Entergy

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10 CFR 50.90

GNRO-2018/00048

October 10, 2018

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

SUBJECT: Response to Request for Additional Information
License Amendment Request, Adoption of Emergency Action Level Scheme Pursuant to NEI 99-01, Revision 6

Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

REFERENCES:

1. Entergy Operations, Inc. (EOI) letter to U. S. Nuclear Regulatory Commission (NRC), "License Amendment Request, Adoption of Emergency Action Level Scheme Pursuant to NEI 99-01, Revision 6," dated April 27, 2018 (ML18117A514)

2. NRC email to EOI, "Final Request for Additional Information – Emergency Action Level Scheme Change (L-2018-LLA-0116)," dated August 30, 2018 (ML18250A304).

Dear Sir or Madam,

In Reference 1, Entergy Operations, Inc. (EOI) requested an amendment to Facility Operating License No. NPF-29 for Grand Gulf Nuclear Station, Unit 1 (GGNS). The proposed change revises the Emergency Plan for GGNS to adopt the revised Emergency Action Level (EAL) scheme described in Nuclear Energy Institute (NEI) 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," which has been endorsed by the Nuclear Regulatory Commission (NRC).

In Reference 2, the NRC requested additional information to complete its review of the proposed license amendment. In response, EOI is providing the requested information in the Enclosure and associated Attachments to this letter.

EOI has reviewed the information supporting a finding of no significant hazards consideration that was previously provided to the NRC in Reference 1. The information provided in this submittal does not affect the basis for concluding that the proposed license amendment does not involve a significant hazards consideration.

There are no regulatory commitments contained within this letter.

Should you have any questions or require additional information, please contact Douglas Neve at 601-437-2103.

I declare under penalty of perjury, the foregoing is true and correct. Executed on October 10, 2018

Sincerely,

Eric Larson

EL/dre

Enclosure: Response to Request for Additional Information
License Amendment Request, Adoption of Emergency Action Level Scheme Pursuant to NEI 99-01, Revision 6

Attachment 1: GGNS Calculation XC-QID17-17001, Radiological Effluent EAL Threshold Values (EP CALC-GGNS-1701)
Attachment 2: GGNS EAL Basis Document
Attachment 3: GGNS EAL Basis Document Markup

cc: with Attachment

U.S. Nuclear Regulatory Commission
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cc: without Attachment

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GNRO-2018/00048, ENCLOSURE

Response to Request for Additional Information
License Amendment Request
Adoption of Emergency Action Level Scheme Pursuant to NEI 99-01, Revision 6

Attachment 1
GGNS Calculation XC-QID17-17001, Radiological Effluent EAL Threshold Values (EP-CALC-GGNS-1701)

Attachment 2
GGNS EAL Basis Document

Attachment 3
GGNS EAL Basis Document Markup

Response to Request for Additional Information
License Amendment Request
Adoption of Emergency Action Level Scheme Pursuant to NEI 99-01, Revision 6

The format for the RAI responses below is as follows: The Request for Additional Information (RAI) is provided verbatim as received from the Nuclear Regulatory Commission (NRC). This is followed by the Grand Gulf Nuclear Station (GGNS) RAI response to the individual question.

RAI 1

It appears that some text is missing in Enclosure 1 of the License Amendment Request (LAR) between pages 6 and 7. Please provide the missing text, or clarify intent.

RAI 1 RESPONSE

GNRO 2018/00008 Enclosure 1 Pages 6 and 7 text is replaced as follows:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed changes to the GGNS EALs do not involve any physical changes to plant equipment or systems and do not alter the assumptions of any accident analyses. The proposed changes do not adversely affect accident initiators or precursors and do not alter design assumptions, plant configuration, or the manner in which the plant is operated and maintained. The proposed changes do not adversely affect the ability of structures, systems or components (SSCs) to perform intended safety functions in mitigating the consequences of an initiating event within the assumed acceptance limits.

   Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The changes do not challenge the integrity or performance of any safety-related systems. No plant equipment is installed or removed, and the changes do not alter the design, physical configuration, or method of operation of any plant SSC. Because EALs are not accident initiators and no physical changes are made to the plant, no new causal mechanisms are introduced.

Therefore, the changes do not create the possibility of a new or different kind of accident from an accident previously evaluated.

3. Do the proposed changes 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 do not impact operation of the plant and no accident analyses are affected by the proposed changes. The changes do not affect the Technical Specifications or the method of operating the plant. 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 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 changes do not involve a significant reduction in a margin of safety.

Based upon the reasoning presented above, Entergy concludes that the requested change involves no significant hazards consideration, as set forth in 10 CFR 50.92(c), *"Issuance of Amendment."*

4.4 Conclusions

In conclusion, based on the considerations discussed

above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

The proposed changes are applicable to emergency planning requirements involving the proposed adoption of the NRC-endorsed EAL guidance as described in NEI 99-01, Revision 6, and do not reduce the capability to meet the emergency planning standards of 10 CFR 50.47(b) and the requirements of 10 CFR 50, Appendix E. The proposed changes 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 individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed changes.

6.0 REFERENCES

1. NEI 99-01, Revision 5, *"Methodology for Development of Emergency Action Levels"* February 2008 (ML080450149)

2. NRC letter *"U.S. Nuclear Regulatory Commission Review and Endorsement of NEI 99-01, Revision 6, Dated November, 2012 (TAC No. D92368),"* March 28, 2013 (ML12346A463)

3. Entergy letter dated December 1, 2011, *"Proposed Emergency Action Levels Using NEI 99-01, Revision 5 Scheme,"* (ML12244A351) (TAC NO. ME7540)

4. NRC Safety Evaluation dated October 10, 2012, *"Emergency Action Level Scheme Upgrade Based on Nuclear Energy Institute 99-01, Revision 5 Entergy Operations, Inc. Grand Gulf Nuclear Station, Unit 1 Docket No. 50-416,"* (TAC Nos. ME7540)

5. NRC letter *"Callaway Plant, Unit 1 - Issuance of Amendment Re: Upgrade to Emergency Action Level Scheme (CAC No. MF4945),"* October 7, 2015 (ML15251A493)

6. NRC letter *"Fermi 2 - Issuance of Amendment to Revise the Emergency Action Level Scheme for the FERMI 2 Emergency Plan (TAC No. MF5048),"* September 29, 2015 (ML15233A084)

7. NRC letter *"South Texas Project, Units 1 and 2 - Re: Upgrade to Emergency Action Level Scheme (TAC Nos. MF4195 and MF4196),"* August 20, 2015 (ML14164A341)

RAI 2

Concerning Table A-1, "Effluent Monitor Classification Thresholds," please address the following:

a. For EAL AU1.1, explain how the proposed threshold values for five different effluent flow paths with different flow rates all have the same threshold value, since it appears that different flow rates would require different alarm setpoints.

b. For EALs AS1.1 and AG1.1, explain why the threshold values have significantly changed from the currently approved EAL scheme. This explanation should include the change from a single threshold value for all gaseous effluent flow paths to separate values for the gaseous effluent flow paths, as well as the reason for the magnitude of the change.

RAI 2 RESPONSE

a. The AU1.1 gaseous effluent EAL thresholds are based on the UFSAR Table 11.3-9 expected annual noble gas effluent release totals for normal coolant (no core damage). The same normal coolant activity source term fractions, based on the total estimated annual activity released, are applied to each release point. The thresholds in counts per minute (cpm) or microcuries per cubic centimeter (μCi/cc) are different for each release point due to the different effluent flow rates as documented in XC-Q1D17-17001, Radiological Effluent EAL Threshold Values (EP-CALC-GGNS-1701). The GGNS effluent parameter display system converts monitor readings into release rate

values of curies per second (Ci/sec), which will be equivalent for all release points for a given 10 CFR 20 annual exposure limit basis.

b. Current AG1.1 and AS1.1 EAL thresholds were developed on a site specific dose model (Dosecalc Program) that is currently in service, process reduction factors for which are built in to the various accident isotopic mix spectrums available to the user. Releases filtered exclusively through the Standby Gas Treatment System are subject to additional reduction of iodine in the Dosecalc Program, but other physical release pathways do not have individual process reduction factors built in.

   The proposed NEI 99-01 based AG1.1, AS1.1 and AA1.1 thresholds are developed using a GGNS site specific dose assessment model which is intended to be used when the proposed EAL scheme is implemented at GGNS. The dose model applies different process reduction factors for the fuel clad accident source term mix depending on the release pathway, which alters the composition of the activity released from each pathway. This method of process reduction is based on NUREG-1940.

   The threshold development methods between the current and the proposed EALs are not the same. Other differences Dosecalc and the intended model process reduction factors, such as X/Q and source term, account for further differences in magnitude.

RAI 3

For EALs AA2.3, AS2.1, and AG2.1, provide additional detail for the basis for rounding the threshold values upwards by approximately 10 inches, as this could result in an early and/or unnecessary declarations for a site area emergency or general emergency classification.

RAI 3 RESPONSE

The original submittal contained two typographical errors. Level 2 is 193 ft 2 1/8 inches NOT 192 ft 2 1/8 inches. Level 3 is 183 ft 2 1/8 inches NOT 182 ft 2 1/8 inches. Changes were made to the applicable pages of the Grand Gulf Nuclear Station EAL Basis Document to correct the typographical error and show that the instrument for spent fuel [pool level is not located in the Control Room. Enclosure Attachments 2 and 3 contain the Grand Gulf Nuclear Station EAL Basis Documents (clean and markup) to make these corrections.

RAI 4

For EALs CS1.3 and CG1.2, the Basis discussion for the proposed threshold value for the containment radiation monitors provides that the detectors are in a position to monitor the containment radiation environment above the refueling cavity elevation. Additional justification is provided in the EAL Comparison Matrix, which states that the threshold value "is indicative of likely core uncovery in the refueling zone." Provide additional detail supporting the threshold value for the proposed containment radiation monitors.

RAI 4 RESPONSE

Upon review of the calculations and the RAI question, Entergy has determined that using the proposed threshold values would have had the potential to cause excessive confusion to the operator making the EAL determination. Therefore, Entergy will apply currently in use NRC approved NEI 99-01 Revision 5 EAL values to the proposed NEI 99-01 Revision 6 scheme to ensure that the EAL are accurately classified. These threshold values were approved via NRC Safety Evaluation dated October 10, 2012, "*Emergency Action Level Scheme Upgrade Based on Nuclear Energy Institute 99-01, Revision 5 Entergy Operations, Inc. Grand Gulf Nuclear Station, Unit 1 Docket No. 50-416,*" (ML12244A351). The NEI 99-01 Revision 5 methodology for these EALs is the same as the NEI 99-01 Revision 6 methodology. Therefore, applying the NEI 99-01 Revision 5 EAL threshold values to NEI 99-01 Revision 6 does not represent a selective application of an EAL threshold from one EAL scheme to another.

RAI 5

The proposed EAL CU3.1 contains the condition "...due to the loss of decay heat removal capability," which is not consistent with NEI 99-01, Revision 6. This could result in potential misclassification for an event that causes reactor coolant system (RCS) temperature to rise above 200 degrees Fahrenheit (°F) when decay heat removal capability has not been lost. Provide additional detail for adding the condition, "...due to the loss of decay heat removal capability," to the EAL CU3.1 threshold value, or revise accordingly.

RAI 5 RESPONSE

Entergy agrees with the noted concern and, subsequently, the subject phrase has been removed from the CU3.1 EAL. The EAL will now state: "UNPLANNED rise in RCS temperature to > 200 °F."

In addition, the first paragraph of the GGNS basis for CU3.1 is changed from "Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200 °F) (ref. 3).

In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification is based on the concurrent loss of reactor vessel level indications per EAL CU3.2" to "Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200 °F) (ref. 3).

In the absence of reliable RCS temperature indication, classification is based on the concurrent loss of reactor vessel level indications per EAL CU3.2."

Enclosure Attachments 2 and 3 contain the Grand Gulf Nuclear Station EAL Basis Documents (clean and markup) to make these corrections.

RAI 6

The proposed EAL CA3.1 Basis discussion (1st paragraph) contains the condition, "...caused by the loss of decay heat removal capability," which is not consistent with NEI 99-01, Revision 6. This could result in potential misclassification for an event other than a loss decay heat removal capability that leads to an unplanned RCS pressure increase. Provide additional detail for the proposed Basis wording, or revise accordingly.

RAI 6 Response

Entergy agrees with the noted concern and, subsequently, the subject phrase has been removed from the 1st paragraph of the EAL CA3.1 Basis Discussion.

The paragraph will now state: "In the absence of reliable RCS temperature indication, classification should be based on the RCS pressure rise criteria when the RCS is intact in Mode 4 or based on time to boil data when in Mode 5 or the RCS is not intact in Mode 4."

Enclosure Attachments 2 and 3 contain the Grand Gulf Nuclear Station EAL Basis Documents (clean and markup) to make these corrections.

RAI 7

For EALs CU5.1 and SU7.1 explain how the INFORM Notification System (INFORM) can be used as a State and local agency communication method. This response should explain whether or not INFORM is independent of the provided telephone systems and if INFORM supports two way communications.

RAI 7 Response

Although INFORM is independent of the other telephone systems provided in the associated tables for these EALs, it does not support two-way communication and is, therefore, removed from Tables C-5 and S-4 in EALs CU5.1 and SU7.1.

Enclosure Attachments 2 and 3 contain the Grand Gulf Nuclear Station EAL Basis Documents (clean and markup) to make these corrections.

RAI 8

The proposed RCS barrier (RCB) 2 on the fission product matrix does not include the high pressure core spray (HPCS) system. The guidance states that the list of systems should also include high pressure coolant injection [high pressure core spray], since a rupture of the HPCS, if not isolated, could rapidly depressurize the reactor pressure vessel. Please justify not including the HPCS as a threshold value for the proposed RCB2.

RAI 8 Response

Review of the RAI determined that the original submittal needs to be revised to include the "HPCS line break unisolable from the reactor" coolant system EAL.

Additionally, clarification has been added to the basis section of "Even though the High Pressure Core Spray (HPCS) injects into the RCS, it is included in this EAL due to the potential for an inter-system loss of coolant back flowing from the discharge lines (via failed isolation valves and check valves) and out

through a break in the piping. A HPCS failure that does not result in back flow of RCS and out through a break should not be considered as meeting the EAL threshold."

Enclosure Attachments 2 and 3 contain the Grand Gulf Nuclear Station EAL Basis Documents (clean and markup) to make these corrections.

RAI 9

The proposed threshold values for fission product barrier degradation based on containment radiation monitors do not appear valid. Considering that the Fuel Clad Barrier (FCB) Loss threshold value should correspond to 2% to 5% clad damage, and the Containment Barrier (CNB) Potential Loss threshold value should be 20%, as provided by NEI 99-01, Revision 6, it would be reasonable for the radiation values to be different by a factor of 4 to 10. However, the value for the CNB Loss radiation monitor reading is 17.5 times higher than the FCB Loss radiation monitor reading.

Additionally, the NRC staff could not determine why the threshold value for the FCB3 Loss is significantly lower than that for River Bend Station (RBS), which is a lower powered Boiling Water Reactor Type 6 (BWR-6) that also has a Mark 3 Containment (400 R/hr for GGNS and 3000 R/hr for RBS), while the CNB threshold values were much closer (7000 R/hr for GGNS and 12000 R/hr for RBS). Please verify that the radiation monitor threshold values for a FCB Loss are based on a loss of the RCS with between approximately 2% and 5% clad damage and that the radiation monitor threshold values for a CNB Potential Loss are based on approximately 20% clad damage.

RAI 9 Response

In accordance with a follow-up clarification call held with NRC, the purpose of the RAI is not to compare the GGNS and RBS calculation results but to ensure the basis for both sites' EAL thresholds is correct.

The NEI 99-01, Revision 6, developer notes direct that "*the reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS radioactivity concentration equal to 300 uCi/gm dose equivalent I-131, into the primary containment atmosphere.*" The associated NEI basis states in part "*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.*"

The 2% - 5% fuel clad damage in the NEI basis for PWRs and BWRs originated in NUMARC/NESP-007 Rev 2 and could be confirmed using typical enrichment values and NUREG-1228 related source term/partitioning assumptions. Currently, most units operate with higher enrichment than what was common in the late 1980s and have been approved for power uprates. Additionally, newer source term and partitioning guidance such as NUREG-1940 are a factor. These combine such that typical reactor coolant concentrations to percent clad damage are approximately half of what was calculated using historical inputs and guidance. In addition, the use of a 300 uCi/gm dose equivalent I-131 (DEI) source term for FCB3 (NEI Fuel Clad Barrier Loss threshold 4.A) provides agreement within the EAL scheme with FCB4 (NEI Fuel Clad Barrier Loss threshold 1.A) which directly refers to a 300 uCi/gm DEI value. The calculated radiation monitor threshold value for Potential Loss of the Containment Barrier in CNB8 is based on 20% clad damage. Therefore, GGNS believes this use of 300 *uCi/gm dose equivalent I-131* more accurately meets the intent of NEI 99-01, Revision 6, developer notes and should be used if there is a disagreement between 300 *uCi/gm dose equivalent I-131*and the 2-5% *fuel clad damage.*

RAI 10

Explain why the Basis discussion (third paragraph) for a Potential Loss under CNB7, which states, *"cannot be maintained above,"* does not use the same wording as the threshold value, which states, *"cannot be restored and maintained within."* This difference in wording could result in an inaccurate or delayed assessment.

RAI 10 Response

Entergy agrees with the noted concern. The first sentence of the third paragraph of the basis for CNB7 is changed from ""the term "*cannot be restored and maintained above*" means the parameter value(s) is not able to be brought within the specified limit" to ""the term "*cannot be restored and maintained within*" means the parameter value(s) is not able to be brought within the specified limit."

