursor to CS1.1. This provides time to increase makeup and isolate leakage prior to core uncovery. Whether or not the actions in progress will be effective should be apparent within 15 minutes. When in Cold Shutdown or Refueling the event can be classified as an Alert due to the significantly reduced decay heat and lower temperature and pressure. This increases the time available to resolve the problem. Significant fuel damage is not expected to occur until after core uncovery has occurred per the analysis referenced in the CG1 basis.

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

VEGP3/4 Basis Reference(s):

1. APP-RCS-M3C-101 RCS Instrumentation and Packaged Mechanical System Interface Requirements
2. APP-PXS-M3-001 Passive Core Cooling System SSD

3. APP-WLS-M3-001 Liquid Waste System SSD

CS1

ECL: Site Area Emergency

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

Operating Mode Applicability: 5 - Cold Shutdown, 6 - Refueling

Emergency Action Levels: (1)

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

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

AND

Core uncover is indicated by EITHER of the following:

- CTMT High Range Radiation (PXS-RY160/161/162/163) reading > 100 R/hr (High-2 alarm)
- Erratic SR Excore Detector indication (RXS-NE001A/B/C/D)

Basis:

EAL (1)

The type and range of RCS/reactor vessel level instrumentation may vary during an outage as the plant moves through various operating modes and refueling evolutions, particularly for an AP1000. As appropriate to the plant design, alternate means of determining RCS/reactor vessel level are installed to assure that the ability to monitor level within the range required by operating procedures will not be interrupted. For the AP1000, the instrumentation range necessary to support implementation of operating procedures in the Cold Shutdown and Refueling modes may be different (e.g., narrower) than that required during modes higher than Cold Shutdown. The basis for this IC and associated EALs is reactor vessel/RCS level below the top of active fuel for great than 30 min. However, the lowest observable level on AP1000 RCS level instrumentation is the lowest observable level on the RCS Hot Leg Level Instrument. Therefore this EAL addresses alternative indication of core uncover once reactor vessel level drops below the bottom of the hot leg such as erratic SR Excore Detector indications.

The Containment High Range Radiation monitors, PXS-RY160, 161, 162, and 163 have a range of 1.0E0 to 1.0E+7 R/hr. The containment radiation High-2 alarm setpoint is 100 R/hr (ref. 1). The alarm setpoint is sufficiently high to be indicative of a significant loss of shielding above the core indicating likely core uncovery.

This IC/EAL addresses a significant and prolonged loss of reactor vessel/ inventory control and makeup capability leading to core uncovery and 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 include 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/RPV level. If RCS/reactor vessel 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). The passive cooling systems should continue to function and provide a sufficient volume of water for cooling, therefore, 30-minute duration allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment, and/or restore level monitoring.

The inability to monitor reactor vessel/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 core uncovery has occurred by observing the alternative indications specified.

These EALs address 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.

VEGP3/4 Basis Reference(s):

1. APP-PXS-M3C-101 Section 5.1.3.6 Radiation and Packaged Mechanical Systems Interface Instrumentation Range and Setpoints Bases

CG1

ECL: General Emergency

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

Operating Mode Applicability: 5 - Cold Shutdown, 6 - Refueling

Emergency Action Levels: (1)

Note 1:	The Emergency Director 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.

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

AND

Core uncover is indicated by EITHER of the following:

- CTMT High Range Radiation (PXS-RY160/161/162/163) reading > 100 R/hr (High-2 alarm)
- Erratic SR Excore Detector indication (RXS-NE001A/B/C/D)

AND

Any of the following indications of containment challenge:

- CONTAINMENT CLOSURE **not** established (Note 6)
- CTMT H_2 Concentration (VLS-AE001/002/003) > 4%
- UNPLANNED increase in Containment pressure that can breach CONTAINMENT CLOSURE

Basis:

CONTAINMENT CLOSURE -Technical Specification Section 3.6 contains the meaning of containment closure for Modes 5 and 6. Containment closure means that all potential escape paths are closed or capable of being closed. Since there are no requirements for containment leak tightness, compliance with Appendix J leakage criteria and test are **not** required.

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.

EAL (1)

The type and range of RCS/reactor vessel level instrumentation may vary during an outage as the plant moves through various operating modes and refueling evolutions, particularly for an AP1000. As appropriate to the plant design, alternate means of determining RCS/reactor vessel level are installed to assure that the ability to monitor level within the range required by operating procedures will not be interrupted. For the AP1000, the instrumentation range necessary to support implementation of operating procedures in the Cold Shutdown and Refueling modes may be different (e.g., narrower) than that required during modes higher than Cold Shutdown. The basis for this IC and associated example EALs is reactor vessel/RCS level below the top of active fuel for greater than 30 min. However, the lowest observable level on AP1000 RCS level instrumentation is the lowest observable level on the RCS Hot Leg Level Instrument. Therefore this EAL addresses alternative indication of core uncover once reactor vessel level drops below the bottom of the hot leg such as erratic SR Excore Detector indications.

The Containment High Range Radiation monitors, PXS-RY160, 161, 162, and 163 have a range of 1.0E0 to 1.0E+7 R/hr. The containment radiation High-2 alarm setpoint is 100 R/hr (ref. 1). The alarm setpoint is sufficiently high to be indicative of a significant loss of shielding above the core indicating likely core uncovery.

Three indications are associated with a challenge to Containment:

- CONTAINMENT CLOSURE is not established.
- 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 mixture of dissolved gases in Containment. However, Containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. A combustible mixture can be formed when hydrogen gas concentration in the Containment atmosphere is greater than 4% by volume.
- An UNPLANNED pressurization that can breach the CONTAINMENT CLOSURE signifies a challenge to the containment pressure retaining capability which is dependent on the status of the containment. If containment integrity is established for full power operation, a breach could occur if the design Containment pressure is exceeded (59 psig). For this condition, a small UNPLANNED pressure rise above atmospheric pressure does not challenge containment. If in refueling operations, however, a breach could occur if the UNPLANNED pressure rise exceeded the capability of a temporary containment seal. This

would occur at a much lower pressure than the containment design pressure. Use of the verb "...can breach..." instead of "breaches" provides the Emergency Director with the latitude to assess the magnitude and rate of the Containment pressure rise with respect to the barrier status (for the existing operating mode) and determine that the containment challenge exists due to elevated pressure either before or at the time that the actual breach of the barrier occurs.

This IC/EAL 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/RPV level. If RCS/reactor vessel 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 indications to assess whether or not containment is challenged.

An UNPLANNED increase in containment pressure may be an indication of energy addition to the containment (RCS heatup or boiling) under conditions when the containment barrier has limited pressure retention capability challenging CONTAINMENT CLOSURE.

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). The passive cooling systems should continue to function and provide a sufficient volume of water for cooling. Therefore, the 30-minute duration allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment, and/or restore level monitoring.

The inability to monitor reactor vessel/ 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 core uncovery has occurred by observing the alternative indications specified.

These EALs address 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*.

VEGP3/4 Basis Reference(s):

1. APP-PXS-M3C-101 Section 5.1.3.6 Radiation and Packaged Mechanical Systems Interface Instrumentation Range and Setpoints Bases
2. UFSAR Table 6.2.1.1-3 Results of Postulated Accidents

CU2

ECL: Unusual Event

Initiating Condition: UNPLANNED increase in RCS temperature

Operating Mode Applicability: 5 - Cold Shutdown, 6 - Refueling

Emergency Action Levels: (1 or 2)

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

(1) UNPLANNED increase in RCS temperature to > 200°F

(2) Loss of **all** RCS temperature AND reactor vessel/RCS level indication for ≥ 15 min. (Note 1)

Basis:

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.

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 RCS-TICA-135A/B, two safety-related thermowell-mounted hot leg wide-range temperature detectors (50°F - 700°F), one in each hot leg. The orientation of the resistance temperature detectors enables measurement of the reactor coolant fluid in the hot leg when in REDUCED INVENTORY conditions. Their range is selected to accommodate the low RCS temperatures that can be attained during shutdown. In addition, at least two incore thermocouple channels are available to measure the core exit temperature during mid-loop RNS operation (ref. 2).

EAL (1)

This IC/EAL 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 Director should refer to IC CA2.

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 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.

EAL (2)

Numerous RCS level monitoring instruments are capable of providing RCS inventory status. These include (ref. 2, 3):

- Pressurizer WR Level: An independent pressurizer level transmitter, calibrated for low temperature conditions, providing water level indication during startup, shutdown, and refueling operations in the Main Control Room and in the Remote Shutdown Workstation.
- RCS Hot Leg Level: There are two safety-related RCS hot leg level channels, one located in each hot leg. These level indicators are provided primarily to monitor the RCS water level during mid-loop operation following shutdown operations.
- Pressurizer Level Narrow Range

This IC/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 Director should also refer to IC CA2.

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 CA2 based on exceeding plant configuration-specific time criteria.

VEGP3/4 Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. UFSAR Appendix 19E Shutdown Evaluation Section 19E.2.1.2.2 RCS Instrumentation
3. APP-RCS-M3C-101 RCS Instrumentation and Packaged Mechanical System Interface Requirements

CA2

ECL: Alert

Initiating Condition: Inability to maintain the plant in cold shutdown

Operating Mode Applicability: 5 - Cold Shutdown, 6 - Refueling

Emergency Action Levels: (1 or 2)

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

(1) UNPLANNED increase in RCS temperature to > 200°F for > the duration specified in Table C-2 (Note 1)

(2) RCS pressure increase due to UNPLANNED RCS heatup > 80 psig (This EAL does **not** apply during water-solid plant conditions)

Table C-2: RCS Heat-up Duration Thresholds		
RCS Status	**Containment Closure Status**	**Heat-up Duration**
INTACT AND **not** at REDUCED INVENTORY	N/A	60 min.*
Not INTACT OR at 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.		

Basis:

CONTAINMENT CLOSURE -Technical Specification Section 3.6 contains the meaning of containment closure for Modes 5 and 6. Containment closure means that all potential escape paths are closed or capable of being closed. Since there are no requirements for containment leak tightness, compliance with Appendix J leakage criteria and test are **not** required.

RCS INTACT - The RCS can act as an effective pressure boundary.

REDUCED INVENTORY - RCS level 3 feet below the reactor vessel flange (8% wide range pressurizer level).

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.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit (200ºF) when the heat removal function is available does not warrant a classification.

EAL (1)

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).

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 RCS-TICA-135A/B, two safety-related thermowell-mounted hot leg wide-range temperature detectors (50°F - 700°F), one in each hot leg. The orientation of the resistance temperature detectors enables measurement of the reactor coolant fluid in the hot leg when in REDUCED INVENTORY conditions. Their range is selected to accommodate the low RCS temperatures that can be attained during shutdown. In addition, at least two incore thermocouple channels are available to measure the core exit temperature during mid-loop RNS operation (ref. 2, 3).

This IC/EAL addresses a condition 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.

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. 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/or 2) there is reduced reactor coolant inventory above the top of the irradiated fuel.

EAL (2)

The minimum readable RCS pressure increment on the wide range RCS pressure instruments (RCS-PICA140A/B/C/D) is 80 psig (0 - 3300 psig based on a 2.4% channel accuracy) (ref. 3).

This IC/EAL 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.

This EAL provides a pressure-based indication of RCS heat-up.

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

VEGP3/4 Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. UFSAR Appendix 19E Shutdown Evaluation Section 19E.2.1.2.2 RCS Instrumentation
3. APP-RCS-M3C-101 RCS Instrumentation and Packaged Mechanical System Interface Requirements

CU3

ECL: Unusual Event

Initiating Condition: Loss of **all** capability to charge at least one Class 1E 24-hr. DC battery for 30 minutes or longer

Operating Mode Applicability: 5 - Cold Shutdown, 6 - Refueling

Emergency Action Levels: (1)

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

(1) Loss of **all** capability to charge at least one Class 1E 24-hr. DC battery for ≥ 30 min. (Note 1)

Basis:

Motor control centers (ECS-EC-121/221) from non-Class 1E busses ECS-ES-1 and ECS-ES-2 power the battery chargers for the Class 1E safety-related batteries. If these busses are deenergized, the safety-related batteries cannot be charged. As used in this EAL the term "capability" means at least one offsite or onsite standby AC power source is either currently powering a non-Class 1E bus (ECS-ES-1 or ECS-ES-2) or is capable of energizing and powering the chargers on at least one non-Class 1E bus (ECS-ES-1 or ECS-ES-2) within 30 min. (ref. 1).

This cold condition EAL is equivalent to the hot condition loss of electrical power EAL SU1.

The Class 1E 24-hr. DC system provides electrical power for safety-related and vital control and monitoring instrumentation loads. The offsite AC power system normally supplies power for the unit in cold shutdown, refueling, and defueled conditions. Both the normal offsite and standby onsite AC power systems are non-Class 1E with no Technical Specification requirements. All safety-related functions associated with the unit in cold shutdown and refueling are provided by the safety-related onsite Class 1E DC power systems including the 120V Vital AC power system supplied from the batteries powering the safety-related DC buses through inverters. The AP1000 also has standby diesel generators that are not safety-related. Storage batteries are the safety-related power source for Class 1E electric power. However, this non-Class 1E AC power is required to charge the safety-related DC batteries and maintain long-term safety-related DC bus voltage.

Loss of Class 1E 24-hr. DC power potentially compromises all safety-related plant systems requiring electric power. The event can be classified as an Unusual Event, because the passive design affords additional and redundant means to remove heat passively or restore power to active components.

30 minutes was chosen to allow sufficient time for plant personnel to attempt to establish a viable offsite or diesel generator AC power supply to the plant-specific AC power buses required to charge one or more Class 1E 24-hr. DC batteries.

Escalation of the emergency classification level would be via IC CA3 if the Class 1E battery voltage drops below 210 VDC.

VEGP3/4 Basis Reference(s):

1. UFSAR Chapter 8 Electric Power

CA3

ECL: Alert

Initiating Condition: Loss of **all** required Class 1E DC power or Class 1E UPS bus power for 15 minutes or longer

Operating Mode Applicability: 5 - Cold Shutdown, 6 - Refueling

Emergency Action Levels: (1 or 2)

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

(1) Indicated voltage < 210 VDC on **all** required Class 1E 24-hr. DC buses (IDSA/B/C/D-DS-1) for ≥ 15 min. (Note 1)

(2) Loss of power to **all** 24-hr. Class 1E UPS buses (IDSA/B/C/D-EA-1) for ≥ 15 min. (Note 1)

Basis:

<u>EAL (1)</u>

Minimum Class 1E 24-hr. DC bus voltage of 210 VDC is based on a minimum of 1.75 V per battery cell (ref. 1).

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

The purpose of this IC/EAL and its associated EALs is to recognize a loss of the Class 1E 24-hr. DC, which provides electrical power for safety-related and vital control and monitoring instrumentation loads. 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.

Routine maintenance on a division related basis is performed during shutdown periods. It is intended that the loss of the operating (operable) divisions is to be considered. If this loss results in the inability to maintain cold shutdown, then the Alert criteria of CA2 - Inability to Maintain Plant in Cold Shutdown would also be met in addition to the CA3 Alert criteria.

The plant-specific minimum bus voltage is the minimum bus voltage necessary for the operation of safety-related instrumentation and controls. This voltage value incorporates a margin significantly longer than the allowed 15 minutes of operation before the onset of inability to operate those loads. Permanent battery cell damage begins to occur at battery voltage < 210 VDC.

<u>EAL (2)</u>

24-hr. Class 1E UPS buses IDSA-EA-1, IDSB-EA-1, IDSC-EA-1, and IDSD-EA-1 provide instrument and control power to Class 1E SAFETY RELATED equipment and systems (ref. 1, 2).

This EAL is the cold condition equivalent of the hot condition loss of all AC power EAL SS1.2.

The purpose of this IC/EAL and its associated EALs is to recognize a loss of the Class 1E 24-hr. UPS which provides electrical power for safety-related and vital control and monitoring instrumentation loads. 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.

Routine maintenance on a division related basis is performed during shutdown periods. It is intended that the loss of the operating (operable) divisions is to be considered. If this loss results in the inability to maintain cold shutdown, then the Alert criteria of CA2 - Inability to Maintain Plant in Cold Shutdown would also be met in addition to the CA3 Alert criteria.

This threshold addresses an event that results in de-energizing all Class 1E UPS control power busses. This condition would result in degraded capability to monitor and control the unit from the Main Control Room resulting in the need for additional personnel to manage the event.

VEGP3/4 Basis Reference(s):

1. UFSAR Table 8.3.2-5 Component Data – Class 1E DC System (Nominal Values)
2. UFSAR Section 8.3.2.1.1.2 Class 1E Uninterruptible Power Supplies

CU4

ECL: Unusual Event

Initiating Condition: Extended loss of **all** offsite and **all** onsite AC power

Operating Mode Applicability: 5 - Cold Shutdown, 6 – Refueling, Def - Defueled

Emergency Action Levels: (1)

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

(1) Extended loss of **all** offsite and **all** onsite AC power to buses ECS-ES-1 and ECS-ES-2 for > 2 hrs. (Note 1)

Basis:

The AC power system is a non-Class 1E system comprised of normal, preferred and maintenance offsite power supplies and standby (diesel generators) onsite power supplies. The normal, preferred, and maintenance offsite power supplies are included in the main AC power system. The standby power is included in the onsite standby power system (ref. 1, 2).

An extended loss of all AC power compromises non-safety systems requiring electric power including RCS heat removal, Spent Fuel Pool heat removal and the inventory makeup portion of the CVS system.

The event is classified as an Unusual Event when in cold shutdown, refueling, or defueled based on significantly reduced decay heat and lower temperature and pressure, increasing the time to restore one of the buses.

In the event of total loss of offsite or onsite standby AC power sources, the system stationary batteries constitute the source of electric power for operation of the Non-Class 1E DC and UPS loads for at least 2 hours.

VEGP3/4 Basis Reference(s):

1. UFSAR Section 8.3 Onsite Power Systems
2. UFSAR Table 8.3.1-3 Component Data – Main AC Power System (Nominal Values)

CU5

ECL: Unusual Event

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

Operating Mode Applicability: 5 - Cold Shutdown, 6 – Refueling, Def - Defueled

Emergency Action Levels: (1 or 2 or 3)

(1) Loss of **all** Table S-1 onsite communication methods

(2) Loss of **all** Table S-1 ORO communication methods

(3) Loss of **all** Table S-1 NRC communication methods

Table C-3 Communication Methods

System	Onsite	ORO	NRC
Plant Page	X		
VEGP Dial	X		
Dedicated Dial	X		
In-Plant Radio	X		
Commercial Phones	X	X	X
Southern Company Communications	X	X	X
ENN		X	
ENS			X

Basis:

The Table C-3 list for onsite communications loss encompasses the loss of all means of routine communications (e.g., commercial and internal telephones, page system, and radios) (ref. 1, 2, 3).

The Table C-3 list for offsite (ORO) communications loss encompasses the loss of all means of communications with offsite authorities (e.g., dedicated and commercial telephone lines).

The Table C-3 list for NRC communications loss includes the FTS (ENS) and commercial telephone lines (ref. 1, 2, 3).

This EAL is the cold condition equivalent of the hot condition EAL SU6.

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 and third EAL conditions address a total loss of the communications methods used to notify all offsite organizations of an emergency declaration. The OROs referred to here are the State of Georgia and South Carolina, Burke, Aiken, Lexington and Barnwell County EOCs as well as the NRC.

VEGP3/4 Basis Reference(s):

1. UFSAR 9.5.2 Communication System
2. SNC Standard Emergency Plan – Section F
3. VEGP Units 3 and 4 Annex Table 5.2.A Emergency Response Communications

CU6

ECL: Unusual Event

Initiating Condition: UNPLANNED partial loss of monitoring or control functions for 15 minutes or longer

Operating Mode Applicability: 5 - Cold Shutdown, 6 – Refueling, Def - Defueled

Emergency Action Levels: (1)

(1) UNPLANNED loss of the ability to monitor or control one or more of the following key safety functions from the Main Control Room required for the current plant operating mode for ≥ 15 min.:

- Decay Heat Removal
- RCS Inventory Control
- Reactivity Control
- Spent Fuel Pool Level and Temperature Control

Basis:

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.

This EAL is the cold condition equivalent of the hot condition EAL SA2.

This IC recognizes the difficulty associated with monitoring and controlling changing plant conditions without the use of a major portion of the control and indication systems. A Notification of Unusual Event level is considered appropriate for this loss of monitoring and control IC due to the inherently safer condition of the core when in the cold condition. The selection of 15 minutes was chosen to allow personnel sufficient time for restoration of required systems in the Main Control Room due to an inadvertent loss.

Systems that provide monitoring and control capability include:

- The Protection and Safety Monitoring System (PMS) provides the functions necessary to protect the plant during normal operations, to shut down the plant, and to maintain the plant in a safe shutdown condition.

- The Plant Control System (PLS) is a non-safety-related system and includes the control functions that provide for the control of the nuclear process, conversion of nuclear energy into heat energy, and transport of the heat energy from the nuclear reactor to the main steam turbine.
- The Data Display and Processing System (DDS) (Plant Computer) consists of a set of graphics workstations that obtain their inputs from real-time data networks and deliver their output to the network for PLANT OPERATORS and other users.
- The Diverse Actuation System (DAS) is a non-safety-related system available to ensure monitoring and control capability as well as providing an alternative means of initiating a reactor trip and selected engineering safety features.

VEGP3/4 Basis Reference(s):

1. UFSAR Section 7.1.1 The AP1000 Instrumentation and Control Architecture

CA7

ECL: Alert

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

Operating Mode Applicability: 5 - Cold Shutdown, 6 - Refueling

Emergency Action Levels: (1)

(1) The occurrence of **any** Table C-4 hazardous event

AND

EITHER of the following:

- 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-4 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

Basis:

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 safe 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.

- The significance of seismic events are discussed under EAL HU2.1 (ref. 1, 5).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2).
- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 145 mph (three second gust). (ref. 3, 5).
- UFSAR Section 1.2.3 Plant Arrangement Description to identify major plant structures containing functions and systems required for safe shutdown of the plant (ref. 4)
- An EXPLOSION (including a steam line 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. The need to classify a steam line break **not** considered an explosion itself is considered in fission product barrier degradation monitoring (EAL Category F).

This IC/EAL 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 division 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 division.

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.