Enclosure Attachments 2 and 3 contain the Grand Gulf Nuclear Station EAL Basis Documents (clean and markup) to make these corrections.

RAI 11

The proposed EAL HU4.2 - Table H-1, "Fire Areas," includes the Containment Building in all modes. This could result in an event declaration due to the spurious actuation of a single fire alarm.

The NRC staff could not determine if the Containment Fire Detection System, in combination with the Containment Ventilation System, supported the inclusion of the Containment Building as a fire area for EAL HU4.2. Provide justification that demonstrates why GGN includes the Containment Building in the Table H-1 for all modes, or modify accordingly.

RAI 11 Response

EAL HU4.1 addresses the condition where a fire is reported and verified in a listed Fire Area. This verification could be from a report in the field or because multiple fire detection device alarms are received. This EAL includes a table that lists fire areas of concern, including containment.

EAL HU4.2 addresses receipt of a single fire detector without a corresponding verification. Entergy proposes to make an exception in EAL HU4.2 to exclude the containment and drywell in Modes 1 and 2. Personnel safety concerns preclude entry into certain areas of containment and there are areas within containment where fire detectors are located that would be inaccessible during these modes due to elevated radiation levels. Industry experience has demonstrated that including containment in Modes 1 and 2 in EAL HU4.2 can lead to unusual event emergency classifications based on a single spurious fire alarm, requiring subsequent emergency retractions.

With regard to containment and drywell fire alarms, it can reasonably be expected that a fire that burns for 15 minutes would produce sufficient products of combustion to cause multiple fire detection devices to alarm. This is due to the products of combustion being transported to other areas inside the containment/drywell due to the forced flow ventilation system in operation. Receipt of a single fire alarm would likely be due to a spurious detector actuation.

There are four Containment Fan Cooler (CFC) units located in the Grand Gulf Nuclear Station Unit 1 (GGNS) containment building. Each CFC fan delivers approximately 75,000 CFM in normal mode and 37,800 CFM in accident mode. The four CFC units operate in accident mode when a Safety Injection Signal is present. Two or three of the four CFCs are operating in normal mode at any given time to cool the Containment in modes 1 or 2. The CFC units draw return air from the containment atmosphere and discharge into a common header which discharges to multiple areas inside containment. This

constant flow of air would draw any smoke/heat towards the cooling units past the installed detectors, thus initiating multiple detector alarms. Actuation of more than one detector is the most reliable indication of an actual fire because of high volumetric air flow throughout the containment building. Due to construction of the intermediate floors and multiple openings in the floors, it can be expected that smoke/heat would migrate throughout containment in a very short period and that 2 or more detectors would alarm. Basing emergency classifications on receiving more than one detector actuation is therefore the most reliable indication of a valid alarm and accurately meets the Initiating Condition of HU4, "FIRE potentially degrading the level of safety of the plant."

The drywell cooling system consists of recirculating fan-coil units and the associated dampers, ducting, and controls required to maintain the design drywell temperature and relative humidity. Each fan-coil unit consists of two full-capacity fans in parallel and two full-capacity cooling coils in series. Six fan-coil units with a capacity of 12,000 cfm per fan are provided to distribute cooling air effectively and with minimum ductwork. Normally, one fan and one coil of each fan-coil unit operate, and the other fan and coil are on standby. Additionally, the drywell cooling system incorporates two recirculation fans with a capacity of 6,000 cfm per fan and associated controls and ducting which transfer air from the upper elevation to the lower elevation. These fans alleviate heat stratification in the drywell. Normally, both fans operate simultaneously. Actuation of more than one smoke detector is the most reliable indication of an actual fire because of high volumetric air flow throughout the drywell. Due to construction of the intermediate floors and multiple openings in the floors, it can be expected that smoke/heat would migrate throughout the drywell in a very short period and that 2 or more detectors would alarm. Basing emergency classifications on receiving more than one detector actuation is therefore the most reliable indication of a valid alarm and accurately meets the Initiating Condition of HU4, "FIRE potentially degrading the level of safety of the plant."

With consideration to the above discussion, Note 11 is added to EAL HU4.2 as follows:

> "During Modes 1 and 2, HU4.2 is not applicable to a single fire alarm in the containment or drywell."

The following information is added to the Basis for HU4.2:

This EAL is not applicable for the containment or drywell in Modes 1 and 2. The air flow design and TS requirements for operation of Containment Fan Coolers and the drywell cooling system are such that multiple detectors would be expected to alarm for a fire in the containment or drywell. A fire in the containment or drywell in these modes would therefore be classified under EAL HU4.1.

Verification of a single containment or drywell fire alarm that is likely to be spurious does not warrant the potential elevated exposure risks and industrial safety risks associated with an emergency entry of containment or drywell in modes 1 and 2. Therefore, GGNS proposes to make EAL HU4.2 applicable to a single fire alarm in containment or drywell in Modes 3, 4and 5.

Consistent with the guidance in Regulatory Issue Summary (RIS) 2003-18, Supplement 2, Use of Nuclear Energy Institute (NEI) 99-01, "Methodology for Development of Emergency Action Levels," Revision 4, dated January 2003, it is reasonable to conclude that the changes proposed to EAL HU4.2 would be considered a deviation from the formally endorsed guidance of NEI 99-01 Revision 6.

The structure of the proposed deviation for HU4 IC/EAL is modelled after Seabrook Station's adoption of NEI 99-01 Revision 6 EALs containing a similar exception, which was approved by the NRC with Amendment 152 to the Seabrook Station Facility Operating License No. NPF-86 on February 10, 2017 (ADAMS Accession No. ML 16358A411). Proposed Emergency Preparedness Frequently Asked Question 2018-003 (ADAMS Accession No. ML18081A309) was also used as a basis for this deviation.

Based on the information above, Entergy considers the proposed revision to be an acceptable deviation from the generic NEI 99-01, Revision 6, guidance. This deviation is consistent with proposed Emergency Plan (EP) Frequently Asked Question (FAQ) 2018-03 (ML18081A309).

Enclosure Attachments 2 and 3 contain the Grand Gulf Nuclear Station EAL Basis Documents (clean and markup) to make these corrections.

RAI 12

The proposed EAL SU4.1 threshold value is based on the Offgas Pretreatment Radiation Monitor High-High Alarm, while the currently approved EAL scheme uses a table that includes various radiation monitor readings, which correspond to various flow rates.

The NRC staff could not determine if a value that was approximately equal to the technical specification allowable limits could be assessed with the proposed threshold value. Provide justification that supports using the Offgas Pretreatment Radiation Monitor High-High Alarm as a threshold value for SU4.1. This justification should include a discussion of the difference between the currently approved EAL scheme (EAL SU9.1) and the proposed EAL SU4.1

RAI 12 Response

During development of the NEI 99-01 Revision 6 EALs for GGNS it was determined that it was more effective and convenient to the Operator to use the Offgas Pretreatment Radiation Monitor High-High Alarm as a threshold value for SU4.1.

GGNS NEI 99-01 Revision 5 EALs used manual comparison of flow rates to radiation monitor readings to determine when Technical Specification (TS) 3.7.5 value of 380 millicuries per second release rate has been met or exceeded. Revision 6 EAL will use an existing Alarm (Offgas Pretreatment Radiation Monitor High-High) that is driven by a computer calculation to determine the Initiating Condition (IC) is met.

The purpose of the Offgas Pretreatment Radiation Monitor High-High Alarm is to come into alarm when the Technical Specification 3.7.5 value of 380 millicuries per second release rate has been met or exceeded. This is the initiating condition for SU4.1. By using this alarm, it allows the Operators to diagnose entry into SU4.1 quickly rather than having to review flowrates and radiation monitor readings. The use of flowrates and radiation monitor readings is a viable contingency action that will be maintained in the GGNS procedures and is added to the basis document information.

Enclosure Attachments 2 and 3 contain the Grand Gulf Nuclear Station EAL Basis Documents (clean and markup) to make these corrections.

GNRO-2018/00048, ENCLOSURE

ATTACHMENT 1

GGNS Calculation XC-QID17-17001, Radiological Effluent EAL Threshold Values

(EP CALC-GGNS-1701)

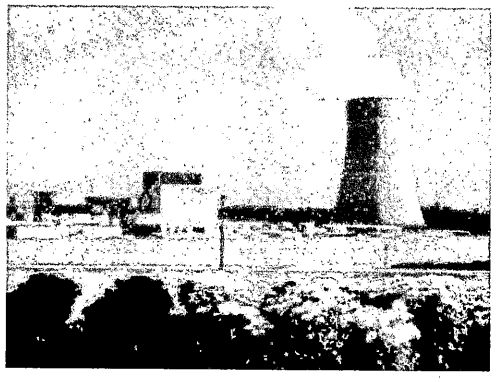
☐ ANO-1	☐ ANO-2	☒ GGNS	☐ IP-2	☐ IP-3	☐ PLP
☐ JAF	☐ PNPS	☐ RBS	☐ VY	☐ W3	
☐ NP-GGNS-3	☐ NP-RBS-3				

CALCULATION COVER PAGE	(1) **EC #** 73157 (2) **Page 1 of** 72

(3) Design Basis Calc. ☒ YES ☐ NO	(4) ☒ CALCULATION ☐ EC Markup
(5) **Calculation No: XC-Q1D17-17001**	(6) **Revision: 0**
(7) **Title: Radiological Effluent EAL Threshold Values (EP-CALC-GGNS-1701)**	(8) **Editorial** ☐ YES ☒ NO
(9) **System(s): D17**	(10) **Review Org (Department):**
(11) **Safety Class:** ☐ **Safety / Quality Related** ☒ **Augmented Quality Program** ☐ **Non-Safety Related**	(12) **Component/Equipment/Structure Type/Number:**
(13) **Document Type: J05.02**	
(14) **Keywords (Description/Topical Codes):** **EAL**	

REVIEWS

(15) Name/Signature/Date	(16) Name/Signature/Date	(17) Name/Signature/Date
Scott McCain, OSSI, Inc.	Robert W. Fuller Robert W. Fuller 2-26-18	Brandon Taylor 2/20/18
Responsible Engineer	☒ **Design Verifier** ☐ **Reviewer** ☐ Comments Attached	**Supervisor/Approval** ☐ Comments Attached



Grand Gulf Nuclear Station (GGNS)

Radiological Effluent EAL Threshold Values

EP-CALC-GGNS-1701
Revision 0

OSSI Author: Scott McCain

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ATTACHMENTS

1. PURPOSE

The Grand Gulf Nuclear Station (GGNS) Emergency Action Level (EAL) Technical Bases Manual contains background information, event declaration thresholds, bases and references for the EAL and Fission Product Barrier (FPB) values used to implement the Nuclear Energy Institute (NEI) 99-01 Revision 6 EAL guidance. This calculation document provides additional technical detail specific to the derivation of the gaseous and liquid radiological effluent EAL values developed in accordance with the guidance in NEI 99-01 Revision 6.

Documentation of the assumptions, calculations and results are provided for the GGNS Ax1 series EAL effluent monitor values associated the NEI 99-01 Revision 6 EALs listed below.

- NEI EAL AU1.1 (gaseous and liquid)
- NEI EAL AA1.1 (gaseous and liquid)
- NEI EAL AS1.1 (gaseous)
- NEI EAL AG1.1 (gaseous)

2. DEVELOPMENT METHODOLOGY AND BASES

2.1. Threshold Limits

2.1.1. AU1.1 Liquid Threshold Limits

Guidance Criteria

The AU1 Initiating Condition (IC) addresses a release of gaseous or liquid radioactivity greater than 2 times the Offsite Dose Calculation Manual (ODCM) limits for 60 minutes or longer.

GGNS Bases

ODCM Section 6.11.1 states that the limits for the concentration of radioactive liquid effluents released from the site to unrestricted areas are as follows:

- Ten (10) times the concentrations specified in 10 CFR Part 20, Appendix B, Table 2, Column 2 for radionuclides other than dissolved or entrained noble gases
- 2.0E-04 µCi/ml total activity for dissolved or entrained noble gases

The site specific AU1.1 liquid effluent EAL threshold values will equate to 2 times the ODCM limit. Refer to Section 4.1 for the threshold calculation related to this limit.

2.1.2. AU1.1 Gaseous Threshold Limits

Guidance Criteria

The AU1 Initiating Condition (IC) addresses a release of gaseous or liquid radioactivity greater than 2 times the Offsite Dose Calculation Manual (ODCM) limits for 60 minutes or longer.

GGNS Bases

ODCM Section 6.11.4 states that the limits for the radioactive gaseous effluents released from the site at or beyond the site boundary are as follows:

- Less than or equal to 500 mrem/yr to the total body (Noble Gases)
- Less than or equal to 3000 mrem/yr to the skin (Noble Gases)
- Less than or equal to 1500 mrem/yr to any organ (I-131, I-133, tritium and radioactive materials in particulate form with half-lives > 8 days)

  ODCM gaseous setpoint calculations are based on the noble gas limits. Organ dose includes inhalation, ingestion and deposition pathways and are applied in unrestricted area site boundary gaseous effluent dose calculations used in the Annual Radioactive Effluent Release Report. Ingestion pathway bases are not compatible or directly comparable with short term event considerations, and are not a significant contribution to the total dose (total body or skin dose limits from noble gas are the major exposure pathway). Thus, the organ dose limit is not applicable for EAL threshold determination.

The site specific AU1.1 gaseous effluent EAL threshold values will equate to 2 times the ODCM limit for the lesser of the total body or skin exposure pathways. Refer to Section 4.2 for the threshold calculation related to this limit.

2.1.3. AA1.1 Liquid Threshold Limits

Guidance Criteria

The AA1 Initiating Condition (IC) addresses a release of radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.

This is based on values at 1% of the EPA Protective Action Guides (PAGs).

Per NEI 99-01, the effluent monitor readings should correspond to the above dose limits at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.

GGNS Bases

The liquid effluent limits are based on the water concentration values given in 10 CFR 20 Appendix B Table 2 Column 2 (see Section 2.1.1 above). The 10 CFR 20 values are equivalent to the radionuclide concentrations which, if ingested continuously over the course of a year, would produce a total effective dose equivalent of 0.05 rem (50 mrem). The EPA PAGs are based on a TEDE dose from immersion, inhalation and deposition.

The 10 CFR 20 limits and the EPA limits do not represent the same type of exposure and thus cannot be compared on a one to one basis.

Thus, the site specific EALs will not contain an AA1.1 liquid effluent monitor threshold value that equates to 1% of the EPA PAG. However, EALs AA1.3 (liquid effluent sample analysis) and AA1.4 (field survey results) will remain applicable for liquid effluent releases that exceed their respective thresholds. Since EALs AA1.3 and AA1.4 are not associated with a calculation no further reference is made to those EALs.

2.1.4. AA1.1 Gaseous Threshold Limits

Guidance Criteria

The AA1 IC addresses a release of radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.

Per NEI 99-01, the effluent monitor readings are based on values at 1% of the EPA Protective Action Guides (PAGs) at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.

GGNS Bases

The gaseous effluent limits for AA1.1 are based on values that equate to an offsite dose greater than 10 mrem TEDE or 50 mrem CDE thyroid, which are 1% of the EPA PAGs.

2.1.5. AS1.1 Gaseous Threshold Limits

Guidance Criteria

The AS1 IC addresses a release of radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.

This is based on values at 10% of the EPA Protective Action Guides (PAGs) at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.

GGNS Bases

The gaseous effluent limits for AS1.1 are based on values that equate to an offsite dose greater than 100 mrem TEDE or 500 mrem CDE thyroid, which are 10% of the EPA PAGs.

2.1.6. AG1.1 Gaseous Threshold Limits

Guidance Criteria

The AG1 IC addresses a release of radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.

This is based on values at 100% of the EPA Protective Action Guides (PAGs) at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.

GGNS Bases

The gaseous effluent limits for AG1.1 are based on values that equate to an offsite dose greater than 1,000 mrem TEDE or 5,000 mrem CDE thyroid, which are 100% of the EPA PAGs.

2.2. Effluent Release Points

Note – All effluent release points assume a background reading of zero to conservatively account for all modes of operation applicable to the EALs.

2.2.1. Liquid Release Points

Guidance Criteria

Per NEI 99-01, the AU1 IC addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways (NEI AU1 EAL #1) and planned batch releases from non-continuous release pathways (NEI AU1 EAL #2).

Per NEI 99-01, the AA1 IC includes events or conditions involving a radiological release, whether gaseous or liquid, monitored or un-monitored. 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.

The "site-specific monitor list and threshold values" should be determined with consideration of the selection of the appropriate installed gaseous and liquid effluent monitors.

GGNS Bases

The single liquid effluent monitor at GGNS (ODCM Figure 1.3-1 and Table 6.3.9-1) is the Liquid Radwaste Effluent Line.

2.2.2. Gaseous Release Points

Guidance Criteria

Per NEI 99-01, the AU1 IC addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways (NEI AU1 EAL #1) and planned batch releases from non-continuous release pathways (NEI AU1 EAL #2).

Per NEI 99-01, the AA1 IC includes events or conditions involving a radiological release, whether gaseous or liquid, monitored or un-monitored. 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.

Per NEI 99-01, the AS1 and AG1 ICs address monitored and un-monitored releases of gaseous radioactivity. 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.

The "site-specific monitor list and threshold values" should include the effluent monitors described in emergency plan and emergency dose assessment procedures.

GGNS Bases

There are seven GGNs release points to the environment. The Offgas Pre- and Post-Treatment Monitors are upstream of the Radwaste Building Vent monitor and thus are not used as separate EAL gaseous effluent threshold values. The seven gaseous effluent monitors at GGNS (ODCM Figure 2.5-1 and Table 6.3.10-1) are as follows:

- SBGT Exhaust A
- SBGT Exhaust B
- Containment Vent
- Fuel Handing Area Vent
- Radwaste Building Vent
- Turbine Building Vent
- Turbine Building occasional release point

In modes 1, 2 and 3, the south-east most smoke hatch of the turbine building may be used as an occasional release point provided that the proper portable monitoring equipment is used. During Modes 4 & 5 up to four roof hatches may be used and release rates estimated based on calculated flow rates and measured activity. Since releases from this point are infrequent, the temporary nature of the equipment, and various configurations involved, equipment substitution may occur with approval of Radiation Protection Manager and Chemistry Manager (UFSAR Section 9.4.4.2, ODCM Figure 2.5-1, 06-HP-1DI7-V-0001 Section 5).

Based upon the variations in portable equipment and set-up, and infrequent occurrence, the Turbine Building occasional release point pathway does not meet the NEI 99-01 criteria as a normally occurring continuous release or a planned batch release point and thus is not used as an effluent monitor EAL threshold.

2.3. Source Term

2.3.1. AU1.1 Liquid Source Term

Guidance Criteria

NEI 99-01 does not provide specific guidance for AU1 liquid source term assumptions.

GGNS Bases

The AU1.1 liquid effluent EAL threshold is based upon measured gamma emitter activity from discharge permits 2017007, 2017008 and 2017009. The total activity of each isotope released (µCi/ml) is normalized to a representative fraction which is then adjusted to the ODCM limit (refer to Section 2.1.1).

	Permit 2017007 (µCi/ml)	Permit 2017008 (µCi/ml)	Permit 2017009 (µCi/ml)	Permit Totals (µCi/ml)	Isotope (Fraction)
Na-24		8.75E-08		8.75E-08	1.4E-02
Mn-54	9.53E-08	3.21E-08	3.35E-07	4.62E-07	7.3E-02
Co-60	6.17E-07	4.23E-07	1.27E-06	2.31E-06	3.6E-01
Zn-65	3.04E-07	4.01E-07	1.32E-06	2.03E-06	3.2E-01
Ag-110m	2.51E-07	4.58E-08		2.97E-07	4.7E-02
Sb-125			1.71E-07	1.71E-07	2.7E-02
Cs-134			3.92E-07	3.92E-07	6.2E-02
Cs-137			6.18E-07	6.18E-07	9.7E-02
Totals	1.27E-06	9.89E-07	4.11E-06	6.36E-06	1.0E+00

2.3.2. AU1.1 Gaseous Source Term

Guidance Criteria

NEI 99-01 does not provide specific guidance for AU1 gaseous source term assumptions.

GGNS Bases

The AU1.1 gaseous effluent EAL threshold is based upon UFSAR Table 11.3-9, Expected Annual Release of Gaseous Effluents, Noble Gas (activity and fractions) for normal coolant (no core damage).

	Release Rate (Ci/y)	Noble Gas Fraction
Ar-41	7.2E+01	9.0E-03
Kr-83m	0.0E+00	0.0E+00
Kr-85m	8.8E+01	1.1E-02
Kr-85	3.9E+02	4.9E-02
Kr-87	6.3E+01	7.8E-03
Kr-88	9.8E+01	1.2E-02
Kr-89	6.1E+02	7.6E-02
Kr-90	0.0E+00	0.0E+00
Xe-131m	2.0E+01	2.5E-03
Xe-133m	0.0E+00	0.0E+00
Xe-133	2.2E+03	2.7E-01
Xe-135m	9.9E+02	1.2E-01

	Release Rate (Ci/y)	Noble Gas Fraction
Xe-135	1.2E+03	1.5E-01
Xe-137	1.3E+03	1.6E-01
Xe-138	1.0E+03	1.2E-01
Totals	**8.0E+03**	**1.0E+00**

2.3.3. AA1.1, AS1.1 and AG1.1 Gaseous Source Terms

Guidance Criteria

NEI 99-01 specifies that the calculation of monitor readings will require use of an assumed release isotopic mix; the selected mix should be the same for ICs AA1, AS1 and AG1.

GGNS Bases

The AA1.1, AS1.1 and AG1.1 gaseous EAL thresholds are based upon the GGNS URI dose model results using input assumptions applicable to the event, pathway and particular monitor.

The source term used in the URI dose model is taken from NUREG-1940 Table 1.1 (URI Requirements Specification Appendix A Section A.2).

The process reductions used in the URI dose model are taken from NUREG-1228 and NUREG-1465 (URI Requirements Specification Appendix A Sections A.4 and A.5).

Note – HUT is hold-up time.

Other than the fuel handling accident scenario, the release paths selected were chosen to represent a LOCA type event with fuel clad damage and process reductions for applicable suppression pool and bypass release pathways.

URI input assumptions for the gaseous release points are as follows:

RCS	Containment HUT <2 hrs Sprays Off	Pool Subcooled	Filters Working	Aux Bldg HUT <2 hrs	SBGT Vent	Env

Release path 'K' selected to model a LOCA type event with fuel clad damage and suppression pool reduction.

RCS	Containment HUT <2 hrs Sprays Off	Pool Subcooled	Filters Working	20" Cont Vent	Env

Release path 'I' selected to model a LOCA type event with fuel clad damage and suppression pool reduction.

RCS	Aux Bldg HUT <2 hrs	RW Bldg HUT <2 hrs	Filters Working	Radwaste Vent	Env

Release path 'R' selected to model a LOCA type event with fuel clad damage and bypass reduction.

RCS	**Turbine Building** HUT 2-24 hrs	**Filters** Working	**Turbine Building Vent**	**Env**

Release path 'C' selected to model a LOCA type event with fuel clad damage and bypass reduction.

Spent Fuel Under Water	**Filters** Working	**Aux Building Vent** HUT 2-24 hrs	**Env**

Release path 'U' selected to model a spent fuel pool accident.

For RCS initiated accidents, a 1 hour time after shutdown (TAS) is used for the source decay period as it is long enough for plant conditions to deteriorate for core damage to occur and a significant release to start.

For the spent fuel accident, the new fuel age option is used with a default of 80 hours for time after shutdown (TAS).

2.4. Effluent Flow

2.4.1. Effluent Liquid Discharge Flow

Guidance Criteria

NEI 99-01 does not provide specific guidance for effluent liquid flow assumptions.

GGNS Bases

Per UFSAR 11.2.1.1, the design objective of the liquid Radwaste system is to collect, process, recycle or dispose of potentially radioactive wastes produced during the operation of the plant. These wastes are grouped as floor drains, equipment drains, and chemical waste. Liquid waste collected in the equipment drain processing system is normally transferred to the condensate storage tank after processing. Chemical wastes are sent to the floor drain collector tank for further processing or returned to the condensate storage tank. Liquid waste collected in the floor drain processing system is normally treated and released to the environment but may be recycled to the condensate storage tank. Any of these treated wastes may be discharged to the environment, providing proper dilution at the discharge basin is maintained; however, normally only processed waste from the floor drain and chemical waste subsystems will be discharged to the environment.

A representative maximum discharge flow rate of 100 gpm and a minimum dilution flow rate of 3,500 gpm is used as the input for purposes of the EAL calculations (liquid release permit reports 2017007, 2017008 and 2017009). Refer to Sections 3.2.3 and 3.2.4 for the input values related to this parameter.

2.4.2. Effluent Gaseous Vent Flow

Guidance Criteria

NEI 99-01 does not provide specific guidance for effluent gaseous vent flow assumptions.

GGNS Bases

Vent flow values for AU1.1 are taken from 08-S-03-22.

Vent flow values for AA1.1, AS1.1 and AG1.1 are taken from System Flow Diagrams.

Refer to Sections 3.3.3 and 3.3.4 for the input values related to the vent flow parameter.

2.5. Release Duration

Guidance Criteria

Per NEI 99-01, the effluent monitor readings for AS1.1 and AG1.1 gaseous EAL threshold values should correspond to a dose at the "site-specific dose receptor point" (consistent with the calculation methodology employed) for one hour of exposure.

GGNS Bases

The effluent monitor readings for AA1.1, AS1.1 and AG1.1 gaseous EAL threshold values are calculated for a release duration of one hour.

2.6. Meteorology

Guidance Criteria

The effluent monitor readings should correspond to the applicable dose limit at the "site-specific dose receptor point." The "site-specific dose receptor point" is the distance(s) and/or locations used by the licensee to distinguish between on-site and offsite doses. The selected distance(s) and/or locations should reflect the content of the emergency plan, and the procedural methodology used to determine offsite doses and protective action recommendations. This is typically the boundary of the Owner Controlled Area.

Monitor readings will be calculated using a set of assumed meteorological data or atmospheric dispersion factors; the data or factors selected for use should be the same for ICs AA1, AS1 and AG1.

GGNS Bases

The site specific meteorology used for the EAL calculation inputs are based upon the UFSAR and ODCM as documented below.

2.6.1. ODCM Gaseous Dispersion Factor (ODCM Table 2.2-3a)

GGNS uses a ground level release model for all effluent release points (ODCM Section 2.3).

The ODCM highest historical annual average X/Q at the site boundary of 4.1E-06 sec/m^3 is based on a wind direction from the Northeast into SW sector 'L' (045°).

2.6.2. Stability Class

UFSAR Section 2.3.2.1.1.1 and Tables 2.3-130A through 130G document the predominant stability class as 'D'. Thus, a stability class of "D" is used as the URI input for purposes of the EAL calculations.

2.6.3. Wind Speed

UFSAR Section 2.3.2.1.1.1, Figure 2.3-2, Figure 2.3-3 and Table 2.3-34 document the annual average wind speed of 4.4 mph. Thus, a wind speed of 4.4 mph is used as the URI input for purposes of the EAL calculations.

2.6.4. Wind Direction

UFSAR Section 2.3.2.1.1.1, Figure 2.3-2, Figure 2.3-3, Table 2.3-34 and Table 2.3-130H document the predominant wind direction for GGNS as to the East sector. Thus, a wind direction input of 270° (winds from) is used as the URI input for purposes of the EAL calculations.

2.6.5. Other Parameters

No precipitation is assumed to occur for the duration of the release and plume transport across the Emergency Planning Zone.

3. DESIGN INPUTS

3.1. General Constants and Conversion Factors

3.1.1. 472 cc/sec per cfm

3.2. Liquid Effluent

3.2.1. Liquid Effluent Monitor Ranges (UFSAR Table 11.5-1)

1) Liquid Radwaste Effluent Line ..... 1E+1 to 1E+6 cpm

3.2.2. Liquid Monitor Response Factor – *MRF* (GIN-2001/01196)

1) Liquid Radwaste Effluent Line ..... 3E+8 cpm per μCi/cc

3.2.3. Liquid Effluent Discharge Source Flow (*f*)

1) Maximum discharge flow (Discharge Permits) ..... 100 gpm

3.2.4. Liquid Effluent Dilution Flow (*F*)

1) Minimum expected dilution flow (Discharge Permits) ..... 3,500 gpm

3.2.5. Liquid Effluent Source Term Limit (EC_i)

	10 CFR 20 Liquid Limit (μCi/ml)	ODCM Liquid Limit (μCi/ml)
Na-24	5.0E-05	5.0E-04
Mn-54	3.0E-05	3.0E-04
Co-60	3.0E-06	3.0E-05
Zn-65	5.0E-06	5.0E-05
Ag-110m	6.0E-06	6.0E-05
Sb-125	3.0E-05	3.0E-04
Cs-134	9.0E-07	9.0E-06
Cs-137	1.0E-06	1.0E-05

3.3. Gaseous Effluent

3.3.1. Monitor Efficiency Factor – Eff

1) GE Monitors – Low Range (08-S-03-22 Section 6.2.2)

- Containment Vent
- Fuel Handing Area Vent
- Radwaste Building Vent
- Turbine Building Vent

7.69E-8 μCi/cc per cpm

2) SPING Monitors – Channel 5 Low Range (08-S-03-22 Section 6.7.5)

- SBGT Exhaust A
- Containment Vent
- Fuel Handing Area Vent
- Radwaste Building Vent
- Turbine Building Vent

3.54E-8 µCi/cc per cpm

3) SPING Monitors – Channel 7 Mid-Range (08-S-03-22 Section 6.7.7)

- SBGT Exhaust A
- Containment Vent
- Radwaste Building Vent
- Turbine Building Vent
- Fuel Handing Area Vent

1.15E-4 µCi/cc per cpm

4) AXM Monitors – Channel 4 Mid-Range (08-S-03-22 Section 6.6.4)

- SBGT Exhaust A
- Containment Vent
- Fuel Handing Area Vent
- Radwaste Building Vent
- Turbine Building Vent

3.04E-6 µCi/cc per cpm

5) AXM Monitors – Channel 3 High Range (08-S-03-22 Section 6.6.3)

- SBGT Exhaust A
- Containment Vent
- Radwaste Building Vent
- Turbine Building Vent
- Fuel Handing Area Vent

1.69E-3 µCi/cc per cpm

6) Canberra Monitors – Normal Range (08-S-03-22)

- SBGT Exhaust B ..... 3.40E-8 µCi/cc per cpm

7) Canberra Monitors – High Range (08-S-03-22)

- SBGT Exhaust B ..... 9.90E-6 µCi/cc per cpm

3.3.2. Gaseous Effluent Monitor Ranges

1) GE Monitors – Low Range (UFSAR Table 11.5-1)

• Containment Vent • Radwaste Building Vent • Turbine Building Vent • Fuel Handing Area Vent	1E+1 to 1E+6 cpm 7.69E-7 to 7.69E-2 µCi/cc

2) SPING Monitors – Channel 5 Low Range (UFSAR Table 11.5-1)

Note – Ranges in cpm were derived from the UFSAR values given in µCi/cc and the monitor efficiency in section 3.3.1.

• SBGT Exhaust A • Containment Vent • Radwaste Building Vent • Turbine Building Vent • Fuel Handing Area Vent	2.8E+0 to 1.7E+6 cpm 1E-7 to 6E-2 µCi/cc

3) SPING Monitors – Channel 7 Mid-Range (UFSAR Table 11.5-1)

Note – Ranges in cpm were derived from the UFSAR values given in µCi/cc and the monitor efficiency in section 3.3.1.

• SBGT Exhaust A • Containment Vent • Radwaste Building Vent • Turbine Building Vent • Fuel Handing Area Vent	1.7E+2 to 3.5E+6 cpm 2E-2 to 4E+2 µCi/cc

4) AXM Monitors – Channel 4 Mid-Range (UFSAR Table 11.5-1)

Note – Ranges in cpm were derived from the UFSAR values given in µCi/cc and the monitor efficiency in section 3.3.1.

• SBGT Exhaust A • Containment Vent • Radwaste Building Vent • Turbine Building Vent • Fuel Handing Area Vent	3.3E+1 to 3.3E+6 cpm 1E-4 to 1E+1 µCi/cc

5) AXM Monitors – Channel 3 High Range (UFSAR Table 11.5-1)

Note – Ranges in cpm were derived from the UFSAR values given in µCi/cc and the monitor efficiency in section 3.3.1.

• SBGT Exhaust A	
• Containment Vent	5.9E+3 to 5.9E+7 cpm
• Radwaste Building Vent	1E+1 to 1E+5 µCi/cc
• Turbine Building Vent	
• Fuel Handing Area Vent	

6) Canberra Monitors – Normal Range (EC 57863 pages 5148-5162)

- SBGT Exhaust B ..... 1E+0 to 1E+9 cpm

7) Canberra Monitors – High Range (EC 57863 pages 5148-5162)

- SBGT Exhaust B ..... 1E+0 to 5E+9 cpm

3.3.3. AU1.1 Gaseous Effluent Source Flow – *f*

1) SBGT Exhaust A and B (08-S-03-22) ..... 4,300 cfm
2) Containment Vent (08-S-03-22) ..... 6,000 cfm
3) Fuel Handing Area Vent (08-S-03-22) ..... 31,000 cfm
4) Radwaste Building Vent (08-S-03-22) ..... 48,000 cfm
5) Turbine Building Vent (08-S-03-22) ..... 15,000 cfm

3.3.4. AA1.1, AS1.1 and AG1.1 Gaseous Effluent Source Flow – *f*

1) SBGT Exhaust A and B (SFD1102) ..... 4,000 cfm
2) Containment Vent (SFD1100) ..... 6,000 cfm
3) Fuel Handing Area Vent (SFD1104A) ..... 24,720 cfm
4) Radwaste Building Vent (SFD0047) ..... 52,495 cfm
5) Turbine Building Vent (SFD1105A) ..... 5,000 cfm

3.3.5. AU1.1 X/Q Dispersion Factor (ODCM Table 2.2.3a)

1) All Ground Level Release Points ..... 4.1E-6 sec/m^3

3.3.6. ODCM Dose Factors (ODCM Table 2.1-1)

	γ – Body Ki (mrem/yr per µCi/m³)	β – Skin Li (mrem/yr per µCi/m³)	γ – Air Mi (mrad/yr per µCi/m³)
Ar-41	8.84E+03	2.69E+03	9.30E+03
Kr-83m	7.56E-02	0.00E+00	1.93E+01
Kr-85m	1.17E+03	1.46E+03	1.23E+03
Kr-85	1.61E+01	1.34E+03	1.72E+01
Kr-87	5.92E+03	9.73E+03	6.17E+03
Kr-88	1.47E+04	2.37E+03	1.52E+04
Kr-89	1.66E+04	1.01E+04	1.73E+04
Kr-90	1.56E+04	7.29E+03	1.63E+04
Xe-131m	9.15E+01	4.76E+02	1.56E+02
Xe-133m	2.51E+02	9.94E+02	3.27E+02
Xe-133	2.94E+02	3.06E+02	3.53E+02
Xe-135m	3.12E+03	7.11E+02	3.36E+03
Xe-135	1.81E+03	1.86E+03	1.92E+03
Xe-137	1.42E+03	1.22E+04	1.51E+03
Xe-138	8.83E+03	4.13E+03	9.21E+03

4. CALCULATIONS

4.1. AU1.1 Liquid Release

4.1.1. Liquid Effluent Monitor ODCM Limit (derived from ODCM Section 1.1.1)

Per ODCM 1.1.1. Step 2, SF is a normally applied administrative safety factor which causes the calculated Dilution Factor to be two (2) times larger than the dilution factor required for compliance with 10x 10CFR20 limits.