Some systems have a mix of equipment that is SAFETY-RELATED and non-safety-related. Example: The safety-related function of ECS is to trip open RCP circuit breakers on a CMT signal. Damage to these breakers would constitute VISIBLE DAMAGE to a SAFETY SYSTEM. However, damage limited to ECS-ES-1 or a Standby Diesel Generator would not meet the intent of VISIBLE DAMAGE to a SAFETY SYSTEM.

Escalation of the ECL would be via IC CS1 or RS1.

VEGP3/4 Basis Reference(s):

1. UFSAR Section 3.7 Seismic Design
2. UFSAR Section 3.4 Water Level (Flood) Design
3. UFSAR Section 3.3 Wind and Tornado Loadings
4. UFSAR Section 1.2.3 Plant Arrangement Description
5. 3(4)-AOP-901 Acts of Nature

Category H – Hazards and Other Conditions Affecting Plant Safety

EAL Group: ALL (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.

The events of this category pertain to the following subcategories:

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 Technological 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 and EXPLOSIONS can pose significant hazards to personnel and reactor safety. Appropriate for classification are FIRES or EXPLOSIONS within the site 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. Main Control Room Evacuation

Events that are indicative of loss of Main Control Room habitability. If the Main Control Room must be evacuated, additional support for monitoring and controlling plant functions is necessary through the emergency response facilities.

7. ED 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 Director the latitude to classify emergency conditions consistent with the established classification criteria based upon Emergency Director judgment.

HU1

ECL: Unusual Event

Initiating Condition: Confirmed SECURITY CONDITION or threat

Operating Mode Applicability: All

Emergency Action Levels: (1 or 2 or 3)

(1) A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by Security Team Leader

(2) Notification of a credible security threat directed at the site

(3) A validated notification from the NRC providing information of an aircraft threat

Basis:

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 VEGP 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 VEGP. 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).

If the Security Team Leader determines that a threat notification is credible, the Security Team Leader will notify the Shift Manager that a “Credible Threat” condition exists for VEGP. Generally, VEGP procedures address standard practices for determining credibility. The three main criteria for determining credibility are: technical feasibility, operational feasibility, and resolve. For VEGP, a validated notification delivered by the FBI, NRC or similar agency is treated as credible (ref. 1, 2).

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 Main Control Room is essential for proper classification of a security-related event (ref. 2).

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]*.

<u>EAL (1)</u>

The first threshold references the Security Team Leader 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 2.390 information.

<u>EAL (2)</u>

The second threshold addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the VEGP Security Plan (ref. 1).

<u>EAL (3)</u>

The third threshold 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 the VEGP Security Plan (ref. 1).

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 Security Plan.

Escalation of the ECL would be via IC HA1.

VEGP3/4 Basis Reference(s):

1. VEGP Security Plan
2. 3(4)-AOP-904 Security Events

HA1

ECL: Alert

Initiating Condition: HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

Operating Mode Applicability: All

Emergency Action Levels: (1 or 2)

(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Team Leader

(2) A validated notification from NRC of an aircraft attack threat within 30 min. of the site

Basis:

HOSTILE ACTION - An act toward VEGP 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 VEGP. 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 between the vehicle barrier system and the PROTECTED AREA barrier.

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 the Security Team Leader and the Main Control Room is essential for proper classification of a security-related event (ref. 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 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.

EAL (1)

The first threshold is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA.

EAL (2)

The second threshold 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 3(4)-AOP-904 Security Events (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 Security Plan.

Escalation of the ECL would be via IC HS1.

VEGP3/4 Basis Reference(s):

1. VEGP Security Plan
2. 3(4)-AOP-904 Security Events

HS1

ECL: Site Area Emergency

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA

Operating Mode Applicability: All

Emergency Action Levels: (1)

(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Team Leader

Basis:

HOSTILE ACTION - An act toward VEGP 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 VEGP. 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 - An area, located within the Site Exclusion Area Boundary, encompassed by physical barriers and to which access is controlled per 10 CFR 73.55. The SNC Power Block Protected Area and the ISFSI Protected Area are two Protected Areas located within the Site Owner Controlled Area.

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 Team Leader and the Main Control Room is essential for proper classification of a security-related event (ref. 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 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 VEGP Security Plan (ref. 1).

Escalation of the ECL would be via IC HG1.

VEGP3/4 Basis Reference(s):

1. VEGP Security Plan
2. 3(4)-AOP-904 Security Events

HG1

ECL: General Emergency

Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility

Operating Mode Applicability: All

Emergency Action Levels: (1)

(1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Team Leader

AND EITHER of the following has occurred:

Any of the following safety functions **cannot** be controlled or maintained:

- Reactivity control
- Core cooling
- RCS heat removal

OR

Damage to spent fuel has occurred or is IMMINENT

Basis:

HOSTILE ACTION - An act toward VEGP 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 VEGP. 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 - An area, located within the Site Exclusion Area Boundary, encompassed by physical barriers and to which access is controlled per 10 CFR 73.55. The SNC Power Block Protected Area and the ISFSI Protected Area are two Protected Areas located within the Site 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.

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 Security Team Leader and the Main Control Room is essential for proper classification of a security-related event (ref. 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]*.

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 VEGP Security Plan (ref. 1).

VEGP3/4 Basis Reference(s):

1. VEGP Security Plan
2. 3(4)-AOP-904 Security Events

HU2

ECL: Unusual Event

Initiating Condition: Seismic event greater than OBE levels

Operating Mode Applicability: All

Emergency Action Levels: (1)

(1) Seismic event greater than Operating Basis Earthquake (OBE) as indicated by VALID SJS-JS-01-ALM2 (OBE Exceedance) alarm

Basis:

This IC/EAL addresses a seismic event that results in accelerations at the plant site greater than the equivalent of an Operating Basis Earthquake (OBE). For the AP1000, the OBE equivalent is defined to be 1/3 the SSE (0.30g) ground acceleration, or 0.10 g in either the horizontal or vertical plane (ref. 2). Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event, also triggering the Seismic Monitoring System (SJS) recorders and the SJS-JS-01-ALM2 – “OBE Exceedance” alarm.

AP1000 instrumentation used to indicate a seismic event greater than the OBE includes four triaxial acceleration sensor units attached to a time-history analyzer recording and playback system. When ground acceleration is sensed above the initiation setpoint, initially set at 0.01 g (ref. 1), the system initiates recording, and provides alarms in the Main Control Room. Additionally, an alarm (SJS-JS-01-ALM2 – OBE Exceedance) is generated in the Main Control Room if an OBE (0.10 g) is detected. If the Seismic Monitoring System (SJS) is functional, SJS-JS-01-ALM2 provides clear indication when the EAL threshold has been exceeded.

If the SJS or its associated alarms are non-functional, then the Shift Manager or Emergency Director should seek external confirmation of the seismic event through an external source such as the USGS or internet news service if a seismic event is felt in the MCR. The verification action must not preclude a timely emergency declaration. The USGS, National Earthquake Information Center (NEIC) can confirm that an earthquake has occurred in the area of the plant. Alternatively, near real-time seismic activity can be accessed via the NEIC website.

The NEIC can be contacted by going to the USGS NEIC website:

http://earthquake.usgs.gov/eqcenter/

or by calling (303) 273-8500 (normal hours), or (303) 273-8428 (off normal hours). Select option #1 and then option #2 and inform the analyst you wish to confirm recent seismic activity in the vicinity of Plant Vogtle. The following VEGP coordinates: **33 deg. 08 min. 30 sec. north latitude, 81 deg. 45 min. 44 sec. west longitude** (ref. 3) may be provided to gain more localized readings on the earthquake. Alternatively, near real-time seismic activity can be accessed via the NEIC website.

An earthquake greater than the equivalent of 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.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA7 or SA8

VEGP3/4 Basis Reference(s):

1. UFSAR Section 3.7.4 Seismic Instrumentation
2. UFSAR Section 2.5.2 Vibratory Ground Motion
3. UFSAR Section 2.1.1.1 Site Location
4. 3(4)-AOP-901 Acts of Nature

HU3

ECL: Unusual Event

Initiating Condition: Hazardous event

Operating Mode Applicability: All

Emergency Action Levels: (1 or 2 or 3 or 4)

Note 8: This EAL does **not** apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

(1) A tornado strike within the PROTECTED AREA

(2) 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

(3) 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)

(4) A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 8)

Basis:

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.

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 - An area, located within the Site Exclusion Area Boundary, encompassed by physical barriers and to which access is controlled per 10 CFR 73.55. The SNC Power Block Protected Area and the ISFSI Protected Area are two Protected Areas located within the Site Owner Controlled Area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the safe 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.

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

EAL (1)

EAL HU3.1 addresses a tornado striking (touching down) within the PROTECTED AREA (ref. 1).

EAL (2)

Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2).

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.

EAL (3)

As used here the term "offsite" is meant to be areas external to the VEGP3/4 PROTECTED AREA, including a hazardous release on VEGP1/2.

94001-C Implementation of the Spill Prevention, Control, Countermeasures (SPCC) Plan and Reportability (ref. 3) provides additional information on hazardous substances and spills.

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.

EAL (4)

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 ECL would be based on ICs in Recognition Categories R, F, S or C.

VEGP3/4 Basis Reference(s):

1. 3(4)-AOP-901 Acts of Nature
2. UFSAR Section 3.4 Water Level (Flood) Design
3. 94001-C Implementation of the Spill Prevention, Control, Countermeasures (SPCC) Plan and Reportability

HU4

ECL: Unusual Event

Initiating Condition: FIRE potentially degrading the level of safety of the plant

Operating Mode Applicability: All

Emergency Action Levels: (1 or 2 or 3 or 4)

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

(1) 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

Table H-1 Fire Areas
• Containment Building • Shield Building • Auxiliary Building

(2) 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)

(3) A FIRE within the plant PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm, or indication (Note 1)

(4) A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

Basis:

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 - An area, located within the Site Exclusion Area Boundary, encompassed by physical barriers and to which access is controlled per 10 CFR 73.55. The SNC Power Block Protected Area and the ISFSI Protected Area are two Protected Areas located within the Site Owner Controlled Area.

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.

EAL (1)

UFSAR Section 1.2.3 Plant Arrangement Description was used to identify areas (Table H-1) containing SAFETY SYSTEM equipment required for accident mitigation and safe shutdown of the plant (ref. 1).

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 fire alarms or indications. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial multiple alarms or indications were received, field verification of a single alarm was received or report from the field was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock runs concurrently with the emergency declaration clock.

EAL (2)

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 verification 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.

EAL (3)

This EAL separates itself from HU4.1 and HU4.2 by evaluating a fire anywhere within the PROTECTED AREA (not restricted to Table H-1 Areas), which may degrade 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.

EAL (4)

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 ECL would be via IC CA7 or SA8.

VEGP3/4 Basis Reference(s):

1. UFSAR Section 1.2.3 Plant Arrangement Description

HA5

ECL: Alert

Initiating Condition: Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

Operating Mode Applicability: All

Emergency Action Levels: (1)

Note 5:	If the equipment in the listed area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.
Note 9:	Classification is not required if either train of RNS can be placed in service for Shutdown Cooling.

(1) Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 area/room

AND

Entry into the area is prohibited or IMPEDED (Note 5)

Table H-2 Safe Operation & Shutdown Areas/Rooms

Building	Room Number	Room/Area Name	Applicable Mode
Auxiliary	12152	Primary Sample Room	3
	12252	Rad Chem Lab	3
	12321	Non-1E Equipment / Penetration Room	4
	12305	Division D I&C Penetration Room	4
	12304	Division B I&C Penetration Room	4
	12561	CCS Valve Room (Note 9)	4
Annex	4000	Annex Building (Note 9)	4
Containment	11209	CVS Room	4

Basis:

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).

The Table H-2 safe operation and shutdown areas/rooms (with entry-related mode applicability) are those plant areas that contain equipment which require a manual/local action as specified in general operating procedures (and procedures referenced by them) used for normal plant operation, cooldown and shutdown. The list specifies the plant operating modes during which entry would be required for each area and thus specifying when a loss of access or IMPEDED access is applicable to this EAL (ref. 1).

Plant areas where actions of a contingent or emergency nature might be needed to be performed. (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) were not considered for inclusion. Additionally, areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections) were not considered for inclusion. Refer to Attachment 3 "Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases.".

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.

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 Director’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 ECL would be via Recognition Category R, C or F ICs.

VEGP3/4 Basis Reference(s):

1. NMP-EP-110-GL04 VEGP 3/4 EAL – ICs, Threshold Values and Basis, Attachment 3 "Safe Operation & Shutdown Areas/Rooms Tables R-1 & H-2 Bases."

HA6

ECL: Alert

Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations

Operating Mode Applicability: All

Emergency Action Levels: (1)

(1) An event has resulted in plant control being transferred from the Main Control Room to the Remote Shutdown Workstation

Basis:

If temporary evacuation of the Main Control Room is required because of some abnormal Main Control Room condition, the operators can establish and maintain safe shutdown conditions for the plant from outside the Main Control Room through the use of controls and monitoring located at the Remote Shutdown Workstation (ref. 1).

This IC addresses an evacuation of the Main Control Room that results in transfer of plant control to alternate locations outside the Main Control Room. The loss of the ability to control the plant from the Main Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Main Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Main Control Room, in addition to responding to the event that required the evacuation of the Main 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 ECL would be via IC HS6.

VEGP3/4 Basis Reference(s):

1. UFSAR Section 7.4.3 Safe Shutdown from Outside the Main Control Room

HS6

ECL: Site Area Emergency

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

Operating Mode Applicability: All

Emergency Action Levels: (1)

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

(1) An event has resulted in plant control being transferred from the Main Control Room to the Remote Shutdown Workstation

<u>AND</u>

Control of **any** of the following key safety functions is **not** reestablished within 15 min. (Note 1):

- Reactivity control
- Core cooling
- RCS heat removal

Basis:

If temporary evacuation of the Main Control Room is required because of some abnormal Main Control Room condition, the operators can establish and maintain safe shutdown conditions for the plant from outside the Main Control Room through the use of controls and monitoring located at the remote shutdown workstation (ref. 1).

This IC addresses an evacuation of the Main 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 Director judgment. The Emergency Director 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 Shutdown Room/Workstation.

Escalation of the ECL would be via IC FG1 or CG1.

VEGP3/4 Basis Reference(s):

1. UFSAR Section 7.4.3 Safe Shutdown from Outside the Main Control Room

HU7

ECL: Unusual Event

Initiating Condition: Other conditions existing that in the judgment of the Emergency Director warrant declaration of a UE

Operating Mode Applicability: All

Emergency Action Levels: (1)

(1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Basis:

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant, and/or placing it in the safe 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.

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 Director to fall under the ECL description for an Unusual Event.

VEGP3/4 Basis Reference(s):

None

HA7

ECL: Alert

Initiating Condition: Other conditions exist that in the judgment of the Emergency Director warrant declaration of an Alert

Operating Mode Applicability: All

Emergency Action Levels: (1)

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

Basis:

HOSTILE ACTION - An act toward VEGP 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 VEGP. 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).

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 Director to fall under the ECL description for an Alert.

VEGP3/4 Basis Reference(s):

None

HS7

ECL: Site Area Emergency

Initiating Condition: Other conditions existing that in the judgment of the Emergency Director warrant declaration of a Site Area Emergency

Operating Mode Applicability: All

Emergency Action Levels: (1)

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

Basis:

HOSTILE ACTION - An act toward VEGP 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 VEGP. 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 - The boundary of a reactor site beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

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

VEGP3/4 Basis Reference(s):

None

HG7

ECL: General Emergency

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency

Operating Mode Applicability: All

Emergency Action Levels: (1)

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

Bases:

HOSTILE ACTION - An act toward VEGP 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 VEGP. 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.

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 Director to fall under the ECL description for a General Emergency.

VEGP3/4 Basis Reference(s):
None

Category S – System Malfunction

EAL Group: HOT Conditions (RCS temperature > 200ºF); 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 Electrical 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 Class 1E DC plant or UPS electrical power.

2. Loss of Monitoring and Control Functions

Loss of monitoring and control functions can challenge the Main Control Room staff ability to maintain SAFETY SYSTEM operability.

3. RCS Activity

During normal operation, reactor coolant fission product activity is very low (fuel defect level < 0.25%). 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 is indicative of fuel failures (2% - 5% clad damage) and is covered under Category F, Fission Product Barrier Degradation. 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.

4. RCS Leakage

The reactor vessel provides a volume for the coolant that covers the reactor core. The reactor 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.

5. RTS Failure

This subcategory includes events related to failure of the Reactor Trip System (RTS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RTS to complete a reactor trip comprise a specific set of analyzed events referred to as Anticipated Transient Without Trip (ATWS) events. For EAL classification however, ATWS is intended to mean any trip failure event that does not achieve reactor shutdown. If RTS 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.

6. 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.

7. Containment Isolation or Pressure Control Failure

Failure of containment isolation capability (under conditions in which the containment is not currently challenged) or failure of Passive Containment Cooling (PCS) warrants emergency classification.

8. 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 the sub-category.

SU1

ECL: Unusual Event

Initiating Condition: Loss of **all** capability to charge at least one Class 1E 24-hr. DC battery for 30 minutes or longer

Operating Mode Applicability: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Safe Shutdown

Emergency Action Levels: (1)

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

(1) Loss of **all** capability to charge at least one Class 1E 24-hr. DC battery for ≥ 30 min. (Note 1)

Bases:

Motor control centers (ECS-EC-121/221) from non-Class 1E busses ECS-ES-1 and ECS-ES-2 power the battery chargers for the Class 1E safety-related batteries. If these busses are deenergized, the safety-related batteries cannot be charged. As used in this EAL the term "capability" means at least one offsite or onsite standby AC power source is either currently powering a non-Class 1E bus (ECS-ES-1 or ECS-ES-2) or is capable of energizing and powering the chargers on at least one non-Class 1E bus (ECS-ES-1 or ECS-ES-2) within 30 min. (ref. 1)

This hot condition EAL is equivalent to the cold condition loss of Class 1E DC or UPS power EAL CU3.

The Class 1E 24-hr. (safety-related) DC system provides electrical power for safety-related and vital control and monitoring instrumentation loads. The offsite AC power system normally supplies power for the unit in cold shutdown, refueling, and defueled conditions. Both the normal offsite and standby onsite AC power systems are non-Class 1E with no Technical Specification requirements. All safety-related functions associated with the unit in cold shutdown and refueling are provided by the safety-related onsite Class 1E DC power systems including the 120V Vital AC power system supplied from the batteries powering the safety-related DC buses through inverters. The Passive ALWRs also have standby diesel generators that are not safety-related. Storage batteries are the safety-related power source for Class 1E electric power. However, this non-Class 1E AC power is required to charge the safety-related DC batteries and maintain long-term safety-related DC bus voltage.

Loss of Class 1E 24-hr. DC power potentially compromises all safety-related plant systems requiring electric power. The event can be classified as an Unusual Event, because the passive design affords additional and redundant means to remove heat passively or restore power to active components.

30 minutes was chosen to allow sufficient time for plant personnel to attempt to establish a viable offsite or diesel generator AC power supply to the plant-specific AC power buses required to charge one or more Class 1E 24-hr. DC batteries.

Escalation of the emergency classification level would be via IC SA1 if at least one Class 1E battery cannot be charged within 60 min.

VEGP3/4 Basis Reference(s):

1. UFSAR Chapter 8 Electric Power

SA1

ECL: Alert

Initiating Condition: Loss of **all** capability to charge at least one Class 1E 24-hr. DC battery for 60 minutes or longer

Operating Mode Applicability: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Safe Shutdown

Emergency Action Levels: (1)

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

(1) Loss of **all** capability to charge at least one Class 1E 24-hr. DC battery for ≥ 60 min. (Note 1)

Bases:

Motor control centers (ECS-EC-121/221) from non-Class 1E busses ECS-ES-1 and ECS-ES-2 power the battery chargers for the Class 1E safety-related batteries. If these busses are deenergized, the safety-related batteries cannot be charged. As used in this EAL the term "capability" means at least one offsite or onsite standby AC power source is either currently powering a non-Class 1E bus (ECS-ES-1 or ECS-ES-2) or is capable of energizing and powering the chargers on at least one non-Class 1E bus (ECS-ES-1 or ECS-ES-2) within 60 min. (ref. 1).

This IC/EAL and the associated threshold is intended to provide an escalation from IC SU1. Prolonged de-energization of the busses that provide power to the Class 1E 24-hr battery chargers degrades the level of safety of the plant by rendering the plant more vulnerable to a complete loss of DC power even though no operator action is required for 72 hours.

60 minutes was selected to allow power restoration procedures to be effective.

The Class 1E (safety-related) 24-hr DC system provides electrical power for Class 1E motor operated valves and vital control and monitoring instrumentation loads. The offsite AC power system normally supplies power for the unit in cold shutdown, refueling, and defueled conditions. Both the normal offsite and standby onsite AC power systems are non-Class 1E with no Technical Specification requirements. The Passive ALWRs also have standby diesel generators that are not safety-related. Storage batteries are the safety-related power source for Class 1E electric power. However, this non-Class 1E AC power is required to charge the safety-related DC batteries and maintain long-term safety-related DC bus voltage.

Escalation of the emergency classification level would be via IC SS1.

VEGP3/4 Basis Reference(s):

1. UFSAR Chapter 8 Electric Power

SS1

ECL: Site Area Emergency

Initiating Condition: Loss of **all** required Class 1E DC power or Class 1E UPS bus power for 15 minutes or longer

Operating Mode Applicability: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Safe Shutdown

Emergency Action Levels: (1 or 2)

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

(1) Indicated voltage < 210 VDC on **all** required Class 1E 24-hr. DC buses (IDSA/B/C/D-DS-1) for ≥ 15 min. (Note 1)

(2) Loss of power to **all** 24-hr. Class 1E UPS buses (IDSA/B/C/D-EA-1) for ≥ 15 min. (Note 1)

Bases:

<u>EAL (1)</u>

Minimum Class 1E 24-hr. DC bus voltage of 210 VDC is based on a minimum of 1.75 V per battery cell (ref. 1).

This EAL is the hot condition equivalent of the cold condition Loss of Class 1E DC or UPS Power EAL CA3.

Loss of all 24-hr. DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system.

This IC/EAL addresses a total loss of Class 1E 24-hr. DC 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.

210 VDC is the minimum bus voltage necessary for the operation of safety-related instrumentation and controls. This voltage value incorporates a margin significantly longer than the allowed 15 minutes of operation before the onset of inability to operate those loads.

<u>EAL (2)</u>

24-hr. Class 1E UPS buses IDSA-EA-1, IDSB-EA-1, IDSC-EA-1, and IDSD-EA-1 provide instrument and control power to Class 1E SAFETY RELATED equipment and systems (ref. 1).

This EAL is the hot condition equivalent of the cold condition Loss of Class 1E DC or UPS Power EAL CA3.

Loss of all 24-hr. UPS power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system.

This IC/EAL addresses a total loss of Class 1E 24-hr. DC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control and spent fuel heat removal. 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.

This threshold addresses an event that results in de-energizing all UPS busses. This condition would result in degraded capability to monitor and control the unit from the Main Control Room resulting in the need for additional personnel to manage the event.

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 or FG1.

VEGP3/4 Basis Reference(s):

1. UFSAR Chapter 8 Electric Power

SA2

ECL: Alert

Initiating Condition: UNPLANNED partial loss of monitoring or control functions for 15 minutes or longer

Operating Mode Applicability: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Safe Shutdown

Emergency Action Levels: (1)

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

(1) UNPLANNED loss of the ability to monitor or control one or more of the following key safety functions from the Main Control Room required for the current plant operating mode for $\geq$ 15 min. (Note 1):

- Reactivity Control
- Core Cooling
- RCS Heat Removal
- Spent Fuel Pool Level and Temperature Control

Bases:

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.

This EAL is the hot condition equivalent of the cold condition EAL CU6.

This IC recognizes the difficulty associated with monitoring and controlling changing plant conditions without the use of a major portion of the control and indication systems under hot conditions.

The selection of 15 minutes was chosen to allow personnel sufficient time for restoration of required systems in the Main Control Room due to an inadvertent loss.

Systems that provide monitoring and control capability include:

- The Protection and Safety Monitoring System (PMS) provides the functions necessary to protect the plant during normal operations, to shut down the plant, and to maintain the plant in a safe shutdown condition.

- The Plant Control System (PLS) is a non-safety-related system and includes the control functions that provide for the control of the nuclear process, conversion of nuclear energy into heat energy, and transport of the heat energy from the nuclear reactor to the main steam turbine.
- The Data Display and Processing System (DDS) (Plant Computer) consists of a set of graphics workstations that obtain their inputs from real-time data networks and deliver their output to the network for PLANT OPERATORS and other users.
- The Diverse Actuation System (DAS) is a non-safety-related system available to ensure monitoring and control capability as well as providing an alternative means of initiating a reactor trip and selected engineering safety features.

VEGP3/4 Basis Reference(s):

1. UFSAR Section 7.1.1 The AP1000 Instrumentation and Control Architecture

SS2

ECL: Site Area Emergency

Initiating Condition: Complete loss of monitoring or control functions for 15 minutes or longer

Operating Mode Applicability: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Safe Shutdown

Emergency Action Levels: (1)

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

(1) Loss of ability to monitor or control **all** of the following key safety functions from the Main Control Room for $\geq$ 15 min. (Note 1):

- Reactivity Control
- Core Cooling
- RCS Heat Removal
- Spent Fuel Pool Level and Temperature Control

Bases:

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.

This IC/EAL recognizes the inability of the Main Control Room staff to monitor and control the plant from the Main Control Room due to loss of normal and safety indication and monitoring systems, and diverse indication and control systems that allow the operators to monitor and safely shutdown the plant. A Site Area Emergency is considered to exist if the Main Control Room staff cannot monitor and control safety functions needed for protection of the public. The loss of ability to monitor or control key safety functions following a Main Control Room evacuation would be classified under HS6.1.

This IC/EAL recognizes the challenge to the Main Control Room staff to monitor and control the plant due to a complete loss of normal and safety indication and monitoring systems. The selection of 15 minutes was chosen to allow personnel sufficient time for restoration of required systems due to an inadvertent loss.

Systems that provide monitoring and control capability include:

- The Protection and Safety Monitoring System (PMS) provides the functions necessary to protect the plant during normal operations, to shutdown the plant, and to maintain the plant in a safe shutdown condition.
- The Plant Control System (PLS) is a nonsafety-related system and includes the control functions that provide for the control of the nuclear process, conversion of nuclear energy into heat energy, and transport of the heat energy from the nuclear reactor to the main steam turbine.
- The Data Display and Processing System (DDS) (Plant Computer) is comprised of a set of graphics workstations that obtains its inputs from real-time data networks and delivers its output to the network for PLANT OPERATORS and other users.
- The Diverse Actuation System (DAS) is a nonsafety-related system available to ensure monitoring and control capability as well as providing an alternative means of initiating a reactor trip and selected engineering safety features.

Escalation of the emergency classification level would be via ICs FG1 or IC RG1.

VEGP3/4 Basis Reference(s):

1. UFSAR Section 7.1.1 The AP1000 Instrumentation and Control Architecture

SU3

ECL: Unusual Event

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

Operating Mode Applicability: 1 - Power Operation, 2 - Startup, 3 - Hot Standby (≥ 500 ºF)

Emergency Action Levels: (1 or 2)

(1) Primary Sampling Liquid Radiation Monitor (PSS-RY050) high alarm

(2) Sample analysis indicates that a reactor coolant activity value > than an allowable limit specified in Technical Specifications (TS 3.4.10)

Bases:

This IC/EAL addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications (ref. 3). This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Mode applicability has been limited to Modes 1, 2 and 3 (≥ 500 ºF) consistent with applicability of the coolant activity Technical Specification limits.

EAL (1)

The primary sampling system (PSS) liquid sample radiation monitor (PSS-RY050) measures and indicates the concentration of radioactive materials in the samples from the reactor coolant system. The liquid sample radiation monitor's primary function is to indicate elevated sample radiation levels following a design basis or severe accident. The monitor may be used to provide early indication of a significant increase in the radioactivity of the reactor coolant indicating a possible fuel cladding breach. When a predetermined setpoint is exceeded, the primary sampling system liquid sample radiation monitor isolates the sample flow by closing the outside containment isolation valve and initiates an alarm in the Main Control Room and locally to alert the Operator (ref. 1, 2).

EAL (2)

The RCS specific activity LCO limits the allowable concentration of iodines and noble gases in the reactor coolant. The LCO limits are established to be consistent with a fuel defect level of 0.25 percent and to ensure that plant operation remains within the conditions assumed for shielding and Design Basis Accident (DBA) release analyses (ref. 3).

The LCO contains specific activity limits for both dose equivalent I-131 and dose equivalent XE-133. The allowable levels are intended to limit the doses due to postulated accidents to within the values calculated in the radiological consequences analyses (ref. 4).

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

VEGP3/4 Basis Reference(s):

1. UFSAR Section 11.5.2.3.1 Fluid Process Monitors
2. UFSAR Table 11.5-2 Area Radiation Monitor Detector Parameters
3. Technical Specifications Section 3.4.10 RCS Specific Activity
4. UFSAR Chapter 15 Accident Analysis

SU4

ECL: Unusual Event

Initiating Condition: RCS leakage for 15 minutes or longer

Operating Mode Applicability: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Safe Shutdown

Emergency Action Levels: (1 or 2 or 3)

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

(1) RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min.

(2) RCS identified leakage > 25 gpm for ≥ 15 min.

(3) Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min.
(Note 1)

Bases:

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS leakage into and external to the containment area is necessary. Quickly separating the identified leakage from the unidentified leakage is necessary to provide quantitative information to the Operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public (ref. 1).

This IC/EALs 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 EAL condition addresses a RCS mass loss not considered either unidentified or identified leakage caused by an UNISOLABLE leak through an interfacing system external to the containment. These EALs 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 EAL were selected because they are usually observable with normal Main 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.

VEGP3/4 Basis Reference(s):

1. Technical Specifications Section 3.4.7 RCS Operational Leakage

SU5

ECL: Unusual Event

Initiating Condition: Automatic or manual trip fails to shut down the reactor

Operating Mode Applicability: 1 - Power Operation, 2 - Startup

Emergency Action Levels: (1 or 2)

Note 7:	A manual action is **any** operator action, or set of actions, taken at the Main Control Room workstations 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.

(1) An automatic trip did **not** reduce reactor power to < 5%
AND

A subsequent manual action taken at the Main Control Room workstations is successful in reducing reactor power < 5% (Note 7)

(2) A manual trip did **not** reduce reactor power to < 5%
AND

A subsequent automatic trip or manual trip action taken at the Main Control Room workstations is successful in reducing reactor power < 5% (Note 7).

Bases:

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a few percent of the original power level and then decays to a level some 8 decades less at a startup rate of about -1/3 DPM. 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 negative startup rate as nuclear power drops into the source range (ref. 1).

A manual action at the Main Control Room workstations 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 includes DAS, one or both PDSP reactor trip switches and opening Rod Drive MG supply and output breakers. 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 Main Control Room (e.g. any location outside the Main Control Room, are not considered to be “at the Main Control Room workstations”.

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 SA5. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA5 or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

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 RTS fails to automatically shut down the reactor, this IC and the EALs are applicable, and must be evaluated.
- If the signal does not cause a plant transient and the failure is determined through other means (e.g., assessment of test results), this IC and the EALs are not applicable and no classification is warranted.

EAL (1)

A reactor trip is automatically initiated by the Reactor Trip System (RTS) when certain continuously monitored parameters exceed predetermined setpoints (ref. 1):

Following an automatic trip signal the Operator verifies that the Reactor Trip Breakers are open and that neutron flux is less than 5%. If either response is not obtained the Operator manually actuates DAS Reactor and Turbine Trip (ref. 1).

In the event that the Operator identifies a reactor trip is imminent and successfully initiates a manual reactor trip before the automatic 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 loss of the fuel clad barrier. If manual reactor trip actions in the Main Control Room fail to reduce reactor power below the power associated with the SAFETY SYSTEM design (< 5%) (ref. 2), the event escalates to an Alert under EAL SA5.1.

This EAL addresses a failure of the RTS 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 Main Control Room workstations 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 Main Control Room workstations (DAS, either PDSP trip switch, or opening Rod Drive MG supply and output breakers) to shut down the reactor (e.g., initiate a manual reactor trip). If any 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 Main Control Room workstations to shut down the reactor (e.g., initiate a manual reactor (trip) using a different method). 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 decay heat removal systems.

<u>EAL (2)</u>

This EAL addresses a failure of a manually initiated trip in the absence of having exceeded an automatic RTS trip setpoint and a subsequent automatic or manual trip is successful in shutting down the reactor (reactor power < 5%) (ref. 2).

Following a manually initiated reactor trip signal the operator verifies that the Reactor Trip Breakers are open and that neutron flux is less than 5%. If either response is not obtained the operator manually actuates DAS Reactor and Turbine Trip (ref. 1).

This EAL addresses a failure of the RTS to initiate or complete a manual reactor trip that results in a reactor shutdown, and either a subsequent operator manual action taken at the Main Control Room workstations 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.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the Main Control Room workstations (DAS,either PDSP trip switch or opening Rod Drive MG supply and output breakers) to shut down the reactor (e.g., initiate a manual reactor(trip) using a different method). 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 decay heat removal systems.

VEGP3/4 Basis Reference(s):

1. E-0 Reactor Trip or Safeguards Actuation
2. F-0 Critical Safety Function Status Trees

SA5

ECL: Alert

Initiating Condition: Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the control room workstations are **not** successful in shutting down the reactor

Operating Mode Applicability: 1 - Power Operation, 2 - Startup

Emergency Action Levels: (1)

Note 7:	A manual action is **any** operator action, or set of actions, taken at the Main Control Room workstations 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.

(1) An automatic or manual trip did **not** reduce reactor power < 5%

AND

Subsequent manual actions taken at the Main Control Room workstations are **not** successful in reducing reactor power < 5% (Note 7)

Bases:

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 energy in excess of the heat load for which the SAFETY SYSTEMS were designed (≥ 5%) (ref. 1, 2).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a few percent of the original power level and then decays to a level some 8 decades less at a startup rate of about -1/3 DPM. 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 negative startup rate as nuclear power drops into the source range (ref. 1).

This IC addresses a failure of the RTS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and subsequent operator manual actions taken at the Main Control Room workstations (DAS, both PDSP trip switches or opening Rod Drive MG supply and output breakers) to shut down 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 shut down by an action taken away from the reactor control consoles since this event entails a significant failure of the RTS.

A manual action at the Main Control Room workstations 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 includes the DAS, PDSP reactor trip switches and opening Rod Drive MG supply and output breakers. 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 Main Control Room workstations (e.g., locally opening breakers). Actions taken at back-panels or other locations within the Control Room (e.g. any location outside the Main Control Room, are not considered to be “at the Main Control Room workstations”).

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 SS5. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS5 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.

VEGP3/4 Basis Reference(s):

1. E-0 Reactor Trip or Safeguards Actuation
2. F-0 Critical Safety Function Status Trees

SS5

ECL: Site Area Emergency

Initiating Condition: Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

Operating Mode Applicability: 1 - Power Operation, 2 - Startup

Emergency Action Levels: (1)

(1) An automatic or manual trip did **not** reduce reactor power < 5%

AND

All manual actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5%

AND

EITHER of the following conditions exist:

- CSFST Core Cooling-RED path conditions met
- CSFST Heat Sink-RED path conditions met

Bases:

This EAL addresses the following:

- Any automatic or manual reactor trip signal 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 (≥ 5%, ref. 2) (EAL SA5.1), and
- Indications that either core cooling is extremely challenged (CSFST Core Cooling-RED path) or heat removal is extremely challenged (CSFST Heat Sink-Red path) (ref. 2.)

At the Site Area Emergency classification level additional capabilities away from the Main Control Room workstations may be considered.

Indication that core cooling is extremely challenged is manifested by entry to Critical Safety Function Status Tree (CSFST) Core Cooling-RED path (ref. 2). Indication that heat removal is extremely challenged is manifested by entry to CSFST Heat Sink-RED path (ref. 2).

This IC/EAL addresses a failure of the RTS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, all subsequent operator actions to manually shut down 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.

VEGP3/4 Basis Reference(s):

1. E-0 Reactor Trip or Safeguards Actuation
2. F-0 Critical Safety Function Status Trees

SU6

ECL: Unusual Event

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

Operating Mode Applicability: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Safe Shutdown

Emergency Action Levels: (1 or 2 or 3)

(1) Loss of **all** Table S-1 onsite communication methods

(2) Loss of **all** Table S-1 ORO communication methods

(3) Loss of **all** Table S-1 NRC communication methods

Table S-1 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Plant Page	X		
VEGP Dial	X		
Dedicated Dial	X		
In-Plant Radio	X		
Commercial Phones	X	X	X
Southern Company Communications	X	X	X
ENN		X	
ENS			X

Basis:

The Table S-1 list for onsite communications loss encompasses the loss of all means of routine communications (e.g., commercial and internal telephones, page system, and radios) (ref. 1, 2, 3).

The Table S-1 list for offsite (ORO) communications loss encompasses the loss of all means of communications with offsite authorities (e.g., dedicated and commercial telephone lines).

The Table S-1 list for NRC communications loss includes the FTS (ENS) and commercial telephone lines (ref. 1, 2, 3).

This IC is the hot condition equivalent of the cold condition IC CU5.

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 and third EAL conditions address a total loss of the communications methods used to notify all offsite organizations of an emergency declaration. The OROs referred to here are the State of Georgia and South Carolina, Burke, Aiken, Lexington and Barnwell County EOCs as well as the NRC.

VEGP3/4 Basis Reference(s):

1. UFSAR 9.5.2 Communication System
2. SNC Standard Emergency Plan – Section F
3. VEGP Units 3 and 4 Annex Table 5.2.A Emergency Response Communications

SU7

ECL: Unusual Event

Initiating Condition: Failure to isolate containment or loss of containment pressure control

Operating Mode Applicability: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Safe Shutdown

Emergency Action Levels: (1 or 2)

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

(1) Failure of containment to isolate when required by an actuation signal

AND

At least one isolation valve in each penetration is **not** closed within 15 min. of the actuation (Note 1)

(2) Containment pressure > 6.2 psig (High 2)

AND

PCS flow **cannot** be established within 15 min. (Note 1)

Basis:

EAL (1)

This IC/EAL addresses a failure of one or more containment penetrations to automatically isolate (close) when required by an actuation signal. Absent challenges to another fission product barrier, this condition represents potential degradation of the level of safety of the plant.

For this EAL the containment isolation signal must be generated as the result of an off-normal/accident condition (e.g., automatic or manual Safeguards or PCS signals); 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 AOPs and EOPs. The 15-minute criterion is included to allow operators time to manually isolate the required penetrations, if possible.

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 and a direct release pathway to the environment as a result of the failed isolation.

EAL (2)

The containment pressure High 2 setpoint (6.2 psig) is the pressure (PCS-PT005/006/007/008) at which PCS should actuate and begin performing its function (ref. 1).

During a DBA, one passive containment cooling water flow path is required to maintain the containment peak pressure and temperature below the design limits (ref. 2).

This IC/EAL 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 this condition represents potential degradation of the level of safety of the plant.

This EAL addresses a condition where containment pressure is greater than the setpoint at which the Passive Containment Cooling System (PCS) is designed to automatically actuate per design. The 15-minute criterion is included to allow operators time to manually align PCS components that may not have automatically actuated, if possible. The inability to actuate the required equipment and perform its safety function indicates that containment heat removal/depressurization systems 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.

VEGP3/4 Basis Reference(s):

1. Technical Specifications Table 3.3.8-1 Engineered Safeguards Actuation System Instrumentation
2. Technical Specifications LCO 3.6.6 Bases

SA8

ECL: Alert

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

Operating Mode Applicability: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Safe Shutdown

Emergency Action Levels: (1)

(1) The occurrence of **any** Table S-2 hazardous event

AND

EITHER of the following:

- 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-2 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

Basis:

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 safe 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.

- The significance of seismic events are discussed under EAL HU2.1 (ref. 1, 5).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2, 3).
- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 145 mph (sustained). (ref. 4, 6).
- UFSAR Section 1.2.3 Plant Arrangement Description to identify major plant structures containing functions and systems required for safe shutdown of the plant (ref. 5)
- An EXPLOSION (including a steam line 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. The need to classify a steam line break not considered an explosion itself is considered in fission product barrier degradation monitoring (EAL Category F).

This IC/EAL 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 division 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 division.

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.

Some systems have a mix of equipment that is SAFETY-RELATED and non-safety-related. Example: The safety-related function of ECS is to trip open RCP circuit breakers on a CMT signal. Damage to these breakers would constitute VISIBLE DAMAGE to a SAFETY SYSTEM. However, damage limited to ECS-ES-1 or a Standby Diesel Generator would not meet the intent of VISIBLE DAMAGE to a SAFETY SYSTEM.

Escalation of the emergency classification level would be via IC FS1 or RS1.

VEGP3/4 Basis Reference(s):

1. UFSAR Section 3.7 Seismic Design
2. UFSAR Section 3.4 Water Level (Flood) Design
3. 3(4)-AOP-908 Internal Flooding Response
4. UFSAR Section 3.3 Wind and Tornado Loadings
5. UFSAR Section 1.2.3 Plant Arrangement Description
6. 3(4)-AOP-901 Acts of Nature

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 Barrier: The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.
B. RCS Barrier: 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 Barrier: The Containment Barrier includes the containment 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 containment 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 ECL:

Alert:

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

Site Area Emergency:

*Loss or potential loss of **any** two barriers*

General Emergency:

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

The logic used for Category F EALs reflects the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier.
- Unusual Event ICs associated with fission product barriers are addressed in Recognition Category S.

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 VEGP Unit 3 and Unit 4 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 containment, a secondary-side system (i.e., steam generator tube leakage), an interfacing system, or outside of the containment building. 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, classification decision-makers 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 Director would have more assurance that there was no immediate need to escalate to a General Emergency.

FA1

ECL: Alert

Initiating Condition: Any loss or **any** potential loss of either Fuel Clad or RCS

Operating Mode Applicability: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Safe Shutdown

Emergency Action Levels: (1)

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

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.

VEGP3/4 Basis Reference(s):

None

FS1

ECL: Site Area Emergency

Initiating Condition: Loss or potential loss of **any** two barriers

Operating Mode Applicability: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Safe Shutdown

Emergency Action Levels: (1)

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

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 Director would have greater assurance that escalation to a General Emergency is less imminent.

VEGP3/4 Basis Reference(s):

None

FG1

ECL: General Emergency

Initiating Condition: Loss of **any** two barriers and loss or potential loss of the third barrier

Operating Mode Applicability: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Safe Shutdown

Emergency Action Levels: (1)

(1) Loss of **any** two barriers

AND

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

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

VEGP3/4 Basis Reference(s):

None

ATTACHMENT 2

FISSION PRODUCT BARRIER

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 the three barriers occupy 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 Barrier Loss column) lists the categories (types) of fission product barrier thresholds. The fission product barrier categories are:

1. RCS or SG Tube Leakage
2. Inadequate Heat Removal
3. RCS Activity / Containment Radiation
4. CTMT Integrity or Bypass
5. Emergency Director Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the category rows and the Loss/Potential Loss columns. The intersection of each category row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If a threshold for a barrier Loss/Potential Loss is not defined, the word "None" is entered in the cell.

Thresholds are assigned letters within each Loss and Potential Loss column beginning with "A." In this manner, a threshold can be identified by its category number and threshold letter. For example, the first Fuel Clad barrier Loss in Category 2 is "FC Loss 2.A," the third Containment barrier Potential Loss in Category 4 is "CTMT P-Loss 4.C," etc.

If a cell in Table F-1 contains more than one threshold, each of the 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 row of fission product barrier Loss and Potential Loss thresholds in that category to determine if any threshold has been exceeded. If a threshold has not been exceeded in that category row, the EAL-user proceeds to the next likely category and continues review of the row of thresholds in the new category

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 Building radiation is sufficiently high a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Containment barrier can exist. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1, FS1, and FA1 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 to category Loss followed by category Potential Loss beginning with Category 1, then 2…5.