The normalized activity mix multiplied by their 10x 10CFR20 limit by definition yield a sum concentration activity equivalent to the 10x 10CFR20 limit, or a compliance dilution factor of 1.

$$compliance\ dilution\ factor = \left[\sum_g \frac{C_g}{EC_g}\right]$$

$$SF = \left[\sum_g \frac{C_g}{EC_g}\right] \times 2$$

Where:	
SF	administrative safety factor
C_g	normalized gamma emitter effluent concentration value (fraction)
EC_g	ODCM limit (µCi/ml)

$$SP = \frac{[\sum_g (C_g \times EC_g)] \times \left(\frac{F+f}{SF}\right)}{f} \times MRF$$

Where:	
SP	radiation monitor setpoint equivalent to the ODCM limit (cpm)
C_g	normalized gamma emitter effluent concentration value (fraction)
EC_g	ODCM limit (µCi/ml)
F	minimum setpoint dilution flow (gpm) = 0.9 x minimum expected dilution flow
SF	administrative safety factor – see above
f	maximum setpoint discharge flow (gpm) = 0.9 x maximum discharge flow
MRF	Monitor Response Factor (cpm/µCi/ml)

4.1.2. AU1.1 Liquid Release EAL Threshold

AU1.1 liquid is two times (2x) the calculated ODCM limit setpoint.

See Attachment 1 for the spreadsheet calculations that develop the AU1.1 liquid effluent EAL threshold values for each applicable monitor.

4.2. AU1.1 Gaseous Release

4.2.1. Gaseous Release at the ODCM Limit

$$SP_{whole\ body} = \frac{\left(\frac{500}{472 \times f \times {}^{X}/_{Q_{RP}} \times \sum(Q_i \times K_i)}\right)}{Eff}$$

$$SP_{skin} = \frac{\left(\frac{3000}{472 \times f \times {}^{X}/_{Q_{RP}} \times \sum(Q_i \times (L_i + 1.1M_i))}\right)}{Eff}$$

Where:	
SP	radiation monitor setpoint equivalent to the ODCM limit (cpm)
500/3000	Dose Limit – 500 whole body or 3000 skin (mrem/yr)
472	conversion factor (cc/ft^3 per sec/min)
f	vent flow (cfm)
X/Q_{RP}	highest land annual average dispersion factor for the release point (sec/m^3)
Q_i	activity released (fraction – unit less)
K_i	whole body dose correction factor (mrem/yr per µCi/m^3)
$L_i + 1.1M_i$	skin dose factor (mrem/yr per µCi/m^3)
Eff	detector efficiency (µCi/cc per cpm)

4.2.2. AU1.1 Gaseous Release EAL Threshold

AU1.1 gaseous is two times (2x) the lesser of the calculated whole body or skin ODCM limit setpoint.

See Attachment 2 for the spreadsheet calculations that develop the AU1.1 gaseous effluent EAL threshold values for each applicable monitor.

4.3. AA1.1, AS1.1 and AG1.1 Gaseous Release

4.3.1. Canberra Monitors

The AA1.1, AS1.1 and AG1.1 gaseous release EAL thresholds for the Canberra monitors are derived from the SBGT A results as follows:

$$SBGT\ B\ (cpm) = \frac{RR\ (\mathrm{Ci/sec}) \times 1E6\ (\mu\mathrm{Ci/Ci})}{Flow\ (cfm) \times 472\ (cc/\sec per\ cfm) \times Eff(\mu\mathrm{Ci/cc\ per\ cpm})}$$

Refer to Attachment 3 for the results of the SBGT B gaseous effluent EAL threshold calculations.

4.3.2. SPING and AXM Monitors

The AA1.1, AS1.1 and AG1.1 gaseous release EAL thresholds for the SPING and AXM monitors are developed using the site specific URI dose assessment model with the inputs described in Section 2 above.

Refer to Attachment 4 for the results of the URI gaseous effluent EAL threshold calculations.

5. CONCLUSIONS

5.1. Effluent Monitor Reading Results in CPM (All Calculated Values Within Range)

	Release Point	Monitor	GE (cpm)	SAE (cpm)	Alert (cpm)	UE (cpm)
Gaseous	SBGT A	SPING 5	N/A	N/A	N/A	9.32E+5
		SPING 7	3.73E+6	3.73E+5	3.74E+4	2.87E+2
		AXM 4	N/A	1.40E+7	1.42E+6	1.08E+4
		AXM 3	2.54E+5	2.54E+4	2.55E+3	N/A
	SBGT B	Canberra normal	N/A	N/A	1.17E+8	9.70E+5
		Canberra high	4.03E+7	4.03E+6	4.03E+5	3.33E+3
	Containment Vent	General Electric	N/A	N/A	N/A	3.07E+5
		SPING 5	N/A	N/A	N/A	6.68E+5
		SPING 7	1.98E+6	1.98E+5	1.98E+4	2.05E+2
		AXM 4	7.48E+7	7.48E+6	7.48E+5	7.77E+3
		AXM 3	1.35E+5	1.35E+4	1.35E+3	N/A
	Radwaste Building Vent	General Electric	N/A	N/A	N/A	3.84E+4
		SPING 5	N/A	N/A	N/A	8.34E+4
		SPING 7	1.72E+4	1.72E+3	1.72E+2	N/A
		AXM 4	6.50E+5	6.50E+4	6.50E+3	9.72E+2
		AXM 3	1.17E+3	1.17E+2	1.17E+1	N/A
	Turbine Building Vent	General Electric	N/A	N/A	N/A	1.23E+5
		SPING 5	N/A	N/A	N/A	2.67E+5
		SPING 7	4.26E+4	4.26E+3	4.26E+2	N/A
		AXM 4	1.61E+6	1.61E+5	1.61E+4	3.11E+3
		AXM 3	2.90E+3	2.90E+2	2.90E+1	N/A
	Fuel Handling (Aux Bldg) Vent	General Electric	N/A	N/A	N/A	5.95E+4
		SPING 5	N/A	N/A	N/A	1.29E+5
		SPING 7	6.43E+6	6.43E+5	6.44E+4	N/A
		AXM 4	N/A	2.43E+7	2.44E+6	1.50E+3
		AXM 3	4.38E+5	4.38E+4	4.39E+3	N/A
Liquid	Radwaste	Liquid Effluent Monitor	N/A	N/A	N/A	7.33E+5

5.2. Effluent Monitor Reading Results in CPM (For Applicable EAL Thresholds)

	Release Point	Monitor	GE (cpm)	SAE (cpm)	Alert (cpm)	UE (cpm)
Gaseous	SBGT A	SPING 5				
		SPING 7		3.73E+5	3.74E+4	2.87E+2
		AXM 3	2.54E+5			
	SBGT B	Canberra normal				9.70E+5
		Canberra high	4.03E+7	4.03E+6	4.03E+5	
	Containment Vent	SPING 5				6.68E+5
		SPING 7		1.98E+5	1.98E+4	
		AXM 3	1.35E+5			
	Radwaste Building Vent	SPING 5				8.34E+4
		SPING 7		1.72E+3	1.72E+2	
		AXM 3	1.17E+3			
	Turbine Building Vent	SPING 5				2.67E+5
		SPING 7		4.26E+3	4.26E+2	
		AXM 3	2.90E+3			
	Fuel Handling (Aux Bldg) Vent	SPING 5				1.29E+5
		SPING 7		6.43E+5	6.44E+4	
		AXM 3	4.38E+5			
Liquid	Radwaste	Liquid Effluent Monitor	N/A	N/A	N/A	7.33E+5

5.3. Gaseous Effluent Monitor Reading Results in Ci/sec

	Release Point	GE (Ci/sec)	SAE (Ci/sec)	Alert (Ci/sec)	UE (Ci/sec)
Gaseous	SBGT A SBGT B	8.1E+2	8.1E+1	8.1E+0	6.7E-2
	Containment Vent	6.4E+2	6.4E+1	6.4E+0	6.7E-2
	Radwaste Building Vent	5.1E+1	5.1E+0	5.1E-1	6.7E-2
	Turbine Building Vent	1.3E+1	1.3E+0	1.3E-1	6.7E-2
	Fuel Handling (Aux Bldg) Vent	8.6E+3	8.6E+2	8.6E+1	6.7E-2

Note UE release rate values assume a common source term and are two times (2x) the lesser of the Attachment 2 calculated total body or skin ODCM limit setpoint in Ci/sec.

6. REFERENCES

6.1. NEI 99-01 Revision 6, Methodology for Development of Emergency Action Levels, November 2012

6.2. 10 CFR 20 Appendix B Table 2 Column 2

6.3. EPA-400-R-92-001, Manual of Protective Actions for Nuclear Incidents, May 1992

6.4. NUREG-1228, Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents, October 1988

6.5. NUREG-1465, Accident Source Terms for Light-Water Nuclear Power Plants, February 1995

6.6. NUREG-1940, RASCAL 4: Description of Models and Methods, December 2012

6.7. Grand Gulf Nuclear Station Offsite Dose Calculation Manual, LBDCR 15051, January 2017

6.8. Unified RASCAL Interface Requirement Specification, Draft 051611

6.9. Unified RASCAL Interface Requirement Specification Grand Gulf Site Annex, Version 2, Draft 022414

6.10. Grand Gulf Nuclear Station UFSAR

- 2.3.2.1.1.1, Wind Distributions {All Meteorological Conditions), Revision 0
- 9.4.4.2, Turbine Building Ventilation System – System Description, Revision 10
- 11.2.1.1, Liquid Radwaste System – Power Generation Design Bases, LDC 05074
- Table 2.3-34, Percentage Frequency of Wind Direction and Speed at Grand Gulf Site Period of Record - August 1972 to July 1973, Revision 0
- Tables 2.3-130A through 130G, Frequency Distribution For Pasquill Stability Class A-G, Revision 0
- Tables 2.3-130H, Frequency Distribution 1972 to 1976, Revision 0
- Table 11.2-10, Liquid Effluents Annual Releases to Discharge Canal, LBDCR 13002
- Table 11.3-9, Expected Annual Release of Gaseous Effluents, LBDCR 13002
- Table 11.5-1, Process and Effluent Radioactivity Monitoring Systems, Revision 2016-00
- Figure 2.3-2, Annual Wind Rose August 1972 - July l973, Revision 0
- Figure 2.3-3, Comparison of Wind Directions and Speeds at Grand Gulf, Miss., 1972 - 1974 and at Jackson, Miss., 1960 - 1964, Revision 0

6.11. GIN-2001/01196, Liquid Process and Liquid Effluent Radiation Monitor Calibration Basis

6.12. GIN-2007/00076, Review of 2001-2005 Annual Average Relative Concentration and Relative Deposition

6.13. 08-S-03-22, Installed Radiation Monitoring System Alarm Setpoint Determination and Control, TCN 012

6.14. 06-HP-1DI7-V-0001, Turbine Building Occasional release point Monitoring Instrumentation, Revision 101

6.15. EC 57863, Replacement of SBGT "B" Radiation Monitoring System, Revision 0

6.16. SFD1102, System Flow Diagram – Standby Gas Treatment System, Revision 003

6.17. SFD1100, System Flow Diagram – Containment Cooling System, Revision 006

6.18. SFD1104A, System Flow Diagram – Fuel Handling Area Ventilation System, Revision 006

6.19. SFD0047, System Flow Diagram – Radwaste Building Ventilation System, Sheet 3, Revision 008

6.20. SFD1105A, Turbine Building Ventilation System, Revision 012

6.21. Liquid Release Permit Report, Permit Number 2017007

6.22. Liquid Release Permit Report, Permit Number 2017008

6.23. Liquid Release Permit Report, Permit Number 2017009

	Permit Number: 2017007 (µCi/ml)	Permit Number: 2017008 (µCi/ml)	Permit Number: 2017009 (µCi/ml)	Total Liquid Discharge (µCi/ml)	Normalized LRW Isotopes (Fraction)	10 CFR 20 Liquid Limit (µCi/ml)	ODCM Liquid Limit [10x 10 CFR 20 Limit] ECi (µCi/ml)	Fraction x ODCM Limit (µCi/ml)
Na-24		8.75E-08		8.75E-08	1.38E-02	5.00E-05	5.00E-04	6.88E-06
Mn-54	9.53E-08	3.21E-08	3.35E-07	4.62E-07	7.27E-02	3.00E-05	3.00E-04	2.18E-05
Co-60	6.17E-07	4.23E-07	1.27E-06	2.31E-06	3.63E-01	3.00E-06	3.00E-05	1.09E-05
Zn-65	3.04E-07	4.01E-07	1.32E-06	2.03E-06	3.18E-01	5.00E-06	5.00E-05	1.59E-05
Ag-110m	2.51E-07	4.58E-08		2.97E-07	4.66E-02	6.00E-06	6.00E-05	2.80E-06
Sb-125			1.71E-07	1.71E-07	2.69E-02	3.00E-05	3.00E-04	8.06E-06
Cs-134			3.92E-07	3.92E-07	6.16E-02	9.00E-07	9.00E-06	5.54E-07
Cs-137			6.18E-07	6.18E-07	9.71E-02	1.00E-06	1.00E-05	9.71E-07
	1.27E-06	9.89E-07	4.11E-06	6.36E-06	1.00E+00			6.79E-05

Monitor	Maximum Setpoint Discharge Flow - f (gpm)	Minimum Setpoint Dilution Flow - F (gpm)	Rad Monitor Setpoint - SP (cpm)	AU1.1 EAL Threshold Value (cpm)
Liquid Radwaste Monitor	9.00E+01	3.15E+03	3.66E+05	**7.33E+05**

Maximum Discharge Flow (gpm):	1.00E+02
Minimum Dilution Flow (gpm):	3.50E+03

Monitor Response Factor - MRF (cpm/µCi/ml):	3.00E+08
Safety Factor:	2

	Total Body Dose Factor - Ki (mrem/yr per µCi/m3)	Skin Beta Dose Factor - Li (mrem/yr per µCi/m3)	Gamma Air Dose Factor - Mi (mrad/yr per µCi/m3)	FSAR Table 11.3-9 Source Term - (Ci/yr)	Source Term Fraction - Qi	Qi x Ki (mrem/yr per µCi/m3)	Qi x (Li + 1.1Mi) (mrem/yr per µCi/m3)
Ar-41	8.8E+03	2.7E+03	9.3E+03	7.2E+01	9.0E-03	7.93E+01	1.16E+02
Kr-83m	7.6E-02	0.0E+00	1.9E+01				
Kr-85m	1.2E+03	1.5E+03	1.2E+03	8.8E+01	1.1E-02	1.28E+01	3.08E+01
Kr-85	1.6E+01	1.3E+03	1.7E+01	3.9E+02	4.9E-02	7.82E-01	6.60E+01
Kr-87	5.9E+03	9.7E+03	6.2E+03	6.3E+01	7.8E-03	4.64E+01	1.30E+02
Kr-88	1.5E+04	2.4E+03	1.5E+04	9.8E+01	1.2E-02	1.79E+02	2.33E+02
Kr-89	1.7E+04	1.0E+04	1.7E+04	6.1E+02	7.6E-02	1.26E+03	2.21E+03
Kr-90	1.6E+04	7.3E+03	1.6E+04				
Xe-131m	9.2E+01	4.8E+02	1.6E+02	2.0E+01	2.5E-03	2.28E-01	1.61E+00
Xe-133m	2.5E+02	9.9E+02	3.3E+02				
Xe-133	2.9E+02	3.1E+02	3.5E+02	2.2E+03	2.7E-01	8.05E+01	1.90E+02
Xe-135m	3.1E+03	7.1E+02	3.4E+03	9.9E+02	1.2E-01	3.85E+02	5.43E+02
Xe-135	1.8E+03	1.9E+03	1.9E+03	1.2E+03	1.5E-01	2.70E+02	5.94E+02
Xe-137	1.4E+03	1.2E+04	1.5E+03	1.3E+03	1.6E-01	2.30E+02	2.24E+03
Xe-138	8.8E+03	4.1E+03	9.2E+03	1.0E+03	1.2E-01	1.10E+03	1.78E+03
				8.0E+03	**1.0E+00**	**3.64E+03**	**8.14E+03**

Calculation Constants

	SBGT A	SBGT B	Cont	FHA	RW	TB
Dispersion - X/Q (sec/m3):	4.10E-06	4.10E-06	4.10E-06	4.10E-06	4.10E-06	4.10E-06
Effluent Flow - f (cfm):	4.30E+03	4.30E+03	6.00E+03	3.10E+04	4.80E+04	1.50E+04
GE Eff (µCi/cc / cpm):			7.69E-08	7.69E-08	7.69E-08	7.69E-08
SPING Ch 5 Eff (µCi/cc / cpm):	3.54E-08		3.54E-08	3.54E-08	3.54E-08	3.54E-08
SPING Ch 7 Eff (µCi/cc / cpm):	1.15E-04		1.15E-04	1.15E-04	1.15E-04	1.15E-04
AXM Ch 4 Eff (µCi/cc / cpm):	3.04E-06		3.04E-06	3.04E-06	3.04E-06	3.04E-06
Canberra Norm Eff (µCi/cc / cpm):		3.40E-08				
Canberra High Eff (µCi/cc / cpm):		9.90E-06				