Table F-1 Fission Product Barrier Threshold Matrix						
	Fuel Clad (FC) Barrier		**Reactor Coolant System (RCS) Barrier**		**Containment (CTMT) Barrier**	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
1 RCS or SG Tube Leakage	None	A. RCS hot leg level < 0.5%	A. An automatic or manual safeguards actuation required by EITHER: • UNISOLABLE RCS leakage • SG tube RUPTURE B. Automatic or manual actuation of ADS	A. Operation of a second CVS makeup pump is required by EITHER: • UNISOLABLE RCS leakage • SG tube leakage B. CSFST Integrity-RED path conditions met	A. A leaking or RUPTURED SG is FAULTED outside of containment	None
2 Inadequate Heat Removal	A. CSFST Core Cooling-RED path conditions met	A. CSFST Core Cooling-ORANGE path conditions met B. CSFST Heat Sink-RED path conditions met AND Heat sink is required	None	A. CSFST Heat Sink-RED path conditions met AND Heat sink is required	None	A. CSFST Core Cooling-RED path conditions met AND Restoration procedures **not** effective within 15 min. (Note 1)
3 RCS Activity / CTMT Radiation	A. PXS-RY160/161/162/163 CTMT High Range Radiation > [*plant-specific*] R/hr B. Dose equivalent I-131 coolant activity > 300 μCi/gm	None	A. PXS-RY160/161/162/163 CTMT High Range Radiation > [*plant-specific*] R/hr	None	None	A. PXS-RY160/161/162/163 CTMT High Range Radiation > [*plant-specific*] R/hr
4 CTMT Integrity or Bypass	None	None	None	None	A. Containment isolation is required AND EITHER: • Containment integrity has been lost based on ED judgment • UNISOLABLE pathway from containment to the environment exists B. Indications of RCS leakage outside of containment	A. CSFST Containment-RED path conditions met B. Containment hydrogen concentration (VLS-AE001/002/003) > 4% C. Containment pressure > 6.2 psig (High 2) AND PCS flow **cannot** be established within 15 min. (Note 1)
5 ED Judgment	A. **Any** condition in the opinion of the ED that indicates loss of the fuel clad barrier	A. **Any** condition in the opinion of the ED that indicates potential loss of the fuel clad barrier	A. **Any** condition in the opinion of the ED that indicates loss of the RCS barrier	A. **Any** condition in the opinion of the ED that indicates potential loss of the RCS barrier	A. **Any** condition in the opinion of the ED that indicates loss of the containment barrier	A. **Any** condition in the opinion of the ED that indicates potential loss of the containment barrier

Barrier: Fuel Clad

Category: 1. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

None

Basis:

None

VEGP3/4 Basis Reference(s):

None

Barrier: Fuel Clad

Category: 1. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

A. RCS hot leg level < 0.5%

Basis:

0.5% is the minimum observable level on the hot leg level instruments RCS-LICA-160A/B.

This reading indicates a reduction in reactor vessel water level that threatens the onset of heat-induced cladding damage.

Short duration spikes in hot leg levels may occur due to flow and pressure oscillations in the RCS induced by variations in break flow and injection flow. When determining if the threshold is exceeded based on hot leg levels being actually voided, consideration should be given to the duration of the low level condition and the indication of both hot leg level instruments.

VEGP3/4 Basis Reference(s):

1. APP-RCS-M3C-101 RCS Instrumentation and Packaged Mechanical System Interface Requirements

Barrier: Fuel Clad

Category: 2. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

A. CSFST Core Cooling-RED path conditions met

Basis:

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path is given in Attachment 2 of F-0 CSFST (Attachment 3 Figure 1 of this document) and indicates significant core exit superheating and core uncovery (ref. 1).

This reading indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

VEGP3/4 Basis Reference(s):

1. F-0 CSFST Attachment 2 Core Cooling

Barrier: Fuel Clad

Category: 2. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

A. CSFST Core Cooling-ORANGE path conditions met

Basis:

Critical Safety Function Status Tree (CSFST) Core Cooling-ORANGE path is given in Attachment 2 of F-0 CSFST (Attachment 3 Figure 1 of this document) and indicates significant core exit superheating and core uncovery (ref. 1).

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

VEGP3/4 Basis Reference(s):

1. F-0 CSFST Attachment 2 Core Cooling

Barrier: Fuel Clad

Category: 2. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

B. CSFST Heat Sink-RED path conditions met
AND
Heat sink is required

Basis:

In combination with RCS Potential Loss 2.A, meeting this threshold results in a Site Area Emergency.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED path is given in F-0 CSFST Attachment 3 (Attachment 3 Figure 2 of this document) and indicates the ultimate heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

The phrase “and heat sink required” precludes the need for classification for conditions in which ADS Stage 1-3 and/or Stage 4 are actuated and “feed” from PXS CMTs, accumulators, and/or IRWST and “bleed” from ADS have been established to maintain core cooling. For this case, the passive safety systems (PXS and PCS) provide the required core cooling function and fuel clad protection. FR-H.1 Response To Loss Of Heat Sink returns the user to the procedure and step in effect if ADS Stage 1-3 actuation has occurred (ref. 2)

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.

VEGP3/4 Basis Reference(s):

1. F-0 CSFST Attachment 2 Core Cooling
2. E-0 Reactor Trip or Safeguards Actuation

Barrier: Fuel Clad

Category: 3. RCS Activity / CTMT Radiation

Degradation Threat: Loss

Threshold:

A. PXS-RY160/161/162/163 CNMT High Range Radiation > [*plant-specific*] R/hr

Basis:

[Developer notes: *See Attachment 5*]

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300μCi/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 3.A 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 emergency classification level to a Site Area Emergency.

VEGP3/4 Basis Reference(s):

1. APP-RCS-M3C-101 RCS Instrumentation and Packaged Mechanical System Interface Requirements
2. [*plant-specific calculation supporting FC Loss threshold*]

Barrier: Fuel Clad

Category: 3. RCS Activity / CTMT Radiation

Degradation Threat: Loss

Threshold:

B. Dose equivalent I-131 coolant activity > 300 μCi/gm

Basis:

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.

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

VEGP3/4 Basis Reference(s):

None

Barrier: Fuel Clad

Category: 3. RCS Activity / CTMT Radiation

Degradation Threat: Potential Loss

Threshold:

None

Basis:

None

VEGP3/4 Basis Reference(s):

None

Barrier: Fuel Clad

Category: 4. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Basis:

None

VEGP3/4 Basis Reference(s):

None

Barrier: Fuel Clad

Category: 4. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Basis:

None

VEGP3/4 Basis Reference(s):

None

Barrier: Fuel Clad

Category: 5. ED Judgment

Degradation Threat: Loss

Threshold:

A. **Any** condition in the opinion of the ED that indicates loss of the fuel clad barrier

Basis:

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad Barrier is lost.

VEGP3/4 Basis Reference(s):

None

Barrier: Fuel Clad

Category: 5. ED Judgment

Degradation Threat: Potential Loss

Threshold:

A. **Any** condition in the opinion of the ED that indicates potential loss of the fuel clad barrier

Basis:

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Fuel Clad 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.

VEGP3/4 Basis Reference(s):

None

Barrier: Reactor Coolant System

Category: 1. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

A. An automatic or manual safeguards actuation required by EITHER:
- UNISOLABLE RCS leakage
- SG tube RUPTURE

Basis:

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

RUPTURED - The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require Safeguards actuation.

This threshold is based on an UNISOLABLE RCS leak of sufficient size to require an automatic or manual safeguards actuation. 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 safeguards actuation 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.

VEGP3/4 Basis Reference(s):

None

Barrier: Reactor Coolant System

Category: 1. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

B.	Automatic or manual actuation of ADS

Basis:

The Automatic Depressurization System (ADS) valves are part of the RCS and interface with the Passive Core Cooling System (PXS). Manual or automatic actuation of the ADS valves in response to a valid initiation setpoint initiates an RCS leak meeting the RCS Loss criteria. Inadvertent opening that can be isolated or momentary jogging of ADS valves for pressure control does not constitute a loss of RCS.

VEGP3/4 Basis Reference(s):

None

Barrier: Reactor Coolant System

Category: 1. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

A. Operation of a second CVS makeup pump is required by EITHER:
- UNISOLABLE RCS leakage
- SG tube leakage

Basis:

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

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 single CVS makeup pump, but safeguards actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs that a second CVS makeup pump be placed in service to restore and maintain pressurizer level. Nominal design flow rate of a single CVS makeup pump is 135 gpm.

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.

VEGP3/4 Basis Reference(s):

None

Barrier: Reactor Coolant System

Category: 1. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

B. CSFST Integrity-RED path conditions met

Basis:

The "Potential Loss" threshold is defined by the CSFST RCS Integrity - RED. The values in this EAL are consistent with the CSFST value (ref. 1). CSFST RCS Integrity - RED Path (Attachment 3 Figure 3 of this document) and associated Operational Curve Limit A (Attachment 3 Figure 4 of this document) indicates an extreme challenge to the safety function when plant parameters are to the right of the limit curve following excessive RCS cooldown under pressure (ref. 1).

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 4 or higher (i.e., hot and pressurized).

VEGP3/4 Basis Reference(s):

1. F-0 CSFST Attachment 4 Integrity

Barrier: Reactor Coolant System

Category: 2. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

Basis:

None

VEGP3/4 Basis Reference(s):

None

Barrier: Reactor Coolant System

Category: 2. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

A. CSFST Heat Sink-RED path conditions met
AND
Heat sink is required

Basis:

In combination with Fuel Clad Potential Loss 2.B, meeting this threshold results in a Site Area Emergency.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED path is given in F-0 CSFST Attachment 3 (Attachment 3 Figure 2 of this document) and indicates the ultimate heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

The phrase “and heat sink required” precludes the need for classification for conditions in which ADS Stage 1-3 and/or Stage 4 are actuated and “feed” from PXS CMTs, accumulators, and/or IRWST and “bleed” from ADS have been established to maintain core cooling. For this case, the passive safety systems (PXS and PCS) provide the required core cooling function and fuel clad protection. FR-H.1 Response To Loss Of Heat Sink returns the user to the procedure and step in effect if ADS Stage 1-3 actuation has occurred (ref. 2).

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators or Passive Residual Heat Removal System (PRHR) (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. This threshold is also not applicable to conditions where heat sink is not required (i.e. steam generator pressures are greater than RCS pressure).

Meeting this threshold results in a Site Area Emergency because this threshold is identical to Fuel Clad Barrier Potential Loss threshold 2.B; 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.

VEGP3/4 Basis Reference(s):

1. F-0 CSFST Attachment 3 Heat Sink
2. E-0 Reactor Trip or Safeguards Actuation

Barrier: Reactor Coolant System

Category: 3. RCS Activity / CTMT Radiation

Degradation Threat: Loss

Threshold:

A. PXS-RY160/161/162/163 CNMT High Range Radiation > [*plant-specific*] R/hr

Basis:

[Developer notes: *See Attachment 5*]

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 3.A since it indicates a loss of the RCS Barrier only.

There is no Potential Loss threshold associated with RCS Activity / Containment Radiation.

VEGP3/4 Basis Reference(s):

1. APP-RCS-M3C-101 RCS Instrumentation and Packaged Mechanical System Interface Requirements
2. [*plant-specific calculation supporting RCS Loss threshold*]

Barrier: Reactor Coolant System

Category: 3. RCS Activity / CTMT Radiation

Degradation Threat: Potential Loss

Threshold:

None

Basis:

None

VEGP3/4 Basis Reference(s):

None

Barrier: Reactor Coolant System

Category: 4. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Basis:

None

VEGP3/4 Basis Reference(s):

None

Barrier: Reactor Coolant System

Category: 4. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Basis:

None

VEGP3/4 Basis Reference(s):

None

Barrier: Reactor Coolant System

Category: 5. ED Judgment

Degradation Threat: Loss

Threshold:

A. **Any** condition in the opinion of the ED that indicates loss of the RCS barrier

Basis:

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is lost.

VEGP3/4 Basis Reference(s):

None

Barrier: Reactor Coolant System

Category: 5. ED Judgment

Degradation Threat: Potential Loss

Threshold:

A. **Any** condition in the opinion of the ED that indicates potential loss of the RCS barrier

Basis:

This threshold addresses any other factors that may be used by the Emergency Director 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.

VEGP3/4 Basis Reference(s):

None

Barrier: Containment

Category: 1. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

A. A leaking or RUPTURED SG is FAULTED outside of containment

Basis:

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.

RUPTURED - The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require Safeguards actuation.

This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for RCS Barrier Potential Loss 1.A and Loss 1.A, respectively. This condition represents a bypass of the containment barrier.

FAULTED is a defined term; 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 Main Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU3 for the fuel clad barrier (i.e., RCS activity values) and IC SU4 for the RCS barrier (i.e., RCS leak rate values).

Steam releases associated with the expected operation of a SG Power Operated Relief Valve (PORV) or Main Steam Safety Valve (MSSV) 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 MSSV) do meet this threshold.

Following a SG tube leak or RUPTURE, there may be minor radiological releases through a secondary-side system component (e.g., condenser vacuum pumps, gland seal exhausters, valve packing, 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 emergency classification levels 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 25 gpm	No classification	No classification
Greater than 25 gpm	Unusual Event per SU4	Unusual Event per SU4
Requires an automatic or manual safeguards actuation (*RCS Barrier Loss*) **OR** operation of a second CVS Makeup pump due to SG tube leakage (*RCS Barrier Potential Loss*)	Site Area Emergency per FS1	Alert per FA1

There is no Containment Potential Loss threshold associated with RCS or SG Tube Leakage.

VEGP3/4 Basis Reference(s):

None

Barrier: Containment

Category: 1. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

Basis:

None

VEGP3/4 Basis Reference(s):

None

Barrier: Containment

Category: 2. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

Basis:

None

VEGP3/4 Basis Reference(s):

None

Barrier: Containment

Category: 2. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

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

A. CSFST Core Cooling-RED path conditions met AND Restoration procedures **not** effective within 15 min. (Note 1)

Basis:

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path is given in Attachment 2 of F-0 CSFST (Attachment 3 Figure 1 of this document) and indicates significant core exit superheating and core uncovery (ref. 1).

The function restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing (ref. 2).

A direct correlation to status trees can be made if the effectiveness of the restoration procedures is also evaluated.

This threshold indicates significant core exit superheating and core uncovery. Since core exit thermocouple (TC) readings are greater than 1,200°F (ref. 1), the Fuel Clad barrier is also lost.

This condition represents an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. For this condition to occur, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.

The restoration procedure is considered "effective" if core exit thermocouple readings are decreasing and/or if reactor vessel level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The Emergency Director should escalate the ECL to a General Emergency as soon as it is determined that the procedure(s) will not be effective.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation in a significant fraction of core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.

VEGP3/4 Basis Reference(s):

1. F-0 CSFST Attachment 3 Heat Sink
2. FR.C-1 Response to Inadequate Core Cooling

Barrier: Containment

Category: 3. RCS Activity / CTMT Radiation

Degradation Threat: Loss

Threshold:

None

Basis:

None

VEGP3/4 Basis Reference(s):

None

Barrier: Containment

Category: 3. RCS Activity / CTMT Radiation

Degradation Threat: Potential Loss

Threshold:

A. PXS-RY160/161/162/163 CNMT High Range Radiation > [*plant-specific*] R/hr

Basis:

[Developer notes: *See Attachment 5*]

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.

VEGP3/4 Basis Reference(s):

1. APP-RCS-M3C-101 RCS Instrumentation and Packaged Mechanical System Interface Requirements
2. [*plant-specific calculation supporting CTMT Potential Loss threshold*]

Barrier: Containment

Category: 4. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

A. Containment isolation is required

AND EITHER:

- Containment integrity has been lost based on ED judgment
- UNISOLABLE pathway from containment to the environment exists

Basis:

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

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 Director 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 Attachment 3 Figure 6. 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.

<u>Second Threshold</u> – 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 not attributable to PCS actuation.

Refer to the top piping run of Attachment 3 Figure 6. 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 Attachment 3 Figure 6. In this simplified example, leakage in an RCP 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 the 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 (design) 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 a closed 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 1.A.

VEGP3/4 Basis Reference(s):

None

Barrier: Containment

Category: 4. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

B. Indications of RCS leakage outside of containment

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 Attachment 3 Figure 6. 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 4.A 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 1.A to be met.

VEGP3/4 Basis Reference(s):

None

Barrier: Containment

Category: 4. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

A. CSFST Containment-RED path conditions met

Basis:

Critical Safety Function Status Tree (CSFST) Containment-RED path (Attachment 3 Figure 5) is entered if Containment pressure is greater than or equal to 59 psig and represents an extreme challenge to safety function (ref. 1).

59 psig is based on the containment design pressure (ref. 2).

If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. To reach this level, there must have been inadequate core cooling for an extended period of time. The RCS and Fuel Clad barriers have already been lost. This would cause an upgrade of ECL from Site Area Emergency to General Emergency based on the potential loss of the third barrier.

VEGP3/4 Basis Reference(s):

1. F-0 CSFST Attachment 5 Containment
2. UFSAR Section 3.8.2.1 Description of the Containment

Barrier: Containment

Category: 4. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

B. Containment hydrogen concentration (VLS-AE001/002/003) > 4%

Basis:

The lower limit of flammability of hydrogen in air is approximately 4%.

In the course of a severe accident, a substantial amount of combustible gases can be generated in-vessel from the oxidation of the zirconium and other metals. The AP1000 containment is provided with nonsafety-related hydrogen igniters to control the concentration of combustible gases. If the igniters operate, combustion of hydrogen plumes may present a thermal load to the containment. Combustible gas can accumulate in the containment at flammable concentrations if the igniter system fails to function (ref. 1).

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.

VEGP3/4 Basis Reference(s):

1. UFSAR Section 19.41 Hydrogen Mixing and Combustion Analysis

Barrier: Containment

Category: 4. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

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

C. Containment pressure > 6.2 psig

AND

PCS flow **cannot** be established within 15 min. (Note 1)

Basis:

The containment pressure high-high setpoint (6.2 psig, ref. 1) is the pressure at which the Passive Containment Cooling (PCS) should actuate and begin performing its function.

During a DBA, one passive containment cooling water flow path is required to maintain the containment peak pressure and temperature below the design limits (ref. 2).

This threshold describes a condition where containment pressure is greater than the setpoint at which the containment energy (heat) removal system, Passive Containment Cooling System (PCS) is designed to automatically actuate, but does not actuate and perform its safety function per design. The 15-minute criterion is included to allow operators time to manually align PCS components that may not have automatically actuated, if possible. This threshold represents a potential loss of containment in that the containment heat removal/depressurization systems (not including containment venting strategies) are either lost or performing in a degraded manner.

VEGP3/4 Basis Reference(s):

1. Technical Specifications LCO 3.3.8
2. Technical Specifications LCO 3.6.6 Bases

Barrier: Containment

Category: 5. ED Judgment

Degradation Threat: Loss

Threshold:

A. **Any** condition in the opinion of the ED that indicates loss of the containment barrier

Basis:

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is lost.

VEGP3/4 Basis Reference(s):

None

Barrier: Containment

Category: 5. ED Judgment

Degradation Threat: Potential Loss

Threshold:

A. **Any** condition in the opinion of the ED that indicates potential loss of the containment barrier

Basis:

This threshold addresses any other factors that may be used by the Emergency Director 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.

VEGP3/4 Basis Reference(s):

None

ATTACHMENT 3

Safe Operation & Shutdown Areas/Rooms

Tables R-1 & H-2 Bases

VEGP3/4 Table R-1 and H-2 Bases

The R-1 and H-2 tables 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 are not included (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations).

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) are not included.

The Main Control Room is not included due to VES/VBS providing adequate control room isolation/breathing air.

Additional actions with rooms were considered (e.g. placing a second train of SWS/CCS in-service, chemical additions, isolating unborated water sources, etc.) but were not necessary to perform a plant shutdown and cooldown to MODE 5.

A review of station operating procedures identified the following mode dependent in-plant actions and associated areas that are required for normal plant operation, cooldown or shutdown:

Table R-1/H-2 Bases

Building	Room Number	Room Name	Applicable Mode	Basis
Auxiliary Building	12152	Primary Sample Room	3	Take Chemistry samples of RCS to ensure RCS is borated prior to lowering Ppzr < P-11.
	12252	Rad Chem Lab	3	Analyze Chemistry samples of RCS to ensure RCS is borated prior to lowering Ppzr < P-11.
	12321	Non-1E Equipment / Penetration Room	4	Restore power to accumulator MOVs (PXS-V027A/B) so accumulators can be isolated. Open RNS-V024 breaker.
	12305	Division D I&C Penetration Room	4	Restore power to RNS loop suction MOVs so RNS can be placed in service for shutdown cooling
	12304	Division B I&C Penetration Room	4	Restore power to RNS loop suction MOVs so RNS can be placed in service for shutdown cooling. Open RNS-V023 breaker
	12561[1]	CCS Valve Room[1]	4	Valve in cooling water to RNS HX B (open CCS-V301B)

Table R-1/H-2 Bases				
Building	**Room Number**	**Room Name**	**Applicable Mode**	**Basis**
Annex Building	4000[1]	Annex Building[1]	4	Valve in cooling water to RNS HX A (open CCS-V301A).
Containment	11209	CVS Room	4	Bypass letdown orifice to keep letdown available. Slowly pressurize RNS Hot Leg suction.

[1] Classification is **not** required if either train of RNS can be placed in service for Shutdown Cooling.

Example 1: If both trains of RNS are available, but then access is prevented to Room 12561, then no classification is required provided the annex building can be accessed to valve-in CCS to the “A” RNS HX.

Example 2: If RNS pump A is unavailable, and an event prevents access to Room 12561, then classification IS warranted because neither train of RNS could be placed in service.

Procedures reviewed

GOP-101 “Power Operations Above 25% Power”

GOP-202 “Plant Shutdown 25% Power To Mode 3” (Westinghouse number GOP-102)

GOP-205 “Plant Cooldown Mode 3 to Mode 5” (Westinghouse number GOP-103)

Table R-1 & H-2 Results

Table R-1/H-2 Safe Operation & Shutdown Areas/Rooms			
Building	**Room Number**	**Room/Area Name**	**Applicable Mode**
Auxiliary	12152	Primary Sample Room	3
	12252	Rad Chem Lab	3
	12321	Non-1E Equipment / Penetration Room	4

Table R-1/H-2 Safe Operation & Shutdown Areas/Rooms			
Building	**Room Number**	**Room/Area Name**	**Applicable Mode**
	12305	Division D I&C Penetration Room	4
	12304	Division B I&C Penetration Room	4
	12561	CCS Valve Room (Note 9)	4
Annex	4000	Annex Building (Note 9)	4
Containment	11209	CVS Room	4

Note 9: Classification is **not** required if either train of RNS can be placed in service for Shutdown Cooling

ATTACHMENT 4

Figures

Figure 1: CSFST Core Cooling

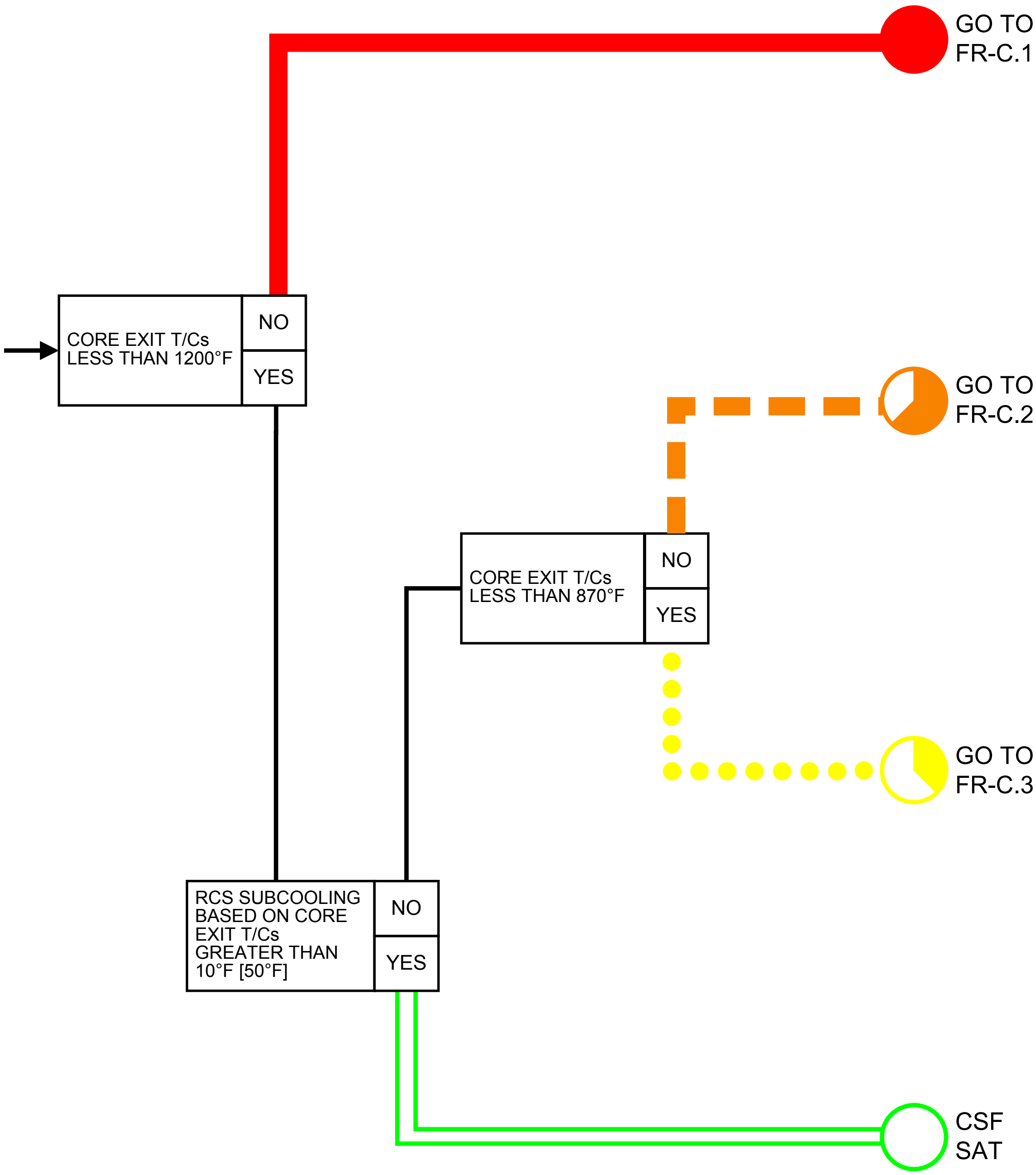


Figure 2: CSFST Heat Sink

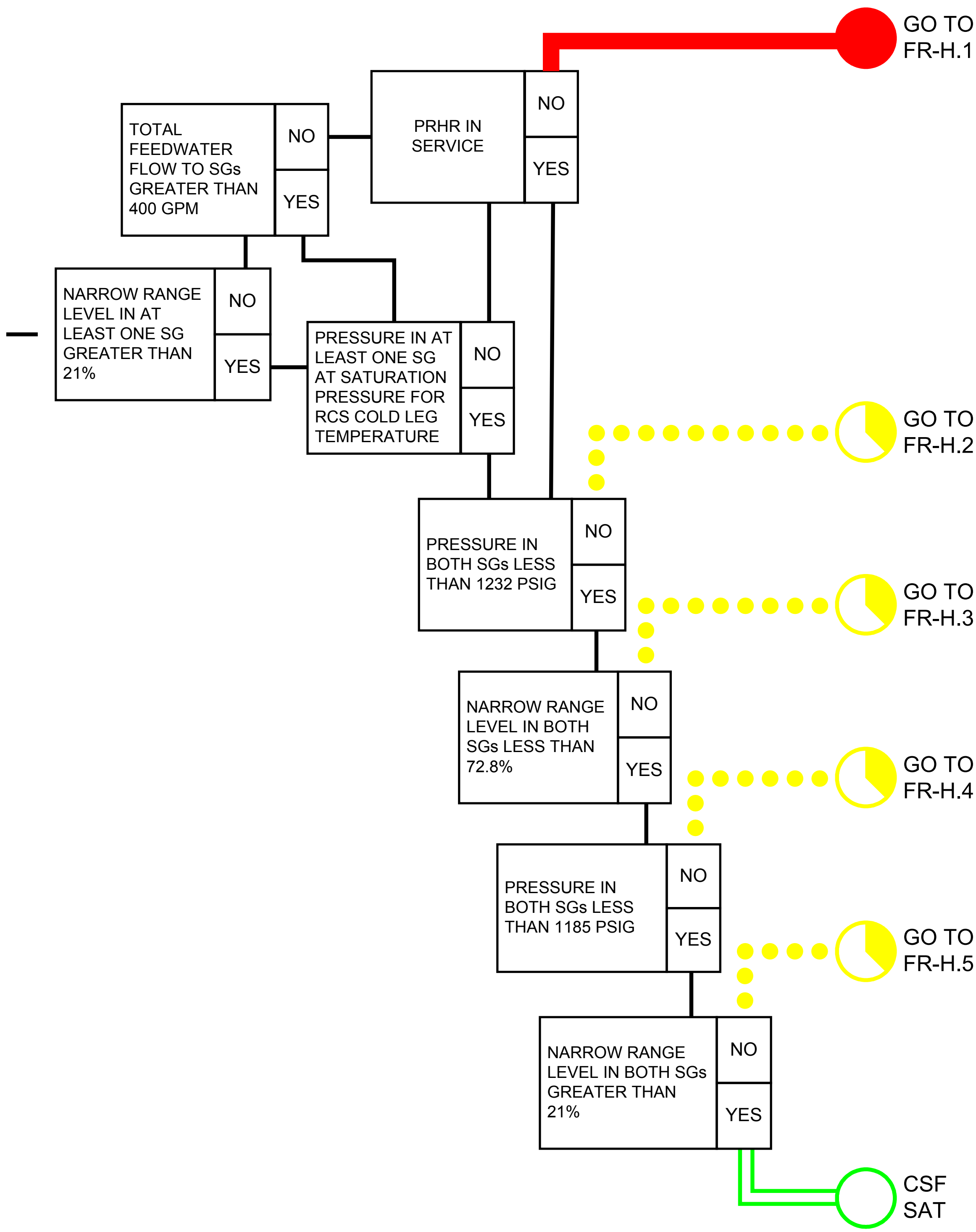


Figure 3: CSFST RCS Integrity

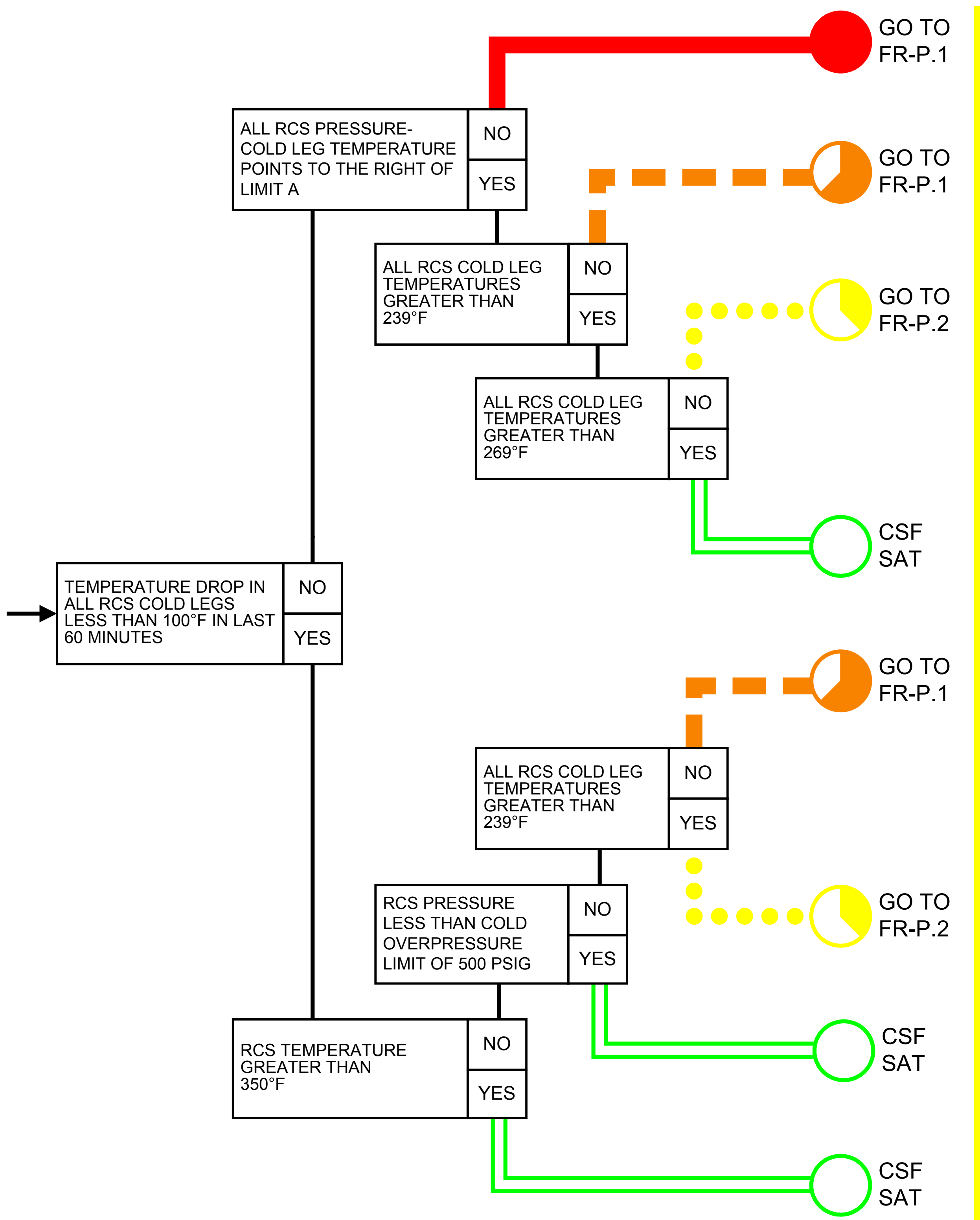


Figure 4: Plant Operational Limits Curve

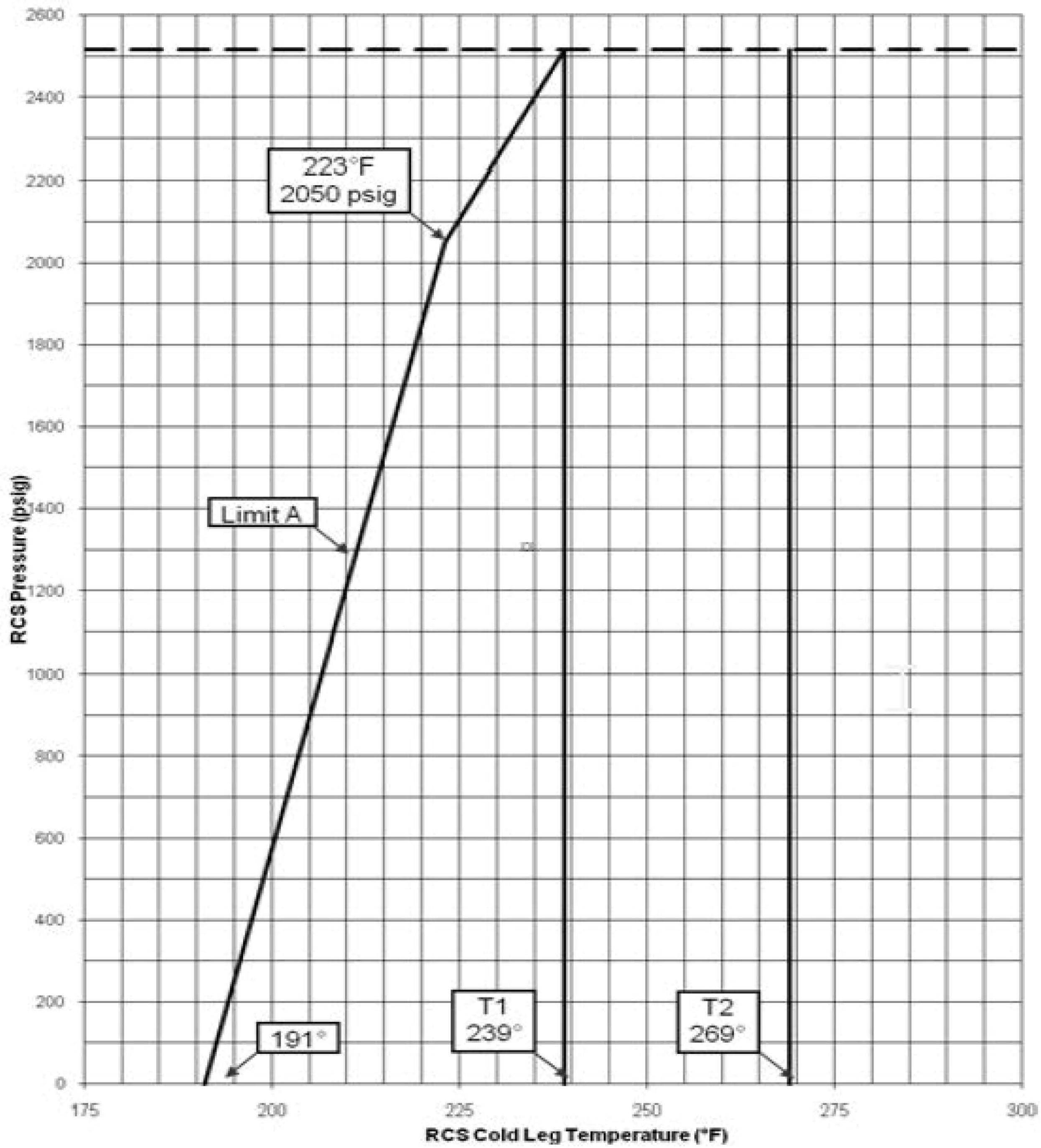


Figure 5: CSFST Containment

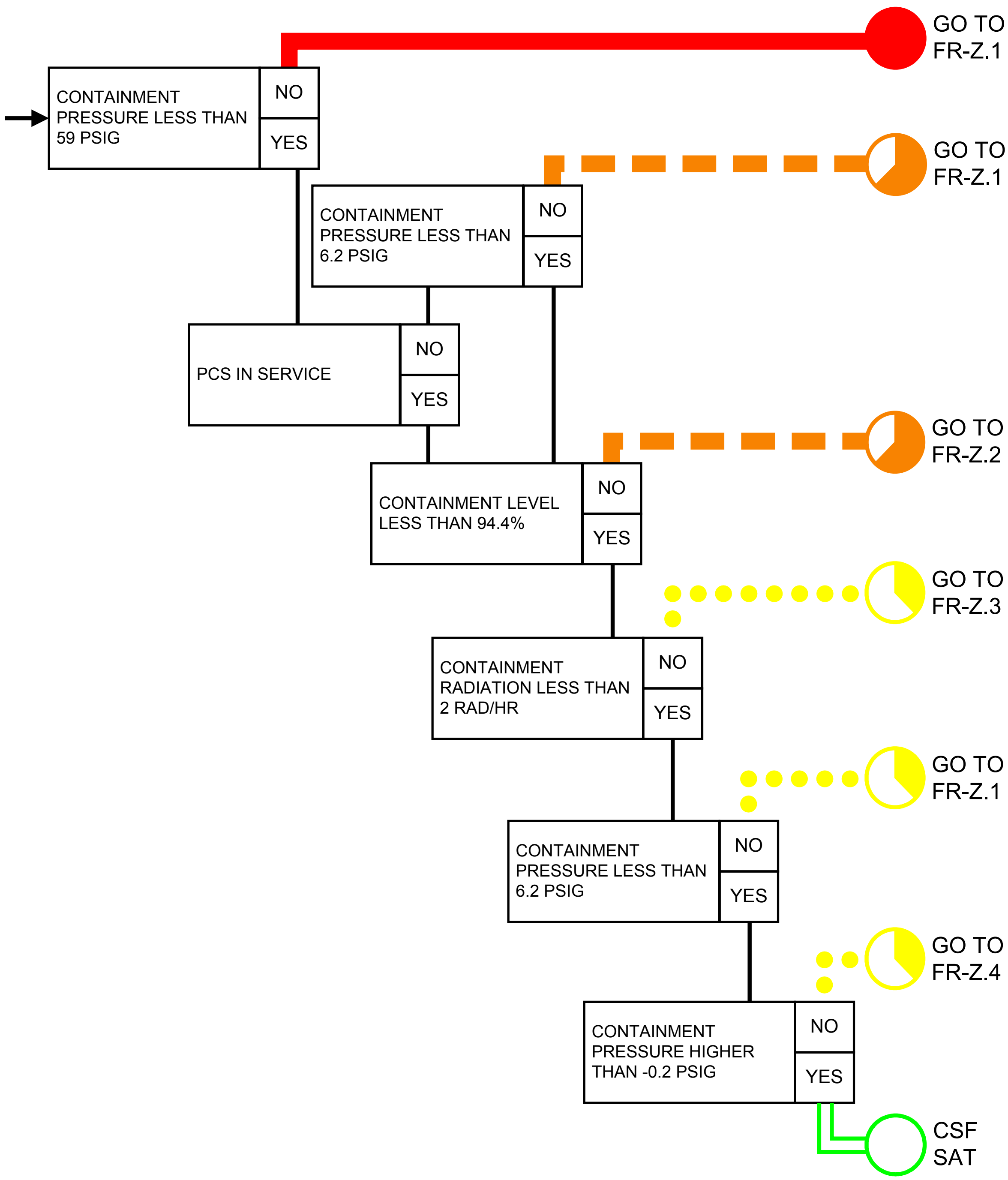
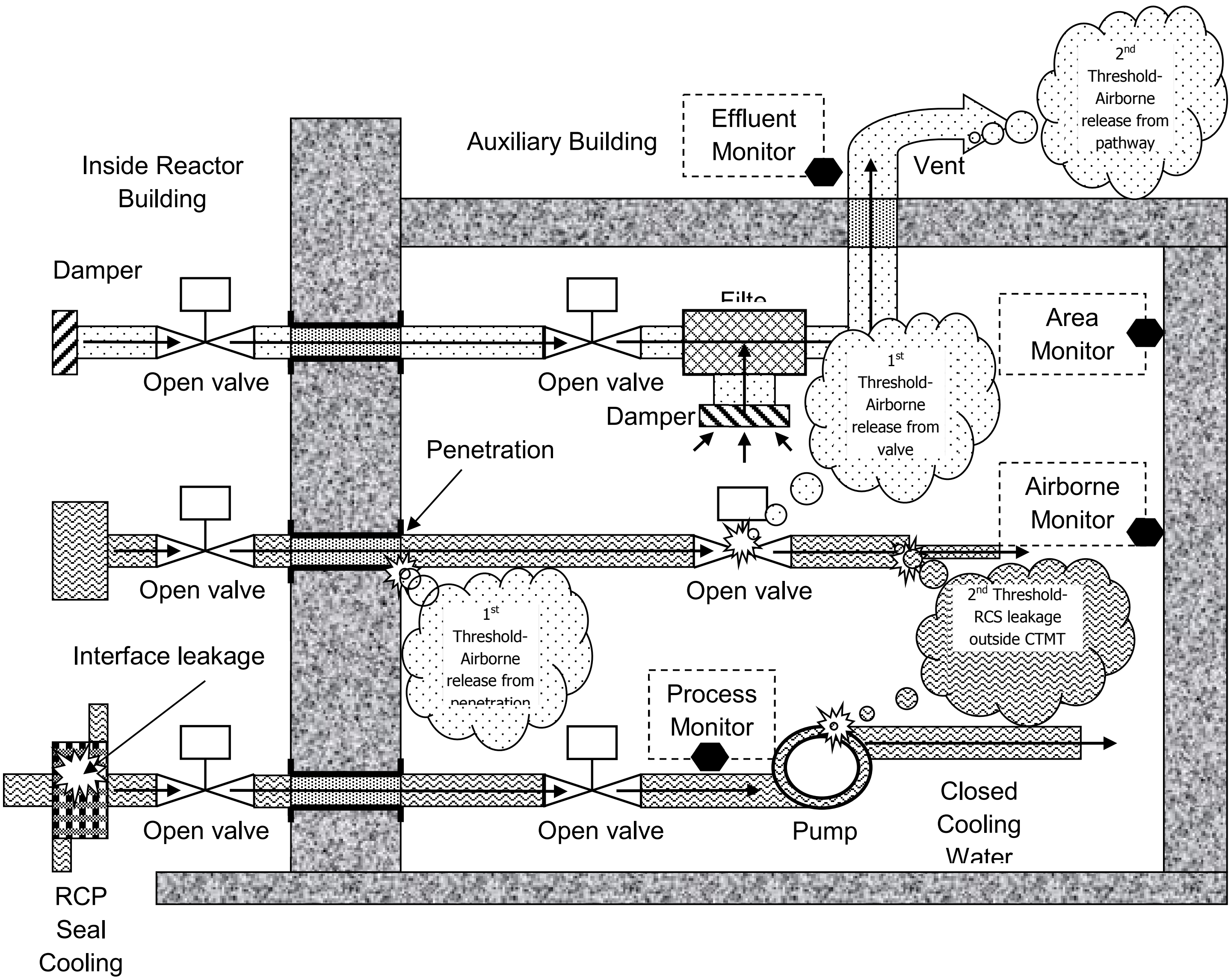