Total Body Limit (mrem/yr):	500	UCF (cc/sec per cfm):	472
Skin Dose Limit (mrem/yr):	3000	DCF (mrad to mrem):	1.1

Calculated Setpoint Results

	SBGT A	SBGT B	Cont	FHA	RW	TB
SP-TB (µCi/sec):	3.35E+04	3.35E+04	3.35E+04	3.35E+04	3.35E+04	3.35E+04
SP-Skin (µCi/sec):	8.99E+04	8.99E+04	8.99E+04	8.99E+04	8.99E+04	8.99E+04
SP-TB (µCi/cc):	1.65E-02	1.65E-02	1.18E-02	2.29E-03	1.48E-03	4.73E-03
SP-Skin (µCi/cc):	4.43E-02	4.43E-02	3.18E-02	6.15E-03	3.97E-03	1.27E-02
GE SP-TB (cpm):			1.54E+05	2.97E+04	1.92E+04	6.15E+04
GE SP-Skin (cpm):			4.13E+05	7.99E+04	5.16E+04	1.65E+05
SPING Ch 5 SP-TB (cpm):	4.66E+05		3.34E+05	6.46E+04	4.17E+04	1.34E+05
SPING Ch 5 SP-Skin (cpm):	1.25E+06		8.97E+05	1.74E+05	1.12E+05	3.59E+05
SPING Ch 7 SP-TB (cpm):	1.43E+02		1.03E+02	1.99E+01	1.28E+01	4.11E+01
SPING Ch 7 SP-Skin (cpm):	3.85E+02		2.76E+02	5.35E+01	3.45E+01	1.10E+02
AXM Ch 4 SP-TB (cpm):	5.42E+03		3.89E+03	7.52E+02	4.86E+02	1.55E+03
AXM Ch 4 SP-Skin (cpm):	1.46E+04		1.04E+04	2.02E+03	1.31E+03	4.18E+03
Canberra Norm SP-TB (cpm):		4.85E+05				
Canberra Norn SP-Skin (cpm):		1.30E+06				
Canberra High SP-TB (cpm):		1.67E+03				
Canberra High SP-Skin (cpm):		4.48E+03				

Calculated UE Threshold Results

GE AU1.1 (cpm):			**3.07E+05**	**5.95E+04**	**3.84E+04**	**1.23E+05**
SPING Ch 5 AU1.1 (cpm):	**9.32E+05**		**6.68E+05**	**1.29E+05**	**8.34E+04**	**2.67E+05**
SPING Ch 7 AU1.1 (cpm):	**2.87E+02**		**2.05E+02**	**3.98E+01**	**2.57E+01**	**8.22E+01**
AXM Ch 4 AU1.1 (cpm):	**1.08E+04**		**7.77E+03**	**1.50E+03**	**9.72E+02**	**3.11E+03**
Canberra Norm AU1.1 (cpm):		**9.70E+05**				
Canberra High AU1.1 (cpm):		**3.33E+03**				
AU1.1 (Ci/sec):	**6.69E-02**	**6.69E-02**	**6.69E-02**	**6.69E-02**	**6.69E-02**	**6.69E-02**

	GE (AG1.1)	SAE (AS1.1)	Alert (AA1.1)
Canberra Norm (cpm):	1.17E+10	1.17E+09	1.17E+08
Canberra High (cpm):	4.03E+07	4.03E+06	4.03E+05

SBGT SPING 7 Release Rate @ General Emergency (Ci/sec):	8.09E+02
UCF (cc/sec per cfm):	472
UCF (µCi per Ci):	1.00E+06
Effluent Flow - f (cfm):	4.30E+03
Normal Channel Eff (µCi/cc/cpm):	3.40E-08
High Channel Eff (µCi/cc/cpm):	9.90E-06

SBGT SPING 7 – General Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 20:30

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain><Aux Bldg><SBGT><Env> — PRF: 1.60E-05

Containment HUT: = < 2 Hours | Containment Sprays: = OFF | Supp Pool Status: = Subcooled | Safety Filters: = Working

HVAC Filters: = N/A | Aux Bldg HUT: = < 2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 1:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

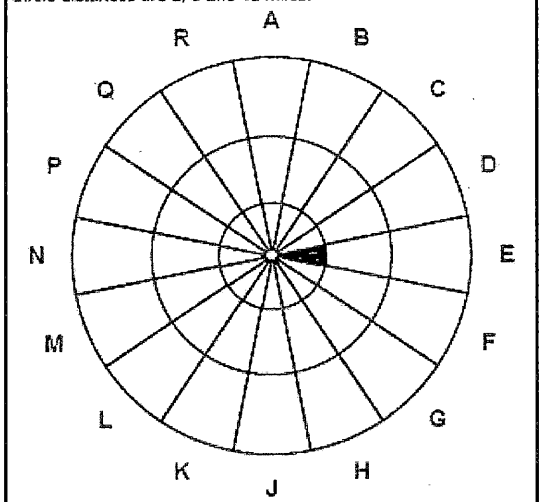
Monitor: SPING ch 7 Readings: 3.73E+06 cpm Flowrate: 4000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.43E+03	9.99E+02	2.73E+00	1.06E+00	1.00E+03	6.92E+01
0.5	1.19E+03	8.32E+02	1.98E+00	7.71E-01	8.35E+02	5.04E+01
0.7	8.44E+02	5.84E+02	1.16E+01	9.49E+00	6.05E+02	3.15E+01
1.0	5.56E+02	3.81E+02	1.12E+01	9.29E+00	4.02E+02	1.83E+01
1.5	3.29E+02	2.24E+02	9.40E+00	7.54E+00	2.41E+02	9.88E+00
2.0	2.59E+02	1.76E+02	7.52E+00	5.77E+00	1.90E+02	6.24E+00
3.0	1.82E+02	1.24E+02	3.87E+00	2.94E+00	1.31E+02	3.60E+00
4.0	1.51E+02	1.03E+02	3.66E+00	2.69E+00	1.09E+02	3.03E+00
5.0	1.30E+02	8.70E+01	3.37E+00	2.39E+00	9.28E+01	2.64E+00
7.0	9.80E+01	6.67E+01	2.82E+00	1.88E+00	7.14E+01	2.14E+00
10.0	6.72E+01	4.52E+01	1.99E+00	1.23E+00	4.84E+01	1.59E+00

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 203034.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



PAGs Exceeded in Designated Areas

Release Rates (Ci / sec)	
Particulate	1.25E-03 (0.0%)
Iodine	2.41E-02 (0.0%)
Noble Gas	8.09E+02 (100.0%)

***** Classification: General Emergency *****

Reviewed By: _____

Page 1 of 3 — Grand Gulf / 2.0.1.0

SBGT SPING 7 – Site Area Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 20:36

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain><Aux Bldg><SBGT><Env> — PRF: 1.60E-05

Containment HUT: = < 2 Hours	Containment Sprays: = OFF	Supp Pool Status: = Subcooled	Safety Filters: = Working
HVAC Filters: = N/A	Aux Bldg HUT: = < 2 Hours	Turbine Bldg HUT: = N/A	RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad — OnSite Lower

Time After S/D (hh:mm): 1:00 — Wind: From 270° @ 4.4 mph

Release Duration (hh:mm): 1:00 — ETE (hh:mm): [N/A] — Stability Class: D

Precipitation: None

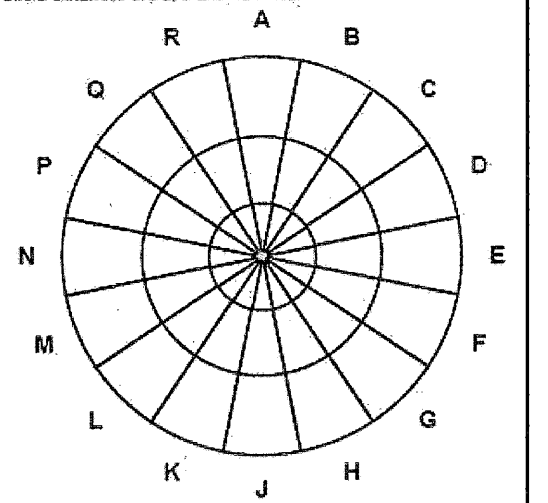
Monitor: SPING ch 7 — Readings: 3.73E+05 cpm — Flowrate: 4000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.43E+02	9.99E+01	2.73E-01	1.06E-01	1.00E+02	6.92E+00
0.5	1.19E+02	8.32E+01	1.98E-01	0.00E+00	8.34E+01	5.04E+00
0.7	8.44E+01	5.84E+01	1.16E+00	9.49E-01	6.05E+01	3.15E+00
1.0	5.56E+01	3.81E+01	1.12E+00	9.29E-01	4.02E+01	1.83E+00
1.5	3.29E+01	2.24E+01	9.40E-01	7.54E-01	2.41E+01	9.88E-01
2.0	2.59E+01	1.76E+01	7.52E-01	5.77E-01	1.90E+01	6.24E-01
3.0	1.82E+01	1.24E+01	3.87E-01	2.94E-01	1.31E+01	3.60E-01
4.0	1.51E+01	1.03E+01	3.66E-01	2.69E-01	1.09E+01	3.03E-01
5.0	1.30E+01	8.70E+00	3.37E-01	2.39E-01	9.28E+00	2.64E-01
7.0	9.80E+00	6.67E+00	2.82E-01	1.88E-01	7.14E+00	2.14E-01
10.0	6.72E+00	4.52E+00	1.99E-01	1.23E-01	4.84E+00	1.59E-01

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 203603.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Classification: Site Area Emergency

Release Rates (Ci / sec)	
Particulate	1.25E-04 (0.0%)
Iodine	2.41E-03 (0.0%)
Noble Gas	8.09E+01 (100.0%)

Reviewed By: _____

Page 1 of 3 — Grand Gulf / 2.0.1.0

SBGT SPING 7 – Alert

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 20:38

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain><Aux Bldg><SBGT><Env> — PRF: 1.60E-05

Containment HUT: = < 2 Hours | Containment Sprays: = OFF | Supp Pool Status: = Subcooled | Safety Filters: = Working

HVAC Filters: = N/A | Aux Bldg HUT: = < 2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 1:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

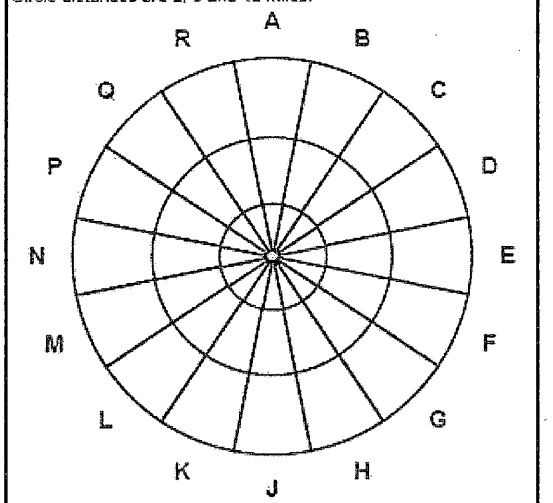
Monitor: SPING ch 7 Readings: 3.74E+04 cpm Flowrate: 4000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.44E+01	1.00E+01	0.00E+00	0.00E+00	1.00E+01	6.93E-01
0.5	1.20E+01	8.36E+00	0.00E+00	0.00E+00	8.36E+00	5.04E-01
0.7	8.48E+00	5.88E+00	1.16E-01	0.00E+00	6.00E+00	3.16E-01
1.0	5.60E+00	3.84E+00	1.14E-01	0.00E+00	3.95E+00	1.84E-01
1.5	3.32E+00	2.26E+00	0.00E+00	0.00E+00	2.26E+00	0.00E+00
2.0	2.61E+00	1.78E+00	0.00E+00	0.00E+00	1.78E+00	0.00E+00
3.0	1.84E+00	1.25E+00	0.00E+00	0.00E+00	1.25E+00	0.00E+00
4.0	1.52E+00	1.04E+00	0.00E+00	0.00E+00	1.04E+00	0.00E+00
5.0	1.31E+00	8.76E-01	0.00E+00	0.00E+00	8.76E-01	0.00E+00
7.0	9.88E-01	6.72E-01	0.00E+00	0.00E+00	6.72E-01	0.00E+00
10.0	6.76E-01	4.55E-01	0.00E+00	0.00E+00	4.55E-01	0.00E+00

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 203804.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Release Rates (Ci / sec)	
Particulate	1.25E-05 (0.0%)
Iodine	2.41E-04 (0.0%)
Noble Gas	8.12E+00 (100.0%)

*** Classification: Validate against Emergency Action Levels ***

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

SBGT AXM 3 – General Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 20:40

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain><Aux Bldg><SBGT><Env> — PRF: 1.60E-05

Containment HUT: = < 2 Hours | Containment Sprays: = OFF | Supp Pool Status: = Subcooled | Safety Filters: = Working

HVAC Filters: = N/A | Aux Bldg HUT: = < 2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 1:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

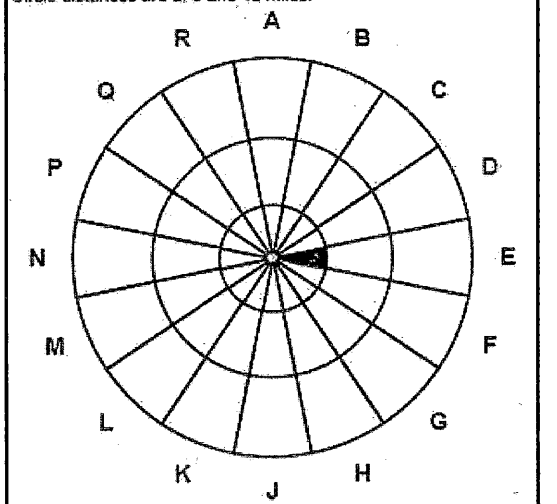
Monitor: AXM ch 3 Readings: 2.54E+05 cpm Flowrate: 4000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.43E+03	9.99E+02	2.73E+00	1.06E+00	1.00E+03	6.92E+01
0.5	1.19E+03	8.32E+02	1.98E+00	7.71E-01	8.35E+02	5.04E+01
0.7	8.44E+02	5.84E+02	1.16E+01	9.50E+00	6.05E+02	3.15E+01
1.0	5.56E+02	3.82E+02	1.13E+01	9.30E+00	4.02E+02	1.83E+01
1.5	3.30E+02	2.24E+02	9.40E+00	7.58E+00	2.41E+02	9.88E+00
2.0	2.59E+02	1.76E+02	7.52E+00	5.78E+00	1.90E+02	6.24E+00
3.0	1.82E+02	1.24E+02	3.87E+00	2.95E+00	1.31E+02	3.60E+00
4.0	1.51E+02	1.03E+02	3.66E+00	2.69E+00	1.09E+02	3.03E+00
5.0	1.30E+02	8.71E+01	3.37E+00	2.39E+00	9.29E+01	2.64E+00
7.0	9.80E+01	6.67E+01	2.82E+00	1.88E+00	7.14E+01	2.14E+00
10.0	6.72E+01	4.52E+01	1.99E+00	1.23E+00	4.84E+01	1.59E+00

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 204045.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



PAGs Exceeded in Designated Areas

Release Rates (Ci / sec)	
Particulate	1.25E-03 (0.0%)
Iodine	2.41E-02 (0.0%)
Noble Gas	8.10E+02 (100.0%)

*** Classification: General Emergency ***

Reviewed By: _____

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SBGT AXM 3 – Site Area Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 20:42

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain><Aux Bldg><SBGT><Env> — PRF: 1.60E-05

Containment HUT: = < 2 Hours	Containment Sprays: = OFF	Supp Pool Status: = Subcooled	Safety Filters: = Working
HVAC Filters: = N/A	Aux Bldg HUT: = < 2 Hours	Turbine Bldg HUT: = N/A	RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad — OnSite Lower

Time After S/D (hh:mm): 1:00 — Wind: From 270° @ 4.4 mph

Release Duration (hh:mm): 1:00 — ETE (hh:mm): [N/A] — Stability Class: D

Precipitation: None

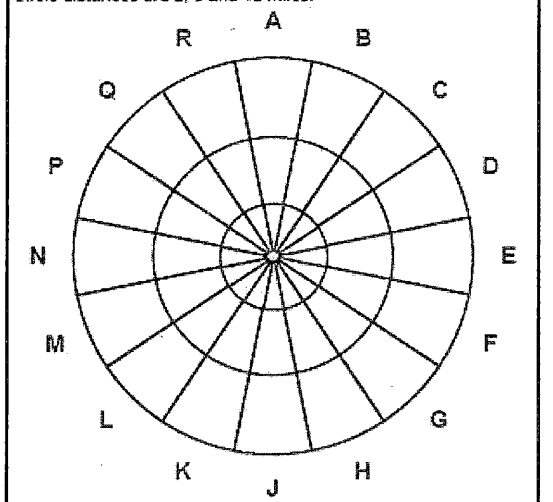
Monitor: AXM ch 3 — Readings: 2.54E+04 cpm — Flowrate: 4000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.43E+02	9.99E+01	2.73E-01	1.06E-01	1.00E+02	6.92E+00
0.5	1.19E+02	8.32E+01	1.98E-01	0.00E+00	8.34E+01	5.04E+00
0.7	8.44E+01	5.84E+01	1.16E+00	9.50E-01	6.05E+01	3.15E+00
1.0	5.56E+01	3.82E+01	1.13E+00	9.30E-01	4.02E+01	1.83E+00
1.5	3.30E+01	2.24E+01	9.40E-01	7.58E-01	2.41E+01	9.88E-01
2.0	2.59E+01	1.76E+01	7.52E-01	5.78E-01	1.90E+01	6.24E-01
3.0	1.82E+01	1.24E+01	3.87E-01	2.95E-01	1.31E+01	3.60E-01
4.0	1.51E+01	1.03E+01	3.66E-01	2.69E-01	1.09E+01	3.03E-01
5.0	1.30E+01	8.71E+00	3.37E-01	2.39E-01	9.29E+00	2.64E-01
7.0	9.80E+00	6.67E+00	2.82E-01	1.88E-01	7.14E+00	2.14E-01
10.0	6.72E+00	4.52E+00	1.99E-01	1.23E-01	4.84E+00	1.59E-01