Figure 6: Containment Integrity or Bypass Examples



ATTACHMENT 5

Developer Notes

Section/IC/EAL #	Developer Notes
4.1.5 RU1	Site-specific – VEGP3/4 Offsite Dose Calculation Manual (ODCM) title shall be incorporated as a developmental and VEGP3/4 bases reference when published.
RU1	The VEGP3/4 Offsite Dose Calculation Manual is being developed including the gaseous or liquid radioactive release limits. Effluent threshold values related to RU1 (2 x ODCM release rate limits) shall be calculated and incorporated once the final ODCM is published and effluent monitor response characteristics are established.
RA1	Effluent threshold values related to RA1 (1% of the EPA Protective Action Guidelines) shall be calculated and incorporated once the final emergency dose assessment methodology/model is approved. The gaseous release values represent offsite dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 10 mRem TEDE or 50 mRem thyroid CDE (1% of the EPA PAGs). The emergency dose assessment methodology/model will be used to calculate monitor readings at or beyond the SITE BOUNDARY that would yield the limiting EPA PAG dose assuming: • Design basis RCS source term • Annual average meteorology (wind speed and stability) • Default release duration • Most limiting wind direction (highest $^{x}/_{Q}$)
RS1	Effluent threshold values related to RS1 (10% of the EPA Protective Action Guidelines) shall be calculated and incorporated once the final emergency dose assessment methodology/model is approved. The gaseous release values represent offsite dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 100 mRem TEDE or 500 mRem thyroid CDE (10% of the EPA PAGs). The emergency dose assessment methodology/model will be used to calculate monitor readings at or beyond the SITE BOUNDARY that would yield the limiting EPA PAG dose assuming: • Design basis RCS source term • Annual average meteorology (wind speed and stability)

Section/IC/EAL #	Developer Notes
	• Default release duration • Most limiting wind direction (highest $^{\chi}/_{Q}$)
RG1	Effluent threshold values related to RG1 (100% of the EPA Protective Action Guidelines) shall be calculated and incorporated once the final emergency dose assessment methodology/model is approved. The column "GE" gaseous release values represent offsite dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 1000 mRem TEDE or 5000 mRem thyroid CDE (100% of the EPA PAGs). The emergency dose assessment methodology/model will be used to calculate monitor readings at or beyond the SITE BOUNDARY that would yield the limiting EPA PAG dose assuming: • Design basis RCS source term • Annual average meteorology (wind speed and stability) • Default release duration • Most limiting wind direction (highest $^{\chi}/_{Q}$)
Fuel Clad Loss 3.A	The radiation monitor reading will correspond to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300μCi/gm dose equivalent I-131. When the site-specific response characteristics are established for APP-PXS-JS-160/161/162/163, the FC Loss threshold value based on 2-5% clad failure coolant activity discharged into containment will be calculated and incorporated.
RCS Loss 3.A	The radiation monitor reading will correspond to an instantaneous release of all reactor coolant mass into the containment, assuming reactor coolant activity at the Technical Specification limit. When the site-specific response characteristics are established for APP-PXS-JS-160/161/162/163, the RCS Loss threshold value based on Technical Specification coolant activity discharged into containment will be calculated and incorporated.
Containment Loss 3.A	The radiation monitor reading will correspond to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity corresponds to 20% clad failure. When the site-specific response characteristics are established for APP-

Section/IC/EAL #	Developer Notes
	PXS-JS-160/161/162/163, the Containment Potential Loss threshold value based on 20% clad failure coolant activity discharged into containment will be calculated and incorporated.

Southern Nuclear Operating Company

ND-16-0150

Enclosure 3

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

EAL Classification Matrices

for the Proposed Emergency Action Levels – For Information Only

(LAR-16-002)

(Enclosure 3 consists of 3 pages, including this cover page)

Hot Condition EAL Matrices

For Information Only

UNITS 3 & 4		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
R Abnormal Rad Levels / Rad Effluent	1 Rad Effluent	RG1 Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE 1 2 3 4 5 6 DEF (1) Reading on **any** of the following effluent radiation monitors for ≥ 15 min. (Notes 1, 2, 3, 4) Plant Vent VFS-RY103 [plant-specific] Plant Vent VFS-RY104A [plant-specific] Plant Vent VFS-RY104B [plant-specific] Turbine Island Vent TDS-RY001A [plant-specific] Turbine Island Vent TDS-RY001B [plant-specific] (2) Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4) (3) Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY: • Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min. • Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation. (Notes 1, 2)	RS1 Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE 1 2 3 4 5 6 DEF (1) Reading on **any** of the following effluent radiation monitors for ≥ 15 min. (Notes 1, 2, 3, 4) Plant Vent VFS-RY103 [plant-specific] Plant Vent VFS-RY104A [plant-specific] Plant Vent VFS-RY104B [plant-specific] Turbine Island Vent TDS-RY001A [plant-specific] Turbine Island Vent TDS-RY001B [plant-specific] (2) Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4) (3) 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)	RA1 Release of gaseous radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE 1 2 3 4 5 6 DEF (1) Reading on **any** of the following effluent radiation monitors for ≥ 15 min. (Notes 1, 2, 3, 4) Plant Vent VFS-RY103 [plant-specific] Plant Vent VFS-RY104A [plant-specific] Plant Vent VFS-RY104B [plant-specific] Turbine Island Vent TDS-RY001A [plant-specific] Turbine Island Vent TDS-RY001B [plant-specific] (2) Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4) (3) 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)	RU1 Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer 1 2 3 4 5 6 DEF (1) Reading on **any** of the following effluent radiation monitors for ≥ 60 min. (Notes 1, 2, 3) Plant Vent VFS-RY103 2 X Hi-Rad alarm Turbine Island Vent TDS-RY001A 2 X Hi-Rad alarm Liquid Radwaste WLS-RY229 2 X Hi-Rad alarm Waste Water WWS-RY021 2 X Hi-Rad alarm (2) Sample analyses for a gaseous or liquid release indicates concentrations or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2)
	2 Irradiated Fuel Event	RG2 Spent fuel pool level (containing irradiated fuel) **cannot** be restored to at least the top of the fuel racks for 60 minutes or longer 1 2 3 4 5 6 DEF (1) Spent Fuel Pool level **cannot** be restored to at least Level 3 (0 ft. on SFS-LT019A/B/C) for ≥ 60 min. (Note 1)	RS2 Spent fuel pool (containing irradiated fuel) level at the top of the fuel racks 1 2 3 4 5 6 DEF (1) Lowering of Spent Fuel Pool level to Level 3 (0 ft. on SFS-LT019A/B/C)	RA2 Significant lowering of water level above, or damage to, irradiated fuel 1 2 3 4 5 6 DEF (1) Uncovery of irradiated fuel in the REFUELING PATHWAY (2) Damage to irradiated fuel resulting in a release of radioactivity as indicated by a Hi-Rad alarm on **any** of the following radiation monitors: • VAS-RY001 Fuel Handling Area Exhaust • VFS-RY001 Containment Air Filtration Exhaust • RMS-RY012 Fuel Handling Area Monitor • RMS-RY020 Fuel Handling Area Monitor • Refueling Bridge Portable Monitor (Refueling Mode) (3) Lowering of Spent Fuel Pool level to Level 2 (10 ft. on SFS-LT019A/B/C)	RU2 UNPLANNED loss of water level above irradiated fuel 1 2 3 4 5 6 DEF (1) UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **any** of the following: • Spent Fuel Pool Low Alarm (SFS-LT019A/B/C) • Level indication trend • Visual observation AND UNPLANNED rise in area radiation levels as indicated by **any** of the following radiation monitors: • RMS-RY012 Fuel Handling Area Monitor • RMS-RY020 Fuel Handling Area Monitor • Refueling Bridge Portable Monitor (Refueling Mode)
	3 Area Radiation Levels	None		RA3 Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown 1 2 3 4 5 6 DEF (1) Dose rate > 15 mR/hr in EITHER of the following areas: • Main Control Room (RMS-RY010) • Central Alarm Station (by survey) (2) An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 area/room (Note 5)	None
H Hazards	1 Security	HG1 HOSTILE ACTION resulting in loss of physical control of the facility 1 2 3 4 5 6 DEF (1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Team Leader AND EITHER of the following has occurred: **Any** of the following safety functions **cannot** be controlled or maintained • Reactivity control • Core cooling • RCS heat removal OR Damage to spent fuel has occurred or is IMMINENT	HS1 HOSTILE ACTION within the PROTECTED AREA 1 2 3 4 5 6 DEF (1) A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Team Leader	HA1 HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes 1 2 3 4 5 6 DEF (1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Team Leader (2) A validated notification from NRC of an aircraft attack threat within 30 min. of the site	HU1 Confirmed SECURITY CONDITION or threat 1 2 3 4 5 6 DEF (1) A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by Security Team Leader (2) Notification of a credible security threat directed at the site (3) A validated notification from the NRC providing information of an aircraft threat
	2 Seismic Event	None	None	None	HU2 Seismic event greater than OBE levels 1 2 3 4 5 6 DEF (1) Seismic event greater than Operating Basis Earthquake (OBE) as indicated by VALID SJS-JS-01-ALM2 (OBE Exceedance) alarm
	3 Natural or Tech. Hazard	None	None	None	HU3 Hazardous event 1 2 3 4 5 6 DEF (1) A tornado strike within the PROTECTED AREA (2) 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 (3) 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) (4) A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 8)
	4 Fire		None	None	HU4 FIRE potentially degrading the level of safety of the plant 1 2 3 4 5 6 DEF (1) 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 (2) 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) (3) A FIRE within the plant PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm, or indication (Note 1) (4) A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish
	5 Hazardous Gases		None	HA5 Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown 1 2 3 4 5 6 DEF (1) Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 area/room AND Entry into the area is prohibited or IMPEDED (Note 5)	None
	6 Main Control Room Evacuation	None	HS6 Inability to control a key safety function from outside the Control Room 1 2 3 4 5 6 DEF (1) An event has resulted in plant control being transferred from the Main Control Room to the Remote Shutdown Workstation AND Control of **any** of the following key safety functions is **not** reestablished within 15 min. (Note 1) • Reactivity control • Core cooling • RCS heat removal	HA6 Control Room evacuation resulting in transfer of plant control to alternate locations 1 2 3 4 5 6 DEF (1) An event has resulted in plant control being transferred from the Main Control Room to the Remote Shutdown Workstation	None
	7 Judgment	HG7 Other conditions existing that in the judgment of the Emergency Director warrant declaration of General Emergency 1 2 3 4 5 6 DEF (1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	HS7 Other conditions existing that in the judgment of the Emergency Director warrant declaration of Site Area Emergency 1 2 3 4 5 6 DEF (1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. **Any** releases are **not** expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the SITE BOUNDARY.	HA7 Other conditions existing that in the judgment of the Emergency Director warrant declaration of an Alert 1 2 3 4 5 6 DEF (1) Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. **Any** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	HU7 Other conditions existing that in the judgment of the Emergency Director warrant declaration of a UE 1 2 3 4 5 6 DEF (1) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Table R-1 Safe Operation & Shutdown Areas/Rooms

Building	Room Number	Room/Area Name	Applicable Mode
Auxiliary	12152	Primary Sample Room	3
	12252	Rad Chem Lab	3
	12321	Non-1E Equipment/ Penetration Room	4
	12305	Division D I&C Penetration Room	4
	12304	Division B I&C Penetration Room	4
	12561	CCS Valve Room (Note 9)	4
Annex	4000	Annex Building (Note 9)	4
Containment	11209	CVS Room	4

Table H-1 Fire Areas

- Containment Building
- Shield Building
- Auxiliary Building

Table H-2 Safe Operation & Shutdown Areas/Rooms

Building	Room Number	Room/Area Name	Applicable Mode
Auxiliary	12152	Primary Sample Room	3
	12252	Rad Chem Lab	3
	12321	Non-1E Equipment/ Penetration Room	4
	12305	Division D I&C Penetration Room	4
	12304	Division B I&C Penetration Room	4
	12561	CCS Valve Room (Note 9)	4
Annex	4000	Annex Building (Note 9)	4
Containment	11209	CVS Room	4

Notes

Note 1: The Emergency Director 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, RS1 and RG1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

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.

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

Note 7: A manual action is **any** operator action, or set of actions, taken in the Main Control Room workstations 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: This EAL does **not** apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 9: Classification is **not** required if either train of RNS can be placed in service for Shutdown Cooling.

Prepared for SNC by: Operations Support Services, Inc. – 2/3/16

UNITS 3 & 4		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
S System Malfunct.	1 Loss of Electrical Power	None	SS1 Loss of **all** required Class 1E DC power or Class 1E UPS bus power for 15 minutes or longer 1 2 3 4 (1) Indicated voltage < 210 VDC on **all** required Class 1E 24-hr. DC buses (IDSA/B/C/D-DS-1) for ≥ 15 min. (Note 1) (2) Loss of power to **all** 24-hr. Class 1E UPS buses (IDSA/B/C/D-EA-1) for ≥ 15 min. (Note 1)	SA1 Loss of **all** capability to charge at least one Class 1E 24-hr. DC battery for 60 minutes or longer 1 2 3 4 (1) Loss of **all** capability to charge at least one Class 1E 24-hr. DC battery for ≥ 60 min. (Note 1)	SU1 Loss of **all** capability to charge at least one Class 1E 24-hr. DC battery for 30 minutes or longer 1 2 3 4 (1) Loss of **all** capability to charge at least one Class 1E 24-hr. DC battery for ≥ 30 min. (Note 1)
	2 Loss of Monitoring & Control Functions	None	SS2 Complete loss of monitoring or control functions for 15 minutes or longer 1 2 3 4 (1) Loss of ability to monitor or control **all** of the following key safety functions from the Main Control Room for ≥ 15 min. (Note 1): • Reactivity Control • Core Cooling • RCS Heat Removal • Spent Fuel Pool Level and Temperature Control	SA2 UNPLANNED partial loss of monitoring or control functions for 15 minutes or longer 1 2 3 4 (1) UNPLANNED loss of the ability to monitor or control one or more of the following key safety functions from the Main Control Room required for the current plant operating mode for ≥ 15 min. (Note 1): • Reactivity Control • Core Cooling • RCS Heat Removal • Spent Fuel Pool Level and Temperature Control	None
	3 RCS Activity	None	None	None	SU3 Reactor coolant activity greater than Technical Specification allowable limits 1 2 3* * ≥ 500 °F (1) Primary Sampling Liquid Radiation Monitor (PSS-RY050) high alarm (2) Sample analysis indicates that a reactor coolant activity value > than an allowable limit specified in Technical Specifications (TS 3.4.10)
	4 RCS Leakage	None	None	None	SU4 RCS leakage for 15 minutes or longer 1 2 3 4 (1) RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. (2) RCS identified leakage > 25 gpm for ≥ 15 min. (3) Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min. (Note 1)
	5 RTS Failure	None	SS5 Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal 1 2 (1) An automatic or manual trip did **not** reduce reactor power < 5% AND **All** manual actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5% AND EITHER of the following conditions exist: • CSFST Core Cooling-**RED** path conditions met • CSFST Heat Sink-**RED** path conditions met	SA5 Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the control room workstations are **not** successful in shutting down the reactor 1 2 (1) An automatic or manual trip did **not** reduce reactor power < 5% AND Subsequent manual actions taken at the Main Control Room workstations are **not** successful in reducing reactor power < 5% (Note 7)	SU5 Automatic or manual trip fails to shut down the reactor 1 2 (1) An automatic trip did **not** reduce reactor power to < 5% AND A subsequent manual action taken at the Main Control Room workstations is successful in reducing reactor power < 5% (Note 7) (2) A manual trip did **not** reduce reactor power to < 5% AND A subsequent automatic trip or manual trip action taken at the Main Control Room workstations is successful in reducing reactor power < 5% (Note 7)
	6 Loss of Comm.	None	None		SU6 Loss of all onsite or offsite communications capabilities 1 2 3 4 (1) Loss of **all** Table C-3 onsite communication methods (2) Loss of **all** Table C-3 ORO communication methods (3) Loss of **all** Table C-3 NRC communication methods
	7 CTMT Isolation Or Pressure Control Failure	None	None	None	SU7 Failure to isolate containment or loss of containment pressure control 1 2 3 4 (1) Failure of containment to isolate when required by an actuation signal AND At least one isolation valve in each penetration is **not** closed within 15 min. of the actuation (Note 1) (2) Containment pressure > 6.2 psig (High 2) AND PCS flow **cannot** be established within 15 min. (Note 1)
	8 Hazardous Event Affecting Safety System	None	None	SA8 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode 1 2 3 4 (1) The occurrence of **any** Table S-2 hazardous event AND EITHER of the following: • 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	
F Fission Product Barrier Degradation		FS1 Loss of **any** two barriers and loss or potential loss of the third barrier 1 2 3 4 (1) Loss of **any** two barriers AND Loss or potential loss of third barrier (Table F-1)	FS1 Loss or potential loss of **any** two barriers 1 2 3 4 (1) Loss or potential loss of **any** two barriers (Table F-1)	FA1 **Any** loss or **any** potential loss of either Fuel Clad or RCS 1 2 3 4 (1) **Any** loss or **any** potential loss of either Fuel Clad or RCS (Table F-1)	None

Table S-1 Communications Methods

System	Onsite	ORO	NRC
Plant Page	X		
VEGP Dial	X		
Dedicated Dial	X		
In-Plant Radio	X		
Commercial Phones	X	X	X
Southern Company Communications	X	X	X
ENN		X	
ENS			X

Table S-2 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

Notes

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

Note 7: A manual action is **any** operator action, or set of actions, taken in the Main Control Room workstations 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

Table F-1 Fission Product Barrier Matrix

	Fuel Clad Barrier		RCS Barrier		Containment Barrier	
	Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
1. RCS or SG Tube Leakage	None	A. RCS hot leg level < 0.5%	A. An automatic or manual safeguards actuation required by EITHER: • UNISOLABLE RCS leakage • SG tube RUPTURE B. Automatic or manual actuation of ADS	A. Operation of a second CVS makeup pump is required by EITHER: • UNISOLABLE RCS leakage • SG tube leakage B. CSFST RCS Integrity-**RED** path conditions met	A. A leaking or RUPTURED SG is FAULTED outside of containment	None
2. Inadequate Heat Removal	A. CSFST Core Cooling-**RED** path conditions met	A. CSFST Core Cooling-ORANGE path conditions met B. CSFST Heat Sink-**RED** path conditions met AND Heat sink required	None	A. CSFST Heat Sink-**RED** path conditions met AND Heat sink required	None	A. CSFST Core Cooling-**RED** path conditions met AND Restoration procedures **not** effective within 15 min. (Note1)
3. RCS Activity / CTMT Radiation	A. PXS-RY160/161/162/163 CTMT High Range Radiation > [site-specific] R/hr B. Dose equivalent I-131 coolant activity > 300 μCi/gm	None	A. PXS-RY160/161/162/163 CTMT High Range Radiation > [site-specific] R/hr	None	None	A. PXS-RY160/161/162/163 CTMT High Range Radiation > [site-specific] R/hr
4. CTMT Integrity or Bypass	None	None	None	None	A. Containment isolation is required AND EITHER: • Containment integrity has been lost based on ED judgment • UNISOLABLE pathway from containment to the environment exists B. Indications of RCS leakage outside of containment	A. CSFST Containment-**RED** path conditions met B. Containment hydrogen concentration (VLS-AE001/002/003) > 4% C. Containment pressure > 6.2 psig (High 2) AND PCS flow **cannot** be established within 15 min. (Note 1)
5. ED Judgment	A. **Any** condition in the opinion of the ED that indicates loss of the fuel clad barrier	A. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier	A. **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	A. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	A. **Any** condition in the opinion of the Emergency Director that indicates loss of the containment barrier	A. **Any** condition in the opinion of the Emergency Director that indicates potential loss of the containment barrier

Modes:

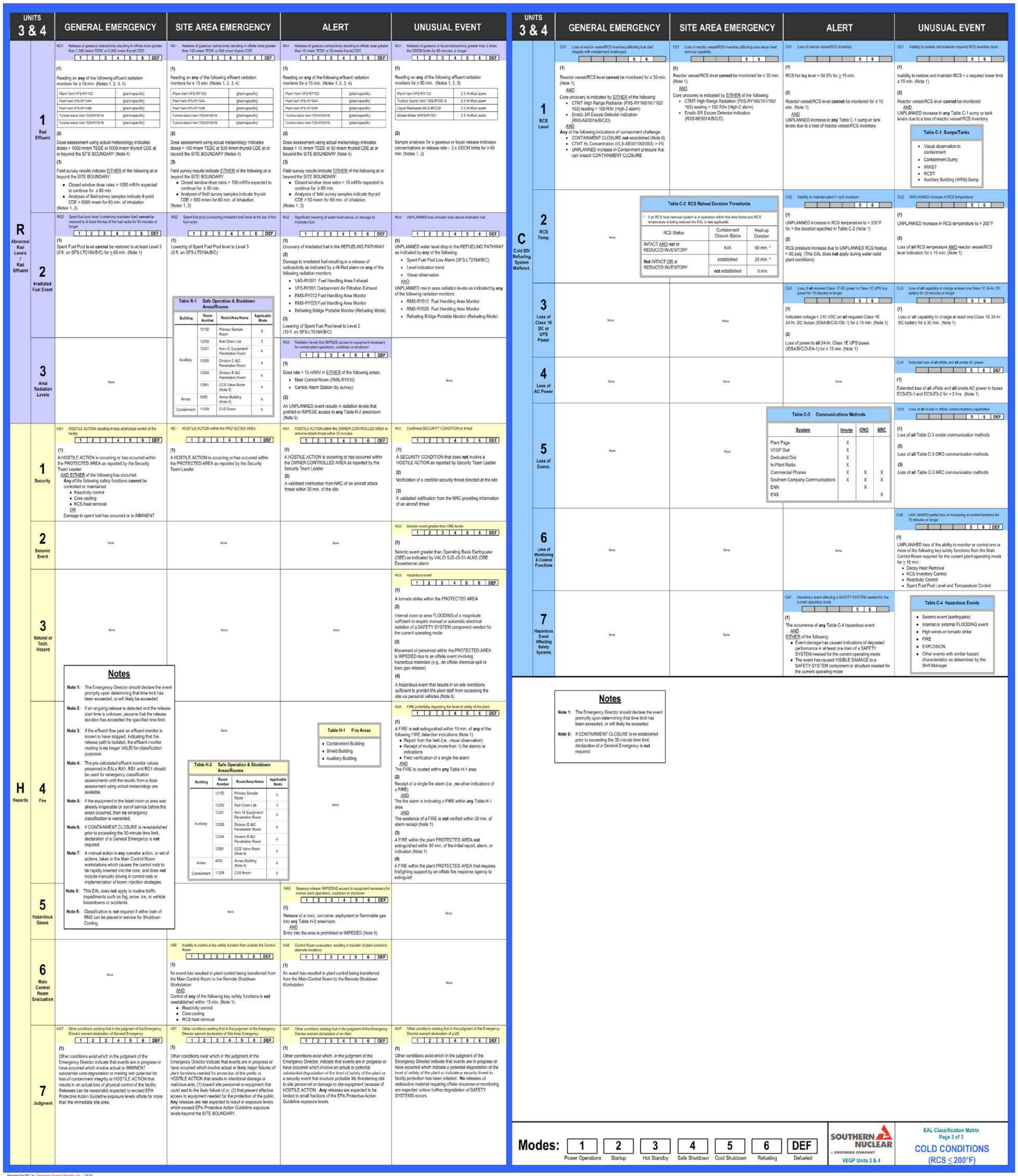
1	2	3	4	5	6	DEF
Power Operations	Startup	Hot Standby	Safe Shutdown	Cold Shutdown	Refueling	Defueled

SOUTHERN NUCLEAR
A SOUTHERN COMPANY
VEGP Units 3 & 4

EAL Classification Matrix
Page 1 of 2
HOT CONDITIONS
(RCS > 200°F)

Cold Condition EAL Matrices

For Information Only



Southern Nuclear Operating Company

ND-16-0150

Enclosure 4

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Proposed License Condition Changes

for Combined License Numbers NPF-91 and NPF-92

(LAR-16-002)

(Enclosure 4 consists of 3 pages, including this cover page)

Mark-up of License Condition 2.D(12)(d) of Combined License Number NPF-91 for VEGP Unit 3

****

(d) No later than 180 days before initial fuel load, SNC shall submit to the Director of NRO, or the Director's designee, in writing, a fully developed set of plant-specific emergency action levels (EALs) for VEGP Unit 3 in accordance with the criteria defined in Amendment No. XX ~~Nuclear Energy Institute (NEI) 07-01, "Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors," Revision 0, with no deviations~~. The EALs shall have been discussed and agreed upon with State and local officials.

No later than 180 days before initial fuel load, SNC shall submit to the Director of NRO, or the Director's designee, in writing, an assessment of emergency response staffing performed in accordance with NEI 10-05, "Assessment of On-Shift Emergency Response Organization Staffing and Capabilities," Revision 0.

****

Mark-up of License Condition 2.D(12)(d) of Combined License Number NPF-92 for VEGP Unit 4

****

(d) No later than 180 days before initial fuel load, SNC shall submit to the Director of NRO, or the Director's designee, in writing, a fully developed set of plant-specific emergency action levels (EALs) for VEGP Unit 4 in accordance with the criteria defined in Amendment No. XX ~~Nuclear Energy Institute (NEI) 07-01, "Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors," Revision 0, with no deviations~~. The EALs shall have been discussed and agreed upon with State and local officials.

No later than 180 days before initial fuel load, SNC shall submit to the Director of NRO, or the Director's designee, in writing, an assessment of emergency response staffing performed in accordance with NEI 10-05, "Assessment of On-Shift Emergency Response Organization Staffing and Capabilities," Revision 0.

****

Southern Nuclear Operating Company

ND-16-0150

Enclosure 5

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Correlation Matrix for the Proposed EALs – For Information Only

(LAR-16-002)

(Enclosure 5 contains 5 pages, including this cover page)

The following is an information only reference correlating Vogtle Electric Generating Plant (VEGP) Units 3 and 4, NEI 07-01 Rev. 0, and NEI 99-01 Rev. 6 EALs.

VEGP Units 3 and 4 EAL #	**07-01 R0 EAL #**	**99-01 R6 EAL #**
RU1 EAL#1	AU1 (1) & (2)	AU1(2)
RU1 EAL#2	AU1(3)	AU1(3)
RA1 EAL#1	AA1 (1) & (2)	AA1(1)
RA1 EAL#2	AA1(5)	AA1(2)
RA1 EAL#3	AA1(4)	AA1(4)
RS1 EAL#1	AS1(1)	AS1(1)
RS1 EAL#2	AS1(2)	AS1(2)
RS1 EAL#3	AS1(4)	AS1(3)
RG1 EAL#1	AG1(1)	AG1(1)
RG1 EAL#2	AG1(2)	AG1(2)
RG1 EAL#3	AG1(4)	AG1(3)
RU2 EAL#1	AU2(1)	AU2(1)
RA2 EAL#1	AA2(1)	AA2(1)
RA2 EAL#2	AA2(2)	AA2(2)
RA2 EAL#3	AU2(1)	AA2(3)
RS2 EAL#1		AS2(1)
RG2 EAL#1		AG2(1)
RA3 EAL#1	AA3(1)	AA3(1)
RA3 EAL#2		AA3(2)
CU1 EAL#1	CU1(1)	CU1(1)
CU1 EAL#2	CU1(1) & (2)	CU1(2)
CA1 EAL#1	CA1(1)	CA1(1)
CA1 EAL#2	CA1(2)	CA1(2)
CS1 EAL#1	CS1(3)	CS1(3)
CG1 EAL#1	CG1(1)	CG1(2)
CU2 EAL#1	CU4(1)	CU3(1)
CU2 EAL#2	CU4(2)	CU3(2)
CA2 EAL#1	CA4(1)	CA3(1)
CA2 EAL#2	CA4(2)	CA3(2)
CU3 EAL#1	CU3(1)	
CA3 EAL#1	CA3(1)	CU4(1)
CA3 EAL#2	CA3(2)	
CU4 EAL#1		CA2(1)
CU5 EAL#1	CU6(1) & (2)	CU5(1), (2), & (3)
CU6 EAL#1	CU7(1)	
CA7 EAL#1	HA1(1), (2), (3), (4), & (5) and HA2(1)	CA6(1)
HU1 EAL#1	HU4(1), (2), & (3)	HU1(1)
HA1 EAL#1	HA4(1) & (2)	HA1(1)
HS1 EAL#1	HS4(1)	HS1(1)

VEGP Units 3 and 4 EAL #	**07-01 R0 EAL #**	**99-01 R6 EAL #**
HG1 EAL#1	HG1(1) & (2)	HG1(1)
HU2 EAL#1	HU1(1)	HU2(1)
HU3 EAL#1	HU1(2)	HU3(1)
HU3 EAL#2		HU3(2)
HU3 EAL#3	HU3(2)	HU3(3)
HU3 EAL#4		HU3(4)
HU4 EAL#1	HU4(1)	HU4(1)
HU4 EAL#2	HU4(1)	HU4(2)
HU4 EAL#3	HU4(1)	HU4(3)
HU4 EAL#4	HU4(1)	HU4(4)
HA5 EAL#1	HA3(1)	HA5(1)
HA6 EAL#1	HA5(1)	HA6(1)
HS6 EAL#1	HS6(1)	HS6(1)
HU7 EAL#1	HU5(1)	HU7(1)
HA7 EAL#1	HA6(1)	HA7(1)
HS7 EAL#1	HS3(1)	HS7(1)
HG7 EAL#1	HG2(1)	HG7(1)
SU1 EAL#1	SU1(1)	SU1(1)
SA1 EAL#1	SA1(1)	SA1(1)
SS1 EAL#1	SS1(1)	SS1(1)
SS1 EAL#2	SS1(2)	SS1(1)
SA2 EAL#1	SA7(1)	SU2(1)
SS2 EAL#1	SS7(1)	SA2(1)
SU3 EAL#1	SU4(1)	SU3(1)
SU3 EAL#2	SU4(2)	SU3(2)
SU4 EAL#1	SU5(1) & (2)	SU4(1), (2), & (3)
SU5 EAL#1	SA2(1)	SU5(1)
SU5 EAL#2		SU5(2)
SA5 EAL#1	SS2(1)	SA5(1)
SS5 EAL#1		SS5(1)
SU6 EAL#1	SU6(1) & (2)	SU6(1), (2), & (3)
SU7 EAL#1	CTMT BARRIER 5A	SU7(1)
SU7 EAL#2	CTMT BARRIER 2A	SU7(2)
SA8 EAL#1	HA1(1), (2), (3), (4), & (5) AND HA2(1)	SA9(1)
FA1 EAL#1	FA1	FA1
FS1 EAL#1	FS1	FS1
FG1 EAL#1	FG1	FG1
TABLE F-1	TABLE 5-7-3	TABLE F-1
FUEL CLAD – 1. RCS OR SG TUBE LEAKAGE - LOSS	N/A	N/A
FUEL CLAD – 1. RCS OR SG TUBE LEAKAGE – POTENTIAL LOSS A	FC POT. LOSS 4A	FC POT. LOSS 1A
FUEL CLAD – 2. INADEQUATE HEAT REMOVAL - LOSS A	FC LOSS 1A & 3A	FC LOSS 2A
FUEL CLAD – 2. INADEQUATE HEAT REMOVAL –	FC POT. LOSS 1A & 3A	FC POT. LOSS 2A

VEGP Units 3 and 4 EAL #	**07-01 R0 EAL #**	**99-01 R6 EAL #**
POTENTIAL LOSS A		
FUEL CLAD – 2. INADEQUATE HEAT REMOVAL – POTENTIAL LOSS B	FC POT. LOSS 1B	FC POT. LOSS 2B
FUEL CLAD – 3. RCS ACTIVITY / CTMT RADIATION – LOSS A	FC LOSS 6A	FC LOSS 3A
FUEL CLAD – 3. RCS ACTIVITY / CTMT RADIATION – LOSS B	FC LOSS 2A	FC LOSS 3B
FUEL CLAD – 4. CTMT INTEGRITY OR BYPASS - LOSS	N/A	N/A
FUEL CLAD – 4. CTMT INTEGRITY OR BYPASS – POTENTIAL LOSS	N/A	N/A
FUEL CLAD – 5. ED JUDGEMENT - LOSS A	FC LOSS 8	FC LOSS 6A
FUEL CLAD – 5. ED JUDGEMENT – POTENTIAL LOSS A	FC POT. LOSS 8	FC POT. LOSS 6A
REACTOR COOLANT SYSTEM – 1. RCS OR SG TUBE LEAKAGE – LOSS A	RCS LOSS 2A	RCS LOSS 1A
REACTOR COOLANT SYSTEM – 1. RCS OR SG TUBE LEAKAGE – LOSS B	RCS LOSS 2B	
REACTOR COOLANT SYSTEM – 1. RCS OR SG TUBE LEAKAGE – POTENTIAL LOSS A	RCS POT. LOSS 1A	RCS POT. LOSS 1A
REACTOR COOLANT SYSTEM – 1. RCS OR SG TUBE LEAKAGE – POTENTIAL LOSS B		RCS POT. LOSS 1B
REACTOR COOLANT SYSTEM – 2. INADEQUATE HEAT REMOVAL - LOSS	N/A	N/A
REACTOR COOLANT SYSTEM – 2. INADEQUATE HEAT REMOVAL – POTENTIAL LOSS A	RCS POT. LOSS 1B	RCS POT. LOSS 2A
REACTOR COOLANT SYSTEM – 3. RCS ACTIVITY / CTMT RADIATION - LOSS A	RCS LOSS 6A	RCS LOSS 3A
REACTOR COOLANT SYSTEM – 3. RCS ACTIVITY / CTMT RADIATION – POTENTIAL LOSS	N/A	N/A
REACTOR COOLANT SYSTEM – 4. CTMT INTEGRITY OR BYPASS - LOSS	N/A	N/A
REACTOR COOLANT SYSTEM – 4. CTMT INTEGRITY OR BYPASS – POTENTIAL LOSS	N/A	N/A
REACTOR COOLANT SYSTEM – 5. ED JUDGEMENT – LOSS A	RCS LOSS 8	RCS LOSS 6A
REACTOR COOLANT SYSTEM – 5. ED JUDGEMENT – POTENTIAL LOSS A	RCS POT. LOSS 8	RCS POT. LOSS 6A
CONTAINMENT – 1. RCS OR SG TUBE LEAKAGE - LOSS A	CTMT LOSS 4A & B	CTMT LOSS 1A
CONTAINMENT – 1. RCS OR SG TUBE LEAKAGE – POTENTIAL LOSS	N/A	N/A
CONTAINMENT – 2. INADEQUATE HEAT REMOVAL - LOSS	N/A	N/A
CONTAINMENT – 2. INADEQUATE HEAT REMOVAL – POTENTIAL LOSS A	CTMT POT. LOSS 3A & B	CTMT POT. LOSS 2A
CONTAINMENT – 3. RCS ACTIVITY / CTMT RADIATION - LOSS		N/A

CONTAINMENT – 3. RCS ACTIVITY / CTMT RADIATION – POTENTIAL LOSS A		CTMT POT. LOSS 3A
CONTAINMENT – 4. CTMT INTEGRITY OR BYPASS – LOSS A	CTMT LOSS 2A & 5A	CTMT LOSS 4A
CONTAINMENT – 4. CTMT INTEGRITY OR BYPASS – LOSS B		CTMT LOSS 4B
CONTAINMENT – 4. CTMT INTEGRITY OR BYPASS – POTENTIAL LOSS A	CTMT POT. LOSS 2A	CTMT POT. LOSS 4A
CONTAINMENT – 4. CTMT INTEGRITY OR BYPASS – POTENTIAL LOSS B	CTMT POT. LOSS 2B	CTMT POT. LOSS 4B
CONTAINMENT – 4. CTMT INTEGRITY OR BYPASS – POTENTIAL LOSS C	CTMT POT. LOSS 2B	CTMT POT. LOSS 4C
CONTAINMENT – 5. ED JUDGEMENT – LOSS A	CTMT LOSS 8	CTMT LOSS 6A
CONTAINMENT – 5. ED JUDGEMENT – POTENTIAL LOSS A	CTMT POT. LOSS 8	CTMT POT. LOSS 6A

Southern Nuclear Operating Company

ND-16-0150

Enclosure 6

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Westinghouse Review Results for the Proposed EALs

(LAR-16-002)

(Enclosure 6 contains 17 pages, including this cover page)

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Plant Vogtle Units 3 & 4
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Mr. Brian Whitley
ND Regulatory Affairs
Southern Nuclear Operating Company

VIA: E-Mail

Our Reference: SVP_SV0_003685

February 4, 2016

Subject: **Response to Southern Nuclear Company (SNC) Request for Emergency Action Level (EAL) Technical Review Final Acceptance**

Reference: **1) Engineering, Procurement, and Construction Agreement for AP1000® Nuclear Power Plants, Dated May 23, 2008 – V.C. Summer Units 2 and 3 ("Agreement")**

2) Letter Dated November 5, 2015 from April R. Rice (SCE&G) to Brian McIntyre (Westinghouse), Subject: V.C. Summer (VCS) Units 2 and 3 Emergency Action Level (EAL) Technical Review Final Acceptance (NND-15-0641)

Attachment: **1) Updated Consolidation Comment Matrix from VCS, WEC, and Vogtle**

Action: **None**

Dear Mr. Whitley:

Westinghouse Electric Company LLC is submitting, in accordance with Reference 1, the response to the request to obtain concurrence the revised EAL Technical Bases Document provided by Reference 2 is still consistent with the design basis of the Westinghouse **AP1000** (DCD Rev. 19).

The consolidated comment matrix provided as Enclosure 1 of Reference 2 has been reviewed and provided, with comments, as Attachment 1 of this letter. Please contact me at (706) 437-7583 if you have any questions.

Sincerely,

Gerard F. Couture

Gerard F. Couture
Consortium Licensing Director
Vogtle Units 3 & 4

cc:	Michael Yox	- SNC
	Kelli A Roberts	- SNC
	Daniel Mickinac	- SNC
	Amanda Pugh	- SNC
	John Crenshaw	- WEC
	Brian McIntyre	- WEC
	Paul Russ	- WEC
	Lee Woodcock	- WEC
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	VogtleProject@westinghouse.com	
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Section/ IC/EAL#	WEC Comments	VCS Response	WEC Follow-up Response	Vogtle 3&4 Comments	VCS/Vogtle Recommended Final Resolution
2.2B	A2. Added the following to the definition of 2.2B. Reactor Coolant System (RCS): PRHR, CMT Purification as these are major systems that are connected to the RCS with isolation valves open during operations. made some changes to PZR safeties and added ADS.	The definition of 2.2B. Reactor Coolant System (RCS) will be worded instead to say "The RCS Barrier includes the RCS primary side and its connections up to and including the reactor coolant pressure boundary isolation valves".	Yes	Were references changed to also recognize Tech Spec LCO 3.5.15 for PIV's? LCO 3.4.7 is referenced. We want to be careful with this, as RCS LEAKAGE does not occur until BOTH valves leak by.	The definition of 2.2B. Reactor Coolant System (RCS) will be worded instead to say "The RCS Barrier includes the RCS primary side and its connections up to and including the reactor coolant pressure boundary isolation valves".
2.2C	A3. The use of "containment building" in "2.2C Containment" could be confused with the shield building.	The decision was made to go with "containment" only and not use containment enclosure or containment building to help define the structure.	Yes	None	The decision was made to go with "containment" only and not use containment enclosure or containment building to help define the structure.
2.6 #6	A6. Minor changes made for consistency with TS.	Incorporate WEC comments Changed "6 Refueling" from "with the vessel head closure bolts" to "with one or more vessel head closure bolts"	N/A	None	Incorporate WEC comments.
3.2.6 para 4	**New Change #1 Proposed by VCS and Vogtle for WEC Review:** 3.2.6 Example provided: "For illustrative purposes, consider the following example:" Though the example is appropriate for a legacy PWR, it is not appropriate for an AP1000 which does not have "the Emergency Feedwater system". Consider modifying the example to include an ATWS occurs and PRHR fails to automatically actuate. Also,	Revised as suggested. First sentence was changed from "and the Emergency Feedwater System fails to automatically start" to "and PRHR fails to automatically actuate". Third sentence was changed from "If an operator manually starts the Emergency Feedwater system in accordance" to	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Revised as suggested.	No further changes are necessary.

Section/ IC/EAL#	WEC Comments	VCS Response	WEC Follow-up Response	Vogtle 3&4 Comments	VCS/Vogtle Recommended Final Resolution
	a similar change is needed in the third sentence.	"If an operator manually actuates the PRHR (either by PMS or DAS) in accordance".			
5.1 Contain-ment Closure	A7. Changed to the wording in the bases for TS 3.6.8. Containment closure is required for TS 3.5.5, 3.6.8 and 3.9.	Remove reference to T/S and incorporate the changes for the definition of "Containment Closure"	Yes	None	Remove reference to T/S and incorporate the changes for the definition.
5.1 Ruptured	**New Change #2 Proposed by VCS and Vogtle for WEC Review:** 5.1 Definitions – RUPTURED includes the term safety injection. Consider changing the term to Safeguards actuation.	Revised as suggested at the end of the sentence.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Revised as suggested.	No further changes are necessary.
5.2 PDSP	**New Change #3 Proposed by VCS and Vogtle for WEC Review:** 5.2 Acronyms & Abbreviations – PDSP should be "Primary Dedicated Safety Panel" not "System"	Revised as suggested.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Revised as suggested.	No further changes are necessary.
RU2.1 Boxed Area and Basis para 2	A8. Added level instruments for the low alarms and added "OR" so that either is applied to level alarm or visual indication.	Incorporate WEC comments. See also New Change #4 for additional changes.	N/A	None	Incorporate WEC comments.

Section/ IC/EAL#	WEC Comments	VCS Response	WEC Follow-up Response	Vogtle 3&4 Comments	VCS/Vogtle Recommended Final Resolution
RU2.1 Boxed Area and Basis para 2	**New Change #4 Proposed by VCS and Vogtle for WEC Review:** RU2.1 Basis: Makeup would be initiated based on level trend or alarm. The first level alarm or trend would be received from non-safety related NR level indication SFS-LT025 with low alarm setpoint at 24.6' above TOF. Safety related level transmitters SFS-LT019A/B/C would provide alarm and Refueling Isolation at setpoint of 23.5' above TOF. Consider referencing their use.	Added reference to level trend and non safety-related level instruments.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Added reference to level trend and non safety-related level instruments.	No further changes are necessary.
RU2.1 Basis para 3	A9. Consider adding level instrumentation MEL identification like Rad Monitors in the following paragraph.	Incorporate WEC comments. First sentence added "(SFS-LT019A/B/C)"	N/A	None	Incorporate WEC comments.
RA3.2	Regarding EAL RA3.2 - There are valves in the CVS system that are required to be accessed for normal operations and heat up and cool down that must be manually operated. CVS- V043 letdown orifice bypass valve in room 11209 must be opened and closed based on RCS pressure during startup and shutdown. CVS-MT-03 room 12255 Chemical mixing tank needs to have access to add LiOH and H2O2 to control RCS pH and oxygen. I read NEI 99-01 and I am pretty sure that this would meet the requirement for normal plant	RA3.2 will be added to the EAL Matrix and Technical Basis Document and list of rooms developed. Access to room 12255 was determined to not be required, since needed actions can be done remotely from the MCR. See also New Change #5 which does not include room 12255 in Tables R-2 and H-2.	N/A	List of Rooms provided.	RA3.2 will be added to the EAL Matrix and Technical Basis Document.