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 204251.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Classification: Site Area Emergency

Release Rates (Ci / sec)	
Particulate	1.25E-04 (0.0%)
Iodine	2.41E-03 (0.0%)
Noble Gas	8.10E+01 (100.0%)

Reviewed By: _____

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SBGT AXM 3 – Alert

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 20:44

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain><Aux Bldg><SBGT><Env> — PRF: 1.60E-05

Containment HUT: = < 2 Hours | Containment Sprays: = OFF | Supp Pool Status: = Subcooled | Safety Filters: = Working

HVAC Filters: = N/A | Aux Bldg HUT: = < 2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad — OnSite Lower

Time After S/D (hh:mm): 1:00 — Wind: From 270° @ 4.4 mph

Release Duration (hh:mm): 1:00 — ETE (hh:mm): [N/A] — Stability Class: D

Precipitation: None

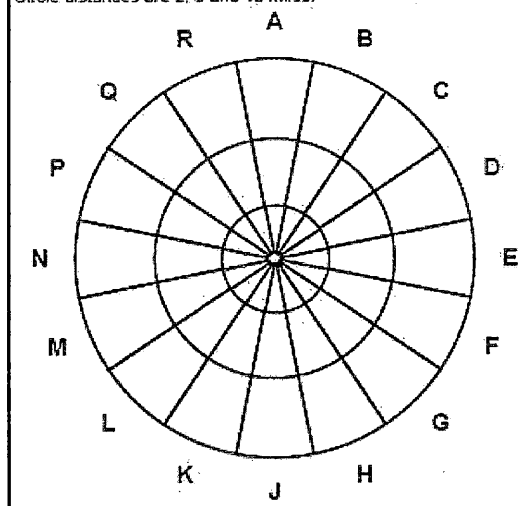
Monitor: AXM ch 3 — Readings: 2.55E+03 cpm — Flowrate: 4000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.44E+01	1.01E+01	0.00E+00	0.00E+00	1.01E+01	6.93E-01
0.5	1.20E+01	8.40E+00	0.00E+00	0.00E+00	8.40E+00	5.04E-01
0.7	8.48E+00	5.88E+00	1.16E-01	0.00E+00	6.00E+00	3.16E-01
1.0	5.60E+00	3.84E+00	1.14E-01	0.00E+00	3.96E+00	1.84E-01
1.5	3.32E+00	2.26E+00	0.00E+00	0.00E+00	2.26E+00	0.00E+00
2.0	2.61E+00	1.78E+00	0.00E+00	0.00E+00	1.78E+00	0.00E+00
3.0	1.84E+00	1.25E+00	0.00E+00	0.00E+00	1.25E+00	0.00E+00
4.0	1.52E+00	1.04E+00	0.00E+00	0.00E+00	1.04E+00	0.00E+00
5.0	1.31E+00	8.77E-01	0.00E+00	0.00E+00	8.77E-01	0.00E+00
7.0	9.88E-01	6.72E-01	0.00E+00	0.00E+00	6.72E-01	0.00E+00
10.0	6.76E-01	4.56E-01	0.00E+00	0.00E+00	4.56E-01	0.00E+00

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 204441.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

*** Classification: Validate against Emergency Action Levels ***

Release Rates (Ci / sec)	
Particulate	1.26E-05 (0.0%)
Iodine	2.42E-04 (0.0%)
Noble Gas	8.13E+00 (100.0%)

Reviewed By: _____

Page 1 of 3 — Grand Gulf / 2.0.1.0

SBGT AXM 4 – Site Area Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 20:47

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain><Aux Bldg><SBGT><Env> — PRF: 1.60E-05

Containment HUT: = < 2 Hours	Containment Sprays: = OFF	Supp Pool Status: = Subcooled	Safety Filters: = Working
HVAC Filters: = N/A	Aux Bldg HUT: = < 2 Hours	Turbine Bldg HUT: = N/A	RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad — OnSite Lower

Time After S/D (hh:mm): 1:00 — Wind: From 270° @ 4.4 mph

Release Duration (hh:mm): 1:00 — ETE (hh:mm): [N/A] — Stability Class: D

Precipitation: None

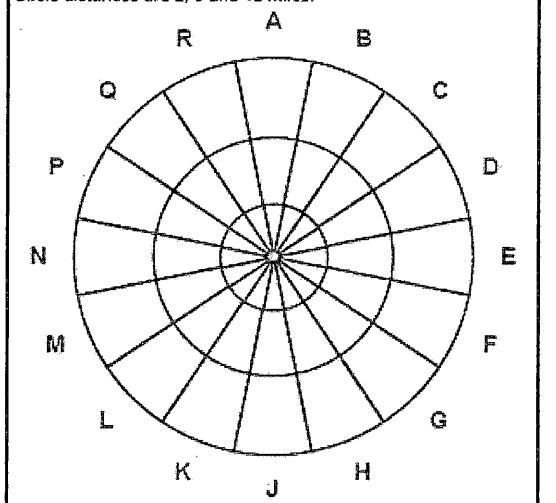
Monitor: AXM ch 4 — Readings: 1.40E+07 cpm — Flowrate: 4000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.42E+02	9.98E+01	2.71E-01	1.05E-01	1.00E+02	6.86E+00
0.5	1.18E+02	8.32E+01	1.97E-01	0.00E+00	8.34E+01	5.00E+00
0.7	8.40E+01	5.80E+01	1.16E+00	9.46E-01	6.01E+01	3.13E+00
1.0	5.52E+01	3.80E+01	1.12E+00	9.26E-01	4.01E+01	1.82E+00
1.5	3.28E+01	2.24E+01	9.40E-01	7.54E-01	2.41E+01	9.80E-01
2.0	2.58E+01	1.76E+01	7.52E-01	5.77E-01	1.89E+01	6.20E-01
3.0	1.82E+01	1.24E+01	3.87E-01	2.94E-01	1.30E+01	3.57E-01
4.0	1.50E+01	1.03E+01	3.66E-01	2.68E-01	1.09E+01	3.01E-01
5.0	1.30E+01	8.68E+00	3.37E-01	2.38E-01	9.25E+00	2.62E-01
7.0	9.76E+00	6.66E+00	2.81E-01	1.87E-01	7.13E+00	2.13E-01
10.0	6.72E+00	4.51E+00	1.99E-01	1.23E-01	4.83E+00	1.58E-01

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 204719.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Classification: Site Area Emergency

Release Rates (Ci / sec)	
Particulate	1.24E-04 (0.0%)
Iodine	2.39E-03 (0.0%)
Noble Gas	8.03E+01 (100.0%)

Reviewed By: _____

Page 1 of 3 — Grand Gulf / 2.0.1.0

SBGT AXM 4 – Alert

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 20:49

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain><Aux Bldg><SBGT><Env> — PRF: 1.60E-05

Containment HUT: = < 2 Hours | Containment Sprays: = OFF | Supp Pool Status: = Subcooled | Safety Filters: = Working

HVAC Filters: = N/A | Aux Bldg HUT: = < 2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad — OnSite Lower

Time After S/D (hh:mm): 1:00 — Wind: From 270° @ 4.4 mph

Release Duration (hh:mm): 1:00 — ETE (hh:mm): [N/A] — Stability Class: D

Precipitation: None

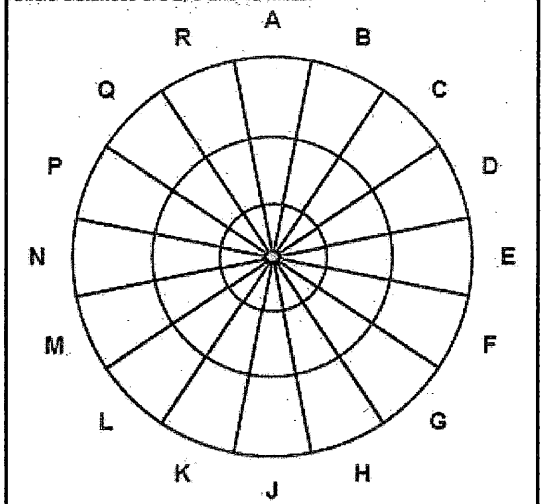
Monitor: AXM ch 4 — Readings: 1.42E+06 cpm — Flowrate: 4000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.44E+01	1.01E+01	0.00E+00	0.00E+00	1.01E+01	6.94E-01
0.5	1.20E+01	8.40E+00	0.00E+00	0.00E+00	8.40E+00	5.04E-01
0.7	8.48E+00	5.88E+00	1.17E-01	0.00E+00	6.00E+00	3.17E-01
1.0	5.60E+00	3.85E+00	1.14E-01	0.00E+00	3.96E+00	1.84E-01
1.5	3.32E+00	2.26E+00	0.00E+00	0.00E+00	2.26E+00	0.00E+00
2.0	2.61E+00	1.78E+00	0.00E+00	0.00E+00	1.78E+00	0.00E+00
3.0	1.84E+00	1.25E+00	0.00E+00	0.00E+00	1.25E+00	0.00E+00
4.0	1.52E+00	1.04E+00	0.00E+00	0.00E+00	1.04E+00	0.00E+00
5.0	1.32E+00	8.78E-01	0.00E+00	0.00E+00	8.78E-01	0.00E+00
7.0	9.88E-01	6.73E-01	0.00E+00	0.00E+00	6.73E-01	0.00E+00
10.0	6.76E-01	4.56E-01	0.00E+00	0.00E+00	4.56E-01	0.00E+00

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 204944.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Release Rates (Ci / sec)	
Particulate	1.26E-05 (0.0%)
Iodine	2.42E-04 (0.0%)
Noble Gas	8.15E+00 (100.0%)

*** Classification: Validate against Emergency Action Levels ***

Reviewed By: _____

Page 1 of 3 — Grand Gulf / 2.0.1.0

Containment Vent SPING 7– General Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 20:55

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain> <HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = < 2 Hours	Containment Sprays: = OFF	Supp Pool Status: = Subcooled	Safety Filters: = N/A
HVAC Filters: = Working	Aux Bldg HUT: = N/A	Turbine Bldg HUT: = N/A	RadWaste Bldg.HUT: = N/A

Source Term: Reactor Core Accident - Clad — OnSite Lower

Time After S/D (hh:mm): 1:00 — Wind: From 270° @ 4.4 mph

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A] — Stability Class: D

Precipitation: None

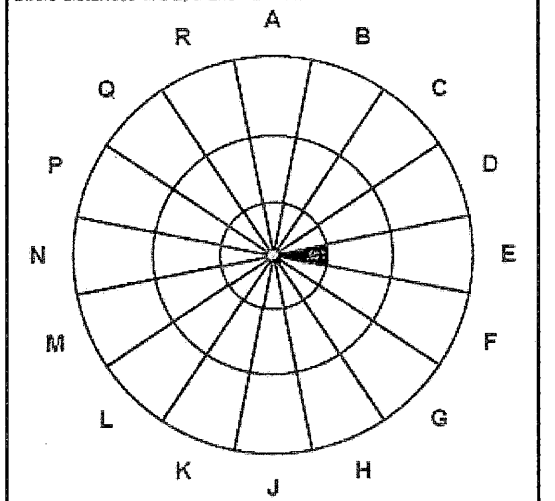
Monitor: SPING ch 7 Readings: 1.98E+06 cpm Flowrate: 6000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S/B	1.21E+03	8.39E+02	1.15E+02	4.89E+01	1.00E+03	2.71E+03
0.5	9.92E+02	6.84E+02	8.80E+01	4.04E+01	8.12E+02	1.90E+03
0.7	6.76E+02	4.64E+02	5.84E+01	2.91E+01	5.51E+02	1.14E+03
1.0	4.28E+02	2.92E+02	3.71E+01	1.96E+01	3.49E+02	6.28E+02
1.5	2.39E+02	1.62E+02	2.13E+01	1.14E+01	1.94E+02	3.28E+02
2.0	1.88E+02	1.28E+02	1.41E+01	7.38E+00	1.49E+02	2.09E+02
3.0	1.53E+02	1.05E+02	8.93E+00	4.58E+00	1.19E+02	1.40E+02
4.0	1.24E+02	8.51E+01	7.71E+00	3.89E+00	9.67E+01	1.16E+02
5.0	1.02E+02	6.77E+01	6.42E+00	3.17E+00	7.73E+01	9.55E+01
7.0	7.84E+01	5.18E+01	5.26E+00	2.47E+00	5.96E+01	7.87E+01
10.0	5.04E+01	3.35E+01	3.69E+00	1.61E+00	3.88E+01	5.75E+01

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 205609.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



PAGs Exceeded in Designated Areas

Release Rates (Ci / sec)	
Particulate	4.98E-02 (0.0%)
Iodine	9.56E-01 (0.1%)
Noble Gas	6.44E+02 (99.8%)

***** Classification: General Emergency *****

Reviewed By: _____

Page 1 of 3 — Grand Gulf / 2.0.1.0

Containment Vent SPING 7– Site Area Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 20:56

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain> <HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = < 2 Hours | Containment Sprays: = OFF | Supp Pool Status: = Subcooled | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = N/A | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident – Clad

Time After S/D (hh:mm): 1:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

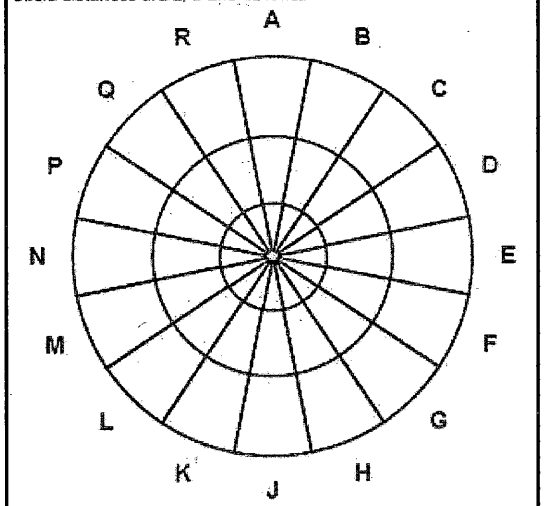
Monitor: SPING ch 7 Readings: 1.98E+05 cpm Flowrate: 6000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.21E+02	8.39E+01	1.15E+01	4.89E+00	1.00E+02	2.71E+02
0.5	9.92E+01	6.84E+01	8.80E+00	4.04E+00	8.12E+01	1.90E+02
0.7	6.76E+01	4.64E+01	5.84E+00	2.91E+00	5.51E+01	1.14E+02
1.0	4.28E+01	2.92E+01	3.71E+00	1.96E+00	3.49E+01	6.28E+01
1.5	2.39E+01	1.62E+01	2.13E+00	1.14E+00	1.94E+01	3.28E+01
2.0	1.88E+01	1.28E+01	1.41E+00	7.38E-01	1.49E+01	2.09E+01
3.0	1.53E+01	1.05E+01	8.93E-01	4.58E-01	1.19E+01	1.40E+01
4.0	1.24E+01	8.51E+00	7.71E-01	3.89E-01	9.67E+00	1.16E+01
5.0	1.02E+01	6.77E+00	6.42E-01	3.17E-01	7.73E+00	9.55E+00
7.0	7.84E+00	5.18E+00	5.26E-01	2.47E-01	5.96E+00	7.87E+00
10.0	5.04E+00	3.35E+00	3.69E-01	1.61E-01	3.88E+00	5.75E+00

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 205648.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Release Rates (Ci / sec)	
Particulate	4.98E-03 (0.0%)
Iodine	9.58E-02 (0.1%)
Noble Gas	6.44E+01 (99.8%)

Classification: Site Area Emergency

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

Containment Vent SPING 7 – Alert

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 20:57

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain> <HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = < 2 Hours — Containment Sprays: = OFF — Supp Pool Status: = Subcooled — Safety Filters: = N/A

HVAC Filters: = Working — Aux Bldg HUT: = N/A — Turbine Bldg HUT: = N/A — RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 1:00

Release Duration (hh:mm): 1:00 — ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

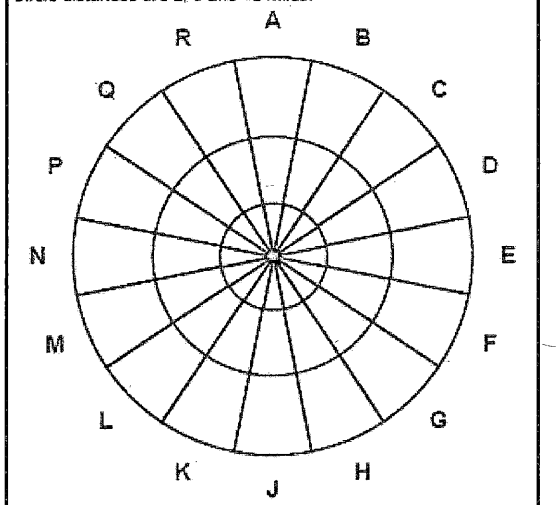
Monitor: SPING ch 7 — Readings: 1.98E+04 cpm — Flowrate: 6000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.21E+01	8.39E+00	1.15E+00	4.89E-01	1.00E+01	2.71E+01
0.5	9.92E+00	6.84E+00	8.80E-01	4.04E-01	8.12E+00	1.90E+01
0.7	6.76E+00	4.64E+00	5.84E-01	2.91E-01	5.51E+00	1.14E+01
1.0	4.28E+00	2.92E+00	3.71E-01	1.96E-01	3.49E+00	6.28E+00
1.5	2.39E+00	1.62E+00	2.13E-01	1.14E-01	1.94E+00	3.28E+00
2.0	1.88E+00	1.28E+00	1.41E-01	0.00E+00	1.42E+00	2.09E+00
3.0	1.53E+00	1.05E+00	0.00E+00	0.00E+00	1.05E+00	1.40E+00
4.0	1.24E+00	8.51E-01	0.00E+00	0.00E+00	8.51E-01	1.16E+00
5.0	1.02E+00	6.77E-01	0.00E+00	0.00E+00	6.77E-01	9.55E-01
7.0	7.84E-01	5.18E-01	0.00E+00	0.00E+00	5.16E-01	7.87E-01
10.0	5.04E-01	3.35E-01	0.00E+00	0.00E+00	3.35E-01	5.75E-01

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 205759.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

*** Classification: Validate against Emergency Action Levels ***

Release Rates (Ci / sec)	
Particulate	4.98E-04 (0.0%)
Iodine	9.58E-03 (0.1%)
Noble Gas	6.44E+00 (99.8%)

Reviewed By: _____

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Containment Vent AXM 3 – General Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 21:00

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain> <HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = < 2 Hours | Containment Sprays: = OFF | Supp Pool Status: = Subcooled | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = N/A | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 1:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

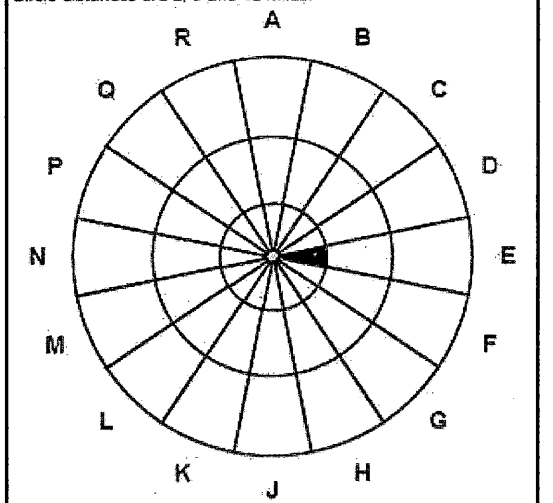
Monitor: AXM ch 3 Readings: 1.35E+05 cpm Flowrate: 6000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.21E+03	8.41E+02	1.16E+02	4.89E+01	1.01E+03	2.71E+03
0.5	9.92E+02	6.84E+02	8.84E+01	4.04E+01	8.13E+02	1.92E+03
0.7	6.80E+02	4.64E+02	5.88E+01	2.91E+01	5.52E+02	1.14E+03
1.0	4.32E+02	2.93E+02	3.72E+01	1.96E+01	3.50E+02	6.32E+02
1.5	2.39E+02	1.62E+02	2.14E+01	1.14E+01	1.94E+02	3.29E+02
2.0	1.88E+02	1.28E+02	1.41E+01	7.39E+00	1.50E+02	2.10E+02
3.0	1.54E+02	1.05E+02	8.95E+00	4.58E+00	1.19E+02	1.41E+02
4.0	1.24E+02	8.52E+01	7.74E+00	3.91E+00	9.69E+01	1.17E+02
5.0	1.02E+02	6.77E+01	6.43E+00	3.17E+00	7.73E+01	9.62E+01
7.0	7.88E+01	5.19E+01	5.28E+00	2.48E+00	5.96E+01	7.91E+01
10.0	5.04E+01	3.36E+01	3.70E+00	1.61E+00	3.89E+01	5.77E+01

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 210013.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



PAGs Exceeded in Designated Areas

Release Rates (Ci / sec)	
Particulate	4.99E-02 (0.0%)
Iodine	9.60E-01 (0.1%)
Noble Gas	6.46E+02 (99.8%)

***** Classification: General Emergency *****

Reviewed By: _____

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Containment Vent AXM 3 – Site Area Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 21:02

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain> <HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = < 2 Hours	Containment Sprays: = OFF	Supp Pool Status: = Subcooled	Safety Filters: = N/A
HVAC Filters: = Working	Aux Bldg HUT: = N/A	Turbine Bldg HUT: = N/A	RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad — OnSite Lower

Time After S/D (hh:mm): 1:00 — Wind: From 270° @ 4.4 mph

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A] — Stability Class: D

Precipitation: None

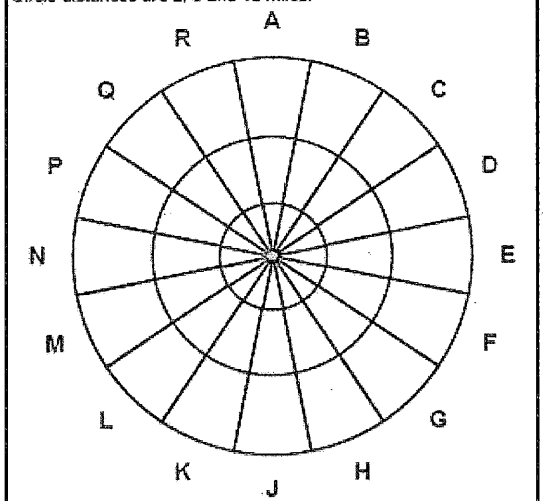
Monitor: AXM ch 3 Readings: 1.35E+04 cpm Flowrate: 6000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.21E+02	8.41E+01	1.16E+01	4.89E+00	1.01E+02	2.71E+02
0.5	9.92E+01	6.84E+01	8.84E+00	4.04E+00	8.13E+01	1.92E+02
0.7	6.80E+01	4.64E+01	5.88E+00	2.91E+00	5.52E+01	1.14E+02
1.0	4.32E+01	2.93E+01	3.72E+00	1.96E+00	3.50E+01	6.32E+01
1.5	2.39E+01	1.62E+01	2.14E+00	1.14E+00	1.94E+01	3.29E+01
2.0	1.88E+01	1.28E+01	1.41E+00	7.39E-01	1.50E+01	2.10E+01
3.0	1.54E+01	1.05E+01	8.95E-01	4.58E-01	1.19E+01	1.41E+01
4.0	1.24E+01	8.52E+00	7.74E-01	3.91E-01	9.69E+00	1.17E+01
5.0	1.02E+01	6.77E+00	6.43E-01	3.17E-01	7.73E+00	9.62E+00
7.0	7.88E+00	5.19E+00	5.28E-01	2.48E-01	5.96E+00	7.91E+00
10.0	5.04E+00	3.36E+00	3.70E-01	1.61E-01	3.89E+00	5.77E+00

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 210213.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Classification: Site Area Emergency

Release Rates (Ci / sec)	
Particulate	4.99E-03 (0.0%)
Iodine	9.60E-02 (0.1%)
Noble Gas	6.46E+01 (99.8%)

Reviewed By: _____

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Containment Vent AXM 3 – Alert

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 21:03

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain> <HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = < 2 Hours	Containment Sprays: = OFF	Supp Pool Status: = Subcooled	Safety Filters: = N/A
HVAC Filters: = Working	Aux Bldg HUT: = N/A	Turbine Bldg HUT: = N/A	RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad
Time After S/D (hh:mm): 1:00
Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower
Wind: From 270° @ 4.4 mph
Stability Class: D
Precipitation: None

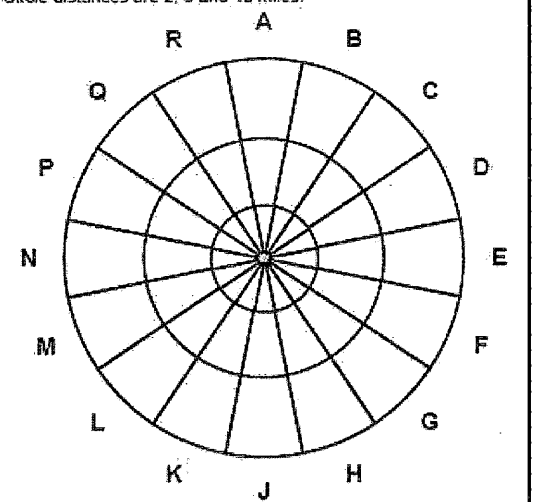
Monitor: AXM ch 3 Readings: 1.35E+03 cpm Flowrate: 6000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.21E+01	8.41E+00	1.16E+00	4.89E-01	1.01E+01	2.71E+01
0.5	9.92E+00	6.84E+00	8.84E-01	4.04E-01	8.13E+00	1.92E+01
0.7	6.80E+00	4.64E+00	5.86E-01	2.91E-01	5.52E+00	1.14E+01
1.0	4.32E+00	2.93E+00	3.72E-01	1.96E-01	3.50E+00	6.32E+00
1.5	2.39E+00	1.62E+00	2.14E-01	1.14E-01	1.94E+00	3.29E+00
2.0	1.88E+00	1.28E+00	1.41E-01	0.00E+00	1.42E+00	2.10E+00
3.0	1.54E+00	1.05E+00	0.00E+00	0.00E+00	1.05E+00	1.41E+00
4.0	1.24E+00	8.52E-01	0.00E+00	0.00E+00	8.52E-01	1.17E+00
5.0	1.02E+00	6.77E-01	0.00E+00	0.00E+00	6.77E-01	9.62E-01
7.0	7.88E-01	5.19E-01	0.00E+00	0.00E+00	5.19E-01	7.91E-01
10.0	5.04E-01	3.36E-01	0.00E+00	0.00E+00	3.36E-01	5.77E-01

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 210343.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Release Rates (Ci / sec)	
Particulate	4.99E-04 (0.0%)
Iodine	9.60E-03 (0.1%)
Noble Gas	6.46E+00 (99.8%)

* * * Classification: Validate against Emergency Action Levels * * *

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

Containment Vent AXM 4 – General Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 21:06

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain> <HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = < 2 Hours	Containment Sprays: = OFF	Supp Pool Status: = Subcooled	Safety Filters: = N/A
HVAC Filters: = Working	Aux Bldg HUT: = N/A	Turbine Bldg HUT: = N/A	RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad — OnSite Lower

Time After S/D (hh:mm): 1:00 — Wind: From 270° @ 4.4 mph

Release Duration (hh:mm): 1:00 — ETE (hh:mm): [N/A] — Stability Class: D

Precipitation: None

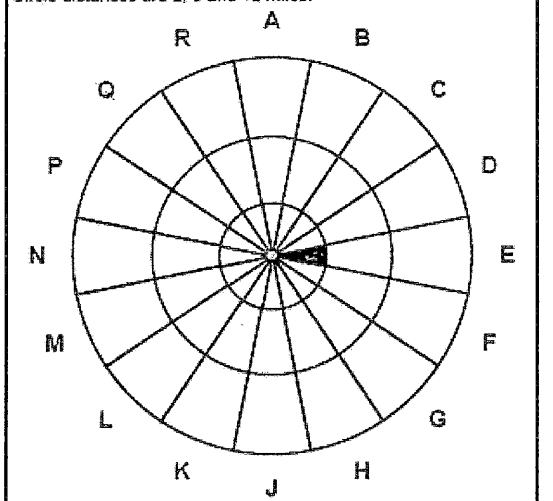
Monitor: AXM ch 4 — Readings: 7.48E+07 cpm — Flowrate: 6000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S/B	1.21E+03	8.39E+02	1.15E+02	4.88E+01	1.00E+03	2.71E+03
0.5	9.88E+02	6.84E+02	8.80E+01	4.02E+01	8.12E+02	1.90E+03
0.7	6.76E+02	4.64E+02	5.84E+01	2.90E+01	5.51E+02	1.14E+03
1.0	4.28E+02	2.92E+02	3.70E+01	1.96E+01	3.49E+02	6.28E+02
1.5	2.38E+02	1.61E+02	2.12E+01	1.14E+01	1.94E+02	3.27E+02
2.0	1.88E+02	1.28E+02	1.40E+01	7.35E+00	1.49E+02	2.09E+02
3.0	1.53E+02	1.05E+02	8.92E+00	4.56E+00	1.18E+02	1.40E+02
4.0	1.23E+02	8.49E+01	7.71E+00	3.89E+00	9.65E+01	1.16E+02
5.0	1.02E+02	6.76E+01	6.41E+00	3.17E+00	7.72E+01	9.55E+01
7.0	7.84E+01	5.18E+01	5.26E+00	2.47E+00	5.96E+01	7.87E+01
10.0	5.04E+01	3.35E+01	3.69E+00	1.60E+00	3.88E+01	5.75E+01

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 210601.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



PAGs Exceeded in Designated Areas

Release Rates (Ci / sec)	
Particulate	4.97E-02 (0.0%)
Iodine	9.57E-01 (0.1%)
Noble Gas	6.44E+02 (99.8%)

***** Classification: General Emergency *****

Reviewed By: _____

Page 1 of 3 — Grand Gulf / 2.0.1.0

Containment Vent AXM 4 – Site Area Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 21:07

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain> <HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = < 2 Hours	Containment Sprays: = OFF	Supp Pool Status: = Subcooled	Safety Filters: = N/A
HVAC Filters: = Working	Aux Bldg HUT: = N/A	Turbine Bldg HUT: = N/A	RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad — OnSite Lower

Time After S/D (hh:mm): 1:00 — Wind: From 270° @ 4.4 mph

Release Duration (hh:mm): 1:00 — ETE (hh:mm): [N/A] — Stability Class: D

Precipitation: None

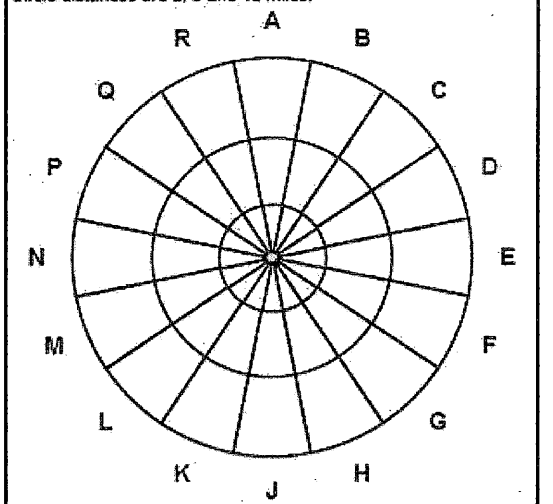
Monitor: AXM ch 4 — Readings: 7.48E+06 cpm — Flowrate: 6000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.21E+02	8.39E+01	1.15E+01	4.88E+00	1.00E+02	2.71E+02
0.5	9.88E+01	6.84E+01	8.80E+00	4.02E+00	8.12E+01	1.90E+02
0.7	6.76E+01	4.64E+01	5.84E+00	2.90E+00	5.51E+01	1.14E+02
1.0	4.28E+01	2.92E+01	3.70E+00	1.96E+00	3.49E+01	6.28E+01
1.5	2.38E+01	1.61E+01	2.12E+00	1.14E+00	1.94E+01	3.27E+01
2.0	1.88E+01	1.28E+01	1.40E+00	7.35E-01	1.49E+01	2.09E+01
3.0	1.53E+01	1.05E+01	8.92E-01	4.56E-01	1.18E+01	1.40E+01
4.0	1.23E+01	8.49E+00	7.71E-01	3.89E-01	9.65E+00	1.16E+01
5.0	1.02E+01	6.76E+00	6.41E-01	3.17E-01	7.72E+00	9.55E+00
7.0	7.84E+00	5.18E+00	5.26E-01	2.47E-01	5.96E+00	7.87E+00
10.0	5.04E+00	3.35E+00	3.69E-01	1.60E-01	3.88E+00	5.75E+00

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 210713.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Classification: Site Area Emergency

Release Rates (Ci / sec)	
Particulate	4.97E-03 (0.0%)
Iodine	9.57E-02 (0.1%)
Noble Gas	6.44E+01 (99.8%)

Reviewed By: _____

Page 1 of 3 — Grand Gulf / 2.0.1.0

Containment Vent AXM 4 – Alert

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 21:08

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS><Supp><Contain> <HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = < 2 Hours | Containment Sprays: = OFF | Supp Pool Status: = Subcooled | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = N/A | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad — OnSite Lower