Section/ IC/EAL#	WEC Comments	VCS Response	WEC Follow-up Response	Vogtle 3&4 Comments	VCS/Vogtle Recommended Final Resolution
	operation and for normal plant cool down.				
RA3.2	**New Change #5 Proposed by VCS and Vogtle for WEC Review:** Although Table R-2/H-2 covers locations that must be accessed for plant shutdown/cooldown, it does not take into account equipment/areas necessary for normal plant operation.	Per analysis by Vogtle and with concurrence of VCS, revised Attachment 3 and Tables R-2 and H-2 as provided and incorporated into latest draft.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Per analysis by Vogtle and with concurrence of VCS, revised Attachment 3 and Tables R-2 and H-2 as provided and incorporated into latest draft.	No further changes are necessary.
CU1.1 Basis	A18. I am not 100% positive but I think that the NEI guidance is based on an UNPLANNED loss of inventory for CU1 and the Plant specific basis is dealing in absolutes (PZR low level 10%) on described levels for Mode 5 and refueling. AP1000 when crediting different systems for protection uses a NR PZR level of 20% with the RCS intact so that PRHR is available for decay heat removal. RCS not intact with PZR NR Level less than 20% IRWST injection with once through cooling is the decay heat removal credited source. This EAL CU1 would be entered if the specified band established for current operations could not be maintained. Also the same would be true for any evolution where a specified band may be established in reduced inventory or Mid Loop condition.	The team believes the original description was correct and would like to know what the thought process was for this conclusion.	This comment was made specifically to the Plant-Specific Basis write-up. For a legacy plant, there is no safety related once through cooling system so in the event of an RCS level loss, loss of shutdown cooling requires an immediate action to establish a once through cooling path. For AP1000, there is a designed safety system to establishing a once through cooling flow path. Low-4 HL level will initiate ADS4 and the IRWST injection valves will open to establish a flow path and cool the core. I thought that the Plant-Specific Basis write- up should include some of the information specific to	Concur with addition of discussion to basis.	Additional guidance will be placed in the basis based on WEC comments. Added new second paragraph to Basis.

Section/ IC/EAL#	WEC Comments	VCS Response	WEC Follow-up Response	Vogtle 3&4 Comments	VCS/Vogtle Recommended Final Resolution
			AP1000. Note: After looking at this again this might be better added to the Plant- Specific basis for CA1.1.		
CU1.2	**New Change #6 Proposed by VCS and Vogtle for WEC Review:** Should Table C-1 also include Aux Building (WRS) Sump. It is very possible for an RCS leak through RNS to go into the aux bldg (WRS) sump.	Added Auxiliary Building (WRS) Sump to Table C-1.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Added Auxiliary Building (WRS) Sump to Table C-1.	No further changes are necessary.
CA1.1 Boxed area	**New Change #7 Proposed by VCS and Vogtle for WEC Review:** CA1.1 Alert definition – Consider changing "Reactor vessel/RCSlevel < 64.5 % for > 15 min." to "RCS Hot Leg level < 64.5% for > 15 min."	Revised as suggested.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Revised as suggested.	No further changes are necessary.
CA1.2 Basis para 4	A21. I do not see 30 minutes as an effective precursor. IF you have a loss of all level indication with an increase in sump/tank level, you could immediately transition to CS1.1 on receipt of a containment hi range rad monitor or erratic source range. You would not need fuel clad breach to get the high Rad reading but only a loss of water shielding. Using the shorter 15 minutes provides for additional time to perform corrective actions after an alert is declared.	Incorporate WEC comments. First sentence changed "30-minute duration" to "15-minute duration"	N/A	None	Incorporate WEC comments.
CS1.1 Basis para	**New Change #8 Proposed by VCS and Vogtle for WEC Review:**	Deleted from second sentence "...and actuates	Concur with this change and its inclusion in the	Deleted "...and actuates both RNS and	No further changes are necessary.

Section/ IC/EAL#	WEC Comments	VCS Response	WEC Follow-up Response	Vogtle 3&4 Comments	VCS/Vogtle Recommended Final Resolution
2	"The containment radiation High-2 alarm setpoint is 100 R/hr and actuates both RNS and CVS isolation (ref. 1). " This statement is misleading for the RNS isolation in MODE5/6. RNS isolation on Ctmt High-2 rad is manually blocked per GOP-205 < P-11 and thus will not be active in MODE 5 or 6. Recommend including note that the RNS isolation is blocked.	both RNS and CVS isolation" from CS1.1 and CG1.1 bases.	November 2015 EAL Technical Bases Document.	CVS isolation" from CS1.1 and CG1.1 bases.	
CG1.1 Boxed area	A23. Added plant specific data.	Incorporate WEC comments.	N/A	Why was "SR" deleted?	Incorporate WEC comments Changed second bullet from "Erratic source range monitor" to "Erratic SR Excore Detector indication (RXS-NE001A/B/C/D)"
CG1.1 Boxed area	A24. Added plant specific data.	Incorporate WEC comments.	N/A	None	Incorporate WEC comments Changed fourth bullet from "Containment hydrogen concentration > 4%" to "CTMT H_2 Concentration (VLS-AE001/002/003) > 4%"
CG1.1 Basis para 2	**New Change #9 Proposed by VCS and Vogtle for WEC Review:** "The containment radiation High-2 alarm setpoint is 100 R/hr and actuates both RNS and CVS isolation (ref. 1). " This statement is misleading for the RNS isolation in MODE5/6. RNS isolation on Ctmt High-2 rad is manually blocked per GOP-205 < P-11 and thus will not be active in MODE 5 or 6.	Deleted from second sentence "...and actuates both RNS and CVS isolation" from CS1.1 and CG1.1 bases.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Deleted "...and actuates both RNS and CVS isolation" from CS1.1 and CG1.1 bases.	No further changes are necessary.

Section/ IC/EAL#	WEC Comments	VCS Response	WEC Follow-up Response	Vogtle 3&4 Comments	VCS/Vogtle Recommended Final Resolution
	Recommend including note that the RNS isolation is blocked.				
CU2.2 Basis para 2	**New Change #10 Proposed by VCS and Vogtle for WEC Review:** Should RCS-LT195A/B/C/D be included as one of the RCS level monitoring instruments (if they are on-scale)?	Added Pressurizer Level Narrow Range as third bullet.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Added Pressurizer Level Narrow Range as third bullet.	No further changes are necessary.
CU3.1 Basis para 1	**New Change #11 Proposed by VCS and Vogtle for WEC Review:** CU3.1 Basis, Plant-Specific, first sentence – change Load centers to Motor Control Centers (ECS-EC-121/221).	Revised as suggested.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Revised as suggested.	No further changes are necessary.
CU3.1 Basis para 3	**New Change #12 Proposed by VCS and Vogtle for WEC Review:** The characterization of the safety related DC as 24 hours is misleading. The safety related DC system has both 24 hour batteries and 72 hour batteries Later EALS refer to minimum class 1E which would be more accurate.	Deleted "safety related" from 3rd para first sentence. Deleted 2nd sentence of 3rd para which stated "It also provides power for safe shutdown when all the on-site and off-site AC power sources are lost and cannot be recovered for 72 hours."	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Deleted "safety related" from 3rd para. Deleted 2nd sentence of 3rd para.	No further changes are necessary.
CA3.1 Basis para 1	A29.Does this include operation of the ADS valves or just safety related instrumentation and controls? I am pretty sure that the minimum ICV of 1.75 volts is the lowest that any cell should be operated at ever and below that voltage you stand a chance of reverse powering cells so using a total bus voltage of 210 volts, this means that the average of all 120	The team accepts the original description as correct and any change from the indicated voltage would have to be evaluated by engineering.	I could not find the exact data for the batteries being purchased. The 1.75 volts minimum ICV voltage came from the SSD. Not sure exactly what ICV voltage with the battery under load causes cells to reverse. For a Gould TLX 39B, it is 1.51 volts. I suggest that you		The 210 Volts and 1.75 Volts are supported by UFSAR Table 8.3.2-5 and ITAAC Table 2.6.3-3, so no further evaluation is needed. Also, added a statement in CA3.1 that states: "Permanent battery cell damage begins to occur at battery voltage <210 VDC." See also New Change #13.

Section/ IC/EAL#	WEC Comments	VCS Response	WEC Follow-up Response	Vogtle 3&4 Comments	VCS/Vogtle Recommended Final Resolution
	cells would be 1.75v with some higher and some lower. As soon as you start reverse powering some of the cells all bets are off if there will be sufficient power left for I&C loads and definitely not any ADS valve motor loads. With ADS loads inoperable in Mode 5 RCS not intact or PZR level less than 20%, you will lose the ability for once through cooling using Stage 1-3 ADS valves and have to rely on DAS actuation of the ADS4 valve. I think the 210 volt value needs to be evaluated. I looked at table 8.3.2-5 in the UFSAR and the Battery bank will be at minimum ICV after discharging 8 hours and the battery 8 hour discharge rate.		get electrical engineering to look at this number. Updated response: Agree with the Final Resolution for WEC item A29. Verified that the Licensing requirement for DC MOVs used by ADS stage 1-3 valves to be capable of performing the safety related design function with the DC voltage supply at 210 volts. This is verified during ITAAC testing. These requirements are listed in UFSAR Table 8.3.2-5 and ITAAC Table 2.6.3-3. This is enhanced with the addition of New Change #13.		
CA3.1 Basis para 5	**New Change #13 Proposed by VCS and Vogtle for WEC Review:** CA3.1 Basis – include permanent battery cell damage begins to occur at battery voltage < 210 vdc.	Added "Permanent battery cell damage begins to occur at battery voltage < 210 VDC" to CA3.1 Basis paragraph 5 as last sentence.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Added "Permanent battery cell damage begins to occur at battery voltage < 210 VDC" to CA3.1 Justification.	No further changes are necessary.
CA3.2 Boxed area and Basis para 1	**New Change #14 Proposed by VCS and Vogtle for WEC Review:** EAL needs to either specify 24 hr UPS busses only or also include IDSB/C-EA-3.	Added "24-hr" in boxed area and Basis paragraph one first sentence.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Added "24-hr."	No further changes are necessary.
CA3.2	**New Change #15 Proposed by VCS**	Changed to UPS.	Concur with this change	Changed to UPS.	No further changes are

Section/ IC/EAL#	WEC Comments	VCS Response	WEC Follow-up Response	Vogtle 3&4 Comments	VCS/Vogtle Recommended Final Resolution
Basis para 3	**and Vogtle for WEC Review:** "The purpose of this IC/EAL and its associated EALs is to recognize a loss of the Class 1E 24-hr. DC." DC should be UPS in first sentence.		and its inclusion in the November 2015 EAL Technical Bases Document.		necessary.
CA4.1 Deleted	A30. I am having a hard time seeing this IC as anything higher than an UE as the plant is designed to cope without power for 72 hours. There are a number of upgrades to alert that would happen as a result of this IC. Most would be based on inability to remove RCS decay heat an upgrade would be to EAL CA2.1.	Incorporate WEC comments. Changed from "CA4.1 Alert" to "CU4.1 Unusual Event".	N/A	None	Incorporate WEC comments.
HU2.1 Boxed area	**New Change #16 Proposed by VCS and Vogtle for WEC Review:** Initiating Condition is "Seismic Event greater than OBE levels". Why aren't we using MCR SJS alarm SJS-JS-01-ALM2 "OBE Exceedance" that provides instantaneous indication in the MCR when OBE earthquake threshold has been exceeded?	Revised threshold to "Seismic event greater than Operating Basis Earthquake (OBE) as indicated by valid SJS-JS-01-ALM2 (OBE Exceedance) alarm"	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Revised threshold to "Seismic event greater than Operating Basis Earthquake (OBE) as indicated by valid SJS-JS-01-ALM2 (OBE Exceedance) alarm"	No further changes are necessary.
HA5.1	Regarding EAL HA5.1 - There are valves in the CVS system that are required to be accessed for normal operations and heat up and cool down that must be manually operated. CVS- V043 letdown orifice bypass valve in room 11209 must be opened and closed based on RCS pressure during startup and shutdown. CVS-MT-03 room 12255	HA5.1 will be added to the EAL Matrix and Technical Basis Document and list of rooms developed. See entries for RA3.2.	N/A	List of Rooms provided.	HA5.1 will be added to the EAL Matrix and Technical Basis Document.

Section/ IC/EAL#	WEC Comments	VCS Response	WEC Follow-up Response	Vogtle 3&4 Comments	VCS/Vogtle Recommended Final Resolution
	Chemical mixing tank needs to have access to add LiOH and H2O2 to control RCS pH and oxygen. I read NEI 99-01 and I am pretty sure that this would meet the requirement for normal plant operation and for normal plant cool down.				
HA5.1	**New Change #17 Proposed by VCS and Vogtle for WEC Review:** Although Table R-2/H-2 covers locations that must be accessed for plant shutdown/cooldown, it does not take into account equipment/areas necessary or normal plant operation.	Per analysis by Vogtle and with concurrence of VCS, revised Attachment 3 and Tables R-2 and H-2 as provided and incorporated into latest draft.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Per analysis by Vogtle and with concurrence of VCS, revised Attachment 3 and Tables R-2 and H-2 as provided and incorporated into latest draft.	No further changes are necessary.
SU1.1 Basis para 1	**New Change #18 Proposed by VCS and Vogtle for WEC Review:** SU1.1 Basis, Plant-Specific, first sentence – change “Load centers” to “Motor Control Centers (ECS-EC-121/221)”	Revised as suggested.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Revised as suggested.	No further changes are necessary.
SA1.1 Basis para 1	**New Change #19 Proposed by VCS and Vogtle for WEC Review:** SA1.1 Basis, Plant-Specific, first sentence – change “Load centers” to “Motor Control Centers (ECS-EC-121/221)”	Revised as suggested.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Revised as suggested.	No further changes are necessary.
SA1.1 Basis para 2	A36. Unless you lose a battery bank, the redundancy is fine. What we are dealing with here is loss of battery capacity as it discharges.	Incorporate WEC comments. In the second sentence, changed “chargers reduces required redundancy and potentially degrades” to “chargers degrades”	N/A	None	Incorporate WEC comments.

Section/ IC/EAL#	WEC Comments	VCS Response	WEC Follow-up Response	Vogtle 3&4 Comments	VCS/Vogtle Recommended Final Resolution
SS1.1 Basis para 1	A39. See the previous comments made on CA3.1 about the 210 VDC minimum DC bus voltage.	The team accepts the original description as correct and any change from the indicated voltage would have to be evaluated by engineering.	See item A29.		See item A29 (EAL CA3.1)
SS1.2 Basis para 4	A40. The ultimate heat sink is the air around containment and loss of 1E DC power does not change this. In a legacy plant SWS and CCS would have been required but for AP1000, heat transfer from the core is to IRWST/Containment sump water to containment to the atmosphere.	Incorporate WEC comments. In the second sentence, delete “and the ultimate heat sink “ from the end of the sentence.	N/A	None	Incorporate WEC comments.
SU3.1 Basis para 3	A47. Why is this excluded from mode 4? NEI-9901 has Mode 4 included. Unlike EAL SU 3.2, The rad monitor will alarm in Mode 4 and it needs to be evaluated. Crud bursts would most likely not give an alarm in Mode 4 so an alarm is still indicative of a cladding breach.	Amend to make mode 3 applicable as determined by team.	I still do not see why this is not applicable in Mode 4. EALs are based on plant conditions and not Tech Specs. A high alarm on the PSS Liquid Monitor Rad Monitor would indicate a plant condition warranting an EAL entry.		Amend to make mode 3 applicable as determined by team. Changed the wording in the first sentence from “limited to Modes 1, 2 and 3 consistent” to “limited to Modes 1, 2 and 3 ($\geq$ 500 ^{o}F) consistent”
SU3.2 Basis para 2 and Reference 2	A49-A50. There are no accidents calculated in TS. Added Reference to UFSAR Chapter 15.	Incorporate WEC comments. In the second sentence, changed from “(ref. 1)” to “(ref. 2)” and added the new Reference 2.	N/A	None	Incorporate WEC comments.
SU5.1 Basis para 6 SU5.2	**New Change #20 Proposed by VCS and Vogtle for WEC Review:** It is unclear whether-or-not credit can be given to manual action in	Added "(DAS, either PDSP trip switch, or opening Rod Drive MG supply and output breakers)" to the	Concur with this change and its inclusion in the November 2015 EAL Technical Bases	Added "(DAS, either PDSP trip switch, or opening Rod Drive MG supply and output	No further changes are necessary.

Section/ IC/EAL#	WEC Comments	VCS Response	WEC Follow-up Response	Vogtle 3&4 Comments	VCS/Vogtle Recommended Final Resolution
Basis para 5 SA5.1 Basis para 3	the MCR to open rod drive MG set supply and output breakers per FR-S.1 step 1.c RNO 3.	first sentence in each of the listed paragraphs.	Document.	breakers)" to the first sentence in each of the listed paragraphs.	
SU7.1 Basis para 2	**New Change #21 Proposed by VCS and Vogtle for WEC Review:** "For this EAL the containment isolation signal must be generated as the result of an off-normal/accident condition (e.g., low pressurizer level or high containment pressure);" Low pressurizer level does not directly cause a T signal; it causes CMT. T signal caused by Safeguards or manual PCS. Reword to low pressurizer pressure.	Revised the first sentence to change the example to "(e.g., automatic or manual Safeguard or PCS signals)"	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Revised the first sentence to change the example to "(e.g., automatic or manual Safeguard or PCS signals)"	No further changes are necessary.
SU7.2 Boxed area	A61. Refer to APP-PCS-M3-001 section 3.1.1 paragraphs on passive containment Heat removal and Table 3-3. This information and the PCS- FT001/2/3/4 need to be added to this discussion to help define adequate flow. As a minimum, Table 3-3 column 1 Minimum flow at elapsed time should be added but a comprehensive discussion on the standpipes and flow instrumentation would be better. Add to plant-specific basis.	Amend description to say "PCS flow cannot be established within 15 min. (Note 1)" and delete "Adequate PCS flow is not established within 15 min. (Note 1)".	Yes	None	Amend description to say "PCS flow cannot be established within 15 min. (Note 1)" and remove references to the indicatorsdelete "Adequate PCS flow is not established within 15 min. (Note 1)".
FC Pot. Loss 1.A	**New Change #22 Proposed by VCS and Vogtle for WEC Review:**	Added suggested paragraph.	Concur with this change and its inclusion in the	Added suggested paragraph.	No further changes are necessary.

Section/ IC/EAL#	WEC Comments	VCS Response	WEC Follow-up Response	Vogtle 3&4 Comments	VCS/Vogtle Recommended Final Resolution
Basis para 3	Something about sustained low hot leg level? Reference E-0 basis and reword as necessary? Add new paragraph to say "Short duration spikes in hot leg levels may occur due to flow and pressure oscillations in the RCS induced by variations in break flow and injection flow. When determining if the threshold is exceeded based on hot leg levels being actually voided, consideration should be given to the duration of the low level condition and the indication of both hot leg level instruments."		November 2015 EAL Technical Bases Document.		
FC Pot. Loss 2.B Boxed area	A66. IF you look at the CSFST, this is just essentially "ONE SG available (Level greater than 21% or feed flow greater than 400 gpm and the SGs are couple to the RCS) OR PRHR is in service and RCS temperatures are being controlled. IF both conditions are not met heat sink is red.	The team will rewrite the Basis third paragraph to give a better explanation of the basis.	N/A	None	The team will rewrite the paragraph to give a better explanation of the basis.
FC Pot. Loss 2.B Basis para 3	A67. Not sure why this is a relaxation?	The team will rewrite the Basis third paragraph to give a better explanation of the basis.	N/A	None	The team will rewrite the paragraph to give a better explanation of the basis
FC Pot. Loss 2.B Basis para 3	A68. For large LOCAs, if the LOCA itself does not depressurize the RCS, eventually ADS4 will depressurize the RCS. PZR level will drop below level where PRHR is effective and cooling will be once	The team will rewrite the Basis third paragraph to give a better explanation of the basis.	N/A	None	The team will rewrite the paragraph to give a better explanation of the basis.

Section/ IC/EAL#	WEC Comments	VCS Response	WEC Follow-up Response	Vogtle 3&4 Comments	VCS/Vogtle Recommended Final Resolution
	through cooling using IRWST/Sump injection to containment heat removal systems.				
RCS Loss 1.A Basis para 3	A69. This has been defined as a Total RCS leak rate less than 100 gpm in AOP-304. (APP-GW-GJP-304 rev. 2) In this case the non- safety CVS makeup which can make up at up to 175 gpm may prevent a Safeguards actuation by holding up RCS pressure.	The team believes 100 gpm is applicable for the operator to identify the rupture since it is greater than what is required for the actuation.	I agree that the 100 gpm is applicable for operator identification. Maybe the term "safeguards actuation" should be changed to "CMT actuation on Low-2 PZR water level" since some of the "Safeguards Actuations" can be caused by other than a SGTR.	Either is sufficient as the threshold states Safeguards actuation and the rupture definition states safety injection.	No action required.
RCS Loss 1.B Basis para 1	A70. You need ADS valves to function if you are using once through cooling for PXS to operate. PRHR will function without and ADS actuation.	Remove the second sentence which says “Opening of ADS valves is required for the PXS to function”.	Yes	None	Remove the second sentence which says “Opening of ADS valves is required for the PXS to function”.
Att. 3 Figure 2	**New Change #23 Proposed by VCS and Vogtle for WEC Review:** ATTACHMENT 3 Figures, Figure 2 – Use Baseline 7 information: 520 gpm should be 400 gpm, 26% [42%] should be 21% in two places	Replaced CSFST with Baseline 7 versions.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Replaced CSFST with Baseline 7 versions.	No further changes are necessary.
Att. 3 Figure 5	**New Change #24 Proposed by VCS and Vogtle for WEC Review:** ATTACHMENT 3 Figures, Figure 5 – Use Baseline 7 information: 96.2% should be 94.4%.	Replaced CSFST with Baseline 7 versions.	Concur with this change and its inclusion in the November 2015 EAL Technical Bases Document.	Replaced CSFST with Baseline 7 versions.	No further changes are necessary.