Time After S/D (hh:mm): 1:00 — Wind: From 270° @ 4.4 mph

Release Duration (hh:mm): 1:00 — ETE (hh:mm): [N/A] — Stability Class: D

Precipitation: None

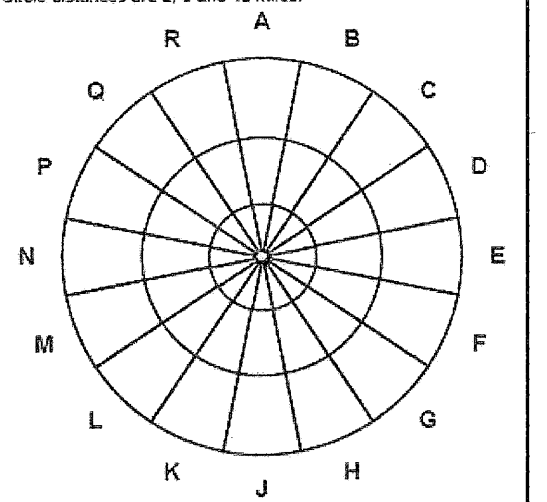
Monitor: AXM ch 4 — Readings: 7.48E+05 cpm — Flowrate: 6000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.21E+01	8.39E+00	1.15E+00	4.88E-01	1.00E+01	2.71E+01
0.5	9.88E+00	6.84E+00	8.80E-01	4.02E-01	8.12E+00	1.90E+01
0.7	6.76E+00	4.64E+00	5.84E-01	2.90E-01	5.51E+00	1.14E+01
1.0	4.28E+00	2.92E+00	3.70E-01	1.96E-01	3.49E+00	6.28E+00
1.5	2.38E+00	1.61E+00	2.12E-01	1.14E-01	1.94E+00	3.27E+00
2.0	1.88E+00	1.28E+00	1.40E-01	0.00E+00	1.42E+00	2.09E+00
3.0	1.53E+00	1.05E+00	0.00E+00	0.00E+00	1.05E+00	1.40E+00
4.0	1.23E+00	8.49E-01	0.00E+00	0.00E+00	8.49E-01	1.16E+00
5.0	1.02E+00	6.76E-01	0.00E+00	0.00E+00	6.76E-01	9.55E-01
7.0	7.84E-01	5.18E-01	0.00E+00	0.00E+00	5.18E-01	7.87E-01
10.0	5.04E-01	3.35E-01	0.00E+00	0.00E+00	3.35E-01	5.75E-01

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 210825.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Release Rates (Ci / sec)	
Particulate	4.97E-04 (0.0%)
Iodine	9.57E-03 (0.1%)
Noble Gas	6.44E+00 (99.8%)

*** Classification: Validate against Emergency Action Levels ***

Reviewed By: _____

Page 1 of 3 — Grand Gulf / 2.0.1.0

Radwaste Vent SPING 7– General Emergency

Dose Assessment

Grand Gulf Thursday, April 6, 2017 08:01

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <Aux Bldg> <Rad Waste> <HVAC Filters> <Env> PRF: 3.20E-02

Containment HUT: = N/A | Containment Sprays: = N/A | Supp Pool Status: = N/A | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = < 2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = < 2 Hours

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 0:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

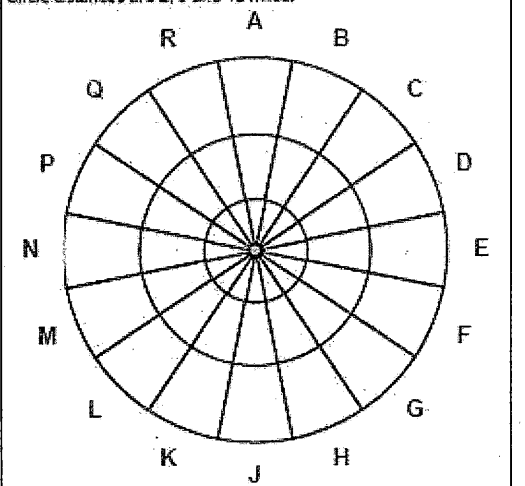
Monitor: SPING ch 7 Readings: 1.72E+04 cpm Flowrate: 52495 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.71E+02	1.12E+02	1.99E+02	8.34E+01	3.94E+02	5.00E+03
0.5	1.35E+02	8.84E+01	1.40E+02	6.04E+01	2.89E+02	3.51E+03
0.7	8.88E+01	5.72E+01	8.36E+01	3.72E+01	1.78E+02	2.08E+03
1.0	5.20E+01	3.31E+01	4.84E+01	2.11E+01	1.01E+02	1.14E+03
1.5	3.06E+01	1.97E+01	2.45E+01	1.10E+01	5.52E+01	5.96E+02
2.0	9.68E+00	5.90E+00	1.64E+01	7.02E+00	2.92E+01	4.00E+02
3.0	1.78E+01	1.16E+01	1.17E+01	5.04E+00	2.84E+01	2.87E+02
4.0	1.28E+01	8.34E+00	9.38E+00	3.90E+00	2.16E+01	2.29E+02
5.0	1.09E+01	7.01E+00	8.65E+00	3.48E+00	1.91E+01	2.12E+02
7.0	8.96E+00	4.34E+00	6.42E+00	2.42E+00	1.32E+01	1.58E+02
10.0	3.84E+00	2.43E+00	4.49E+00	1.56E+00	8.49E+00	1.12E+02

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 04062017 080114.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Classification: Site Area Emergency

Release Rates (Ci / sec)	
Particulate	9.73E-02 (0.2%)
Iodine	2.39E+00 (4.6%)
Noble Gas	4.90E+01 (95.2%)

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

Radwaste Vent SPING 7 – Site Area Emergency

Dose Assessment

Grand Gulf **Thursday, April 6, 2017 08:01**

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <Aux Bldg> <Rad Waste> <HVAC Filters> <Env> PRF: 3.20E-02

Containment HUT: = N/A | Containment Sprays: = N/A | Supp Pool Status: = N/A | Safety Filters: = N/A

HVAC Filters = Working | Aux Bldg HUT: = < 2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = < 2 Hours

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 0:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

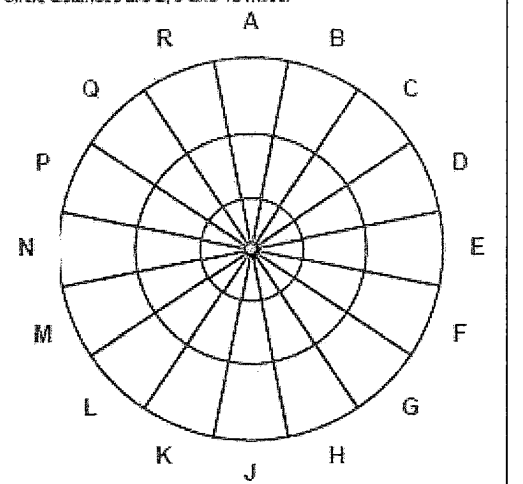
Monitor: SPING ch 7 Readings: 1.72E+03 cpm Flowrate: 52495 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DOE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DOE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.71E+01	1.12E+01	1.99E+01	8.34E+00	3.94E+01	5.00E+02
0.5	1.35E+01	8.84E+00	1.40E+01	6.04E+00	2.89E+01	3.51E+02
0.7	8.88E+00	5.72E+00	8.36E+00	3.72E+00	1.78E+01	2.08E+02
1.0	5.20E+00	3.31E+00	4.84E+00	2.11E+00	1.01E+01	1.14E+02
1.5	3.06E+00	1.97E+00	2.45E+00	1.10E+00	5.52E+00	5.96E+01
2.0	9.89E-01	5.80E-01	1.64E+00	7.02E-01	2.92E+00	4.00E+01
3.0	1.76E+00	1.16E+00	1.17E+00	5.04E-01	2.84E+00	2.87E+01
4.0	1.28E+00	8.34E-01	9.38E-01	3.90E-01	2.16E+00	2.29E+01
5.0	1.09E+00	7.01E-01	8.65E-01	3.48E-01	1.91E+00	2.12E+01
7.0	8.90E-01	4.34E-01	6.42E-01	2.42E-01	1.32E+00	1.58E+01
10.0	3.04E-01	2.43E-01	4.49E-01	1.56E-01	8.49E-01	1.12E+01

Assessment Data Results Saved to File.
Grand Gulf 10Miles Monitored Release 04062017 080142.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

* * * Classification: Validate against Emergency Action Levels * * *

Release Rates (Ci / sec)	
Particulate	9.73E-03 (0.2%)
Iodine	2.39E-01 (4.8%)
Noble Gas	4.90E+00 (95.2%)

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

Radwaste Vent SPING 7 – Alert

Dose Assessment

Grand Gulf **Thursday, April 6, 2017 08:02**

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <Aux Bldg> <Rad Waste> <HVAC Filters> <Env> PRF: 3.20E-02

Containment HUT: = N/A | Containment Sprays: = N/A | Supp Pool Status: = N/A | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = < 2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = < 2 Hours

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 0:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

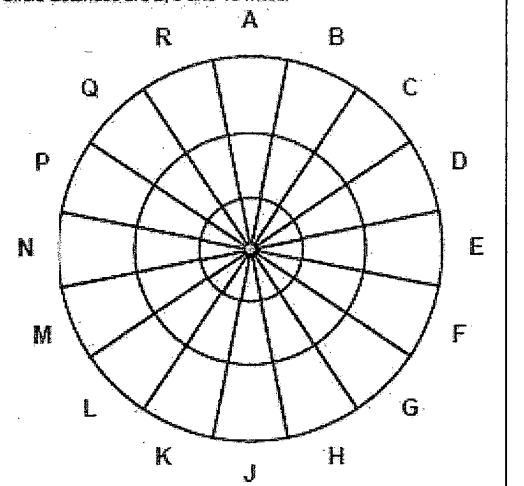
Monitor: SPING ch 7 Readings: 1.72E+02 cpm Flowrate: 52495 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.71E+00	1.12E+00	1.99E+00	8.34E-01	3.94E+00	5.00E+01
0.5	1.35E+00	8.84E-01	1.40E+00	6.04E-01	2.89E+00	3.51E+01
0.7	8.88E-01	5.72E-01	8.38E-01	3.72E-01	1.78E+00	2.08E+01
1.0	5.20E-01	3.31E-01	4.64E-01	2.11E-01	1.01E+00	1.14E+01
1.5	3.08E-01	1.97E-01	2.45E-01	1.10E-01	5.52E-01	5.96E+00
2.0	0.00E+00	0.00E+00	1.64E-01	0.00E+00	1.64E-01	4.00E+00
3.0	1.78E-01	1.18E-01	1.17E-01	0.00E+00	2.33E-01	2.87E+00
4.0	1.28E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.29E+00
5.0	1.08E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.12E+00
7.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.58E+00
10.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.12E+00

Assessment Data Results Saved to File.
Grand Gulf 10Miles Monitored Release 04062017 080251.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

***** Classification: Validate against Emergency Action Levels *****

Release Rates (Ci / sec)	
Particulate	9.73E-04 (0.2%)
Iodine	2.39E-02 (4.8%)
Noble Gas	4.90E-01 (95.2%)

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

Radwaste Vent AXM 3– General Emergency

Dose Assessment

Grand Gulf **Thursday, April 6, 2017 07:55**

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <Aux Bldg> <Rad Waste> <HVAC Filters> <Env> PRF: 3.20E-02

Containment HUT: = N/A | Containment Sprays: = N/A | Supp Pool Status: = N/A | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = < 2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = < 2 Hours

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 0:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

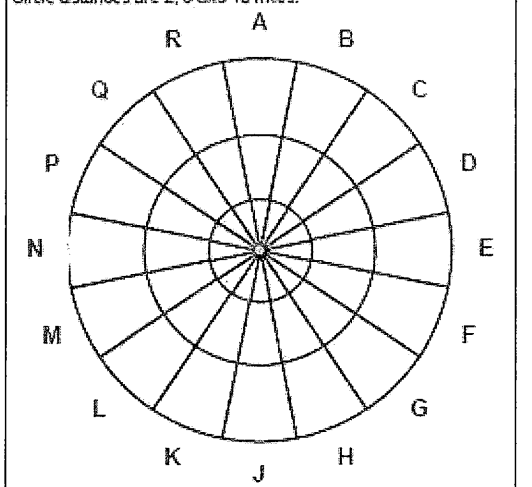
Monitor: AXM ch 3 Readings: 1.17E+03 cpm Flowrate: 52495 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.71E+02	1.12E+02	1.99E+02	8.34E+01	3.94E+02	5.00E+03
0.5	1.35E+02	8.84E+01	1.40E+02	6.04E+01	2.89E+02	3.51E+03
0.7	9.88E+01	5.72E+01	8.36E+01	3.72E+01	1.78E+02	2.08E+03
1.0	5.20E+01	3.31E+01	4.84E+01	2.11E+01	1.01E+02	1.14E+03
1.5	3.06E+01	1.97E+01	2.45E+01	1.10E+01	5.62E+01	5.98E+02
2.0	9.68E+00	5.80E+00	1.64E+01	7.02E+00	2.92E+01	4.00E+02
3.0	1.78E+01	1.16E+01	1.17E+01	5.04E+00	2.84E+01	2.87E+02
4.0	1.28E+01	8.34E+00	9.38E+00	3.90E+00	2.16E+01	2.29E+02
5.0	1.09E+01	7.01E+00	8.65E+00	3.48E+00	1.91E+01	2.12E+02
7.0	6.96E+00	4.34E+00	6.42E+00	2.42E+00	1.32E+01	1.58E+02
10.0	3.64E+00	2.43E+00	4.49E+00	1.56E+00	8.49E+00	1.12E+02

Assessment Data Results Saved to File.
Grand Gulf 10Miles Monitored Release 04062017 075525.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Classification: Site/Area Emergency

Release Rates (Ci / sec)	
Particulate	9.72E-02 (0.2%)
Iodine	2.38E+00 (4.6%)
Noble Gas	4.90E+01 (95.2%)

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

Radwaste Vent AXM 3 – Site Area Emergency

Dose Assessment

Grand Gulf **Thursday, April 6, 2017 07:57**

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <Aux Bldg> <Rad Waste> <HVAC Filters> <Env> PRF: 3.20E-02

Containment HUT: = N/A | Containment Sprays: = N/A | Supp Pool Status: = N/A | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = < 2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = < 2 Hours

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 0:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

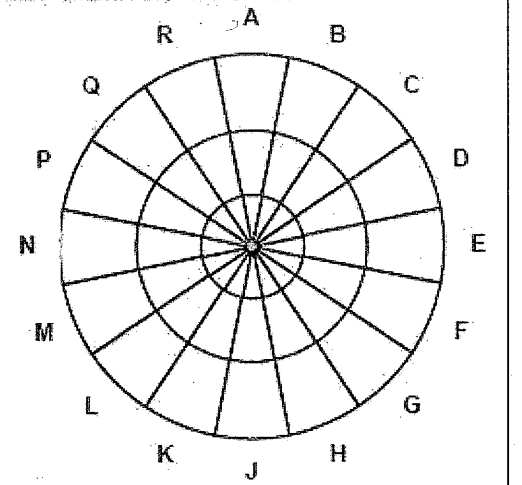
Monitor: AXM ch 3 Readings: 1.17E+02 cpm Flowrate: 52495 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.71E+01	1.12E+01	1.99E+01	8.34E+00	3.94E+01	5.00E+02
0.5	1.35E+01	8.84E+00	1.40E+01	6.04E+00	2.89E+01	3.51E+02
0.7	8.88E+00	5.72E+00	8.36E+00	3.72E+00	1.78E+01	2.08E+02
1.0	5.20E+00	3.31E+00	4.84E+00	2.11E+00	1.01E+01	1.14E+02
1.5	3.08E+00	1.97E+00	2.45E+00	1.10E+00	5.52E+00	5.98E+01
2.0	9.68E-01	5.80E-01	1.64E+00	7.02E-01	2.92E+00	4.00E+01
3.0	1.78E+00	1.16E+00	1.17E+00	5.04E-01	2.84E+00	2.87E+01
4.0	1.28E+00	8.34E-01	9.38E-01	3.90E-01	2.16E+00	2.29E+01
5.0	1.09E+00	7.01E-01	8.65E-01	3.48E-01	1.91E+00	2.12E+01
7.0	6.98E-01	4.34E-01	6.42E-01	2.42E-01	1.32E+00	1.58E+01
10.0	3.84E-01	2.43E-01	4.49E-01	1.56E-01	8.49E-01	1.12E+01

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitered Release 04062017 075738.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

***** Classification: Validate against Emergency Action Levels *****

Release Rates (Ci / sec)	
Particulate	9.72E-03 (0.2%)
Iodine	2.38E-01 (4.8%)
Noble Gas	4.90E+00 (95.2%)

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

Radwaste Vent AXM 3 – Alert

Dose Assessment

Grand Gulf **Thursday, April 6, 2017 07:56**

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <Aux Bldg> <Rad Waste> <HVAC Filters> <Env> PRF: 3.20E-02

Containment HUT: = N/A | Containment Sprays: = N/A | Supp Pool Status: = N/A | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = < 2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = < 2 Hours

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 0:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

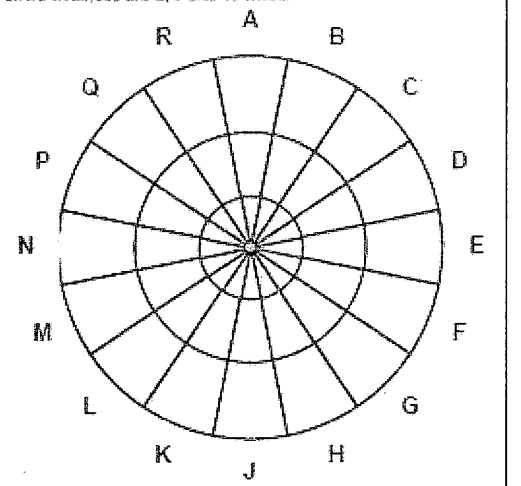
Monitor: AXM ch 3 Readings: 1.17E+01 cpm Flowrate: 52495 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.71E+00	1.12E+00	1.99E+00	8.34E-01	3.94E+00	5.00E+01
0.5	1.35E+00	8.84E-01	1.40E+00	6.04E-01	2.89E+00	3.51E+01
0.7	8.98E-01	5.72E-01	8.36E-01	3.72E-01	1.78E+00	2.08E+01
1.0	5.20E-01	3.31E-01	4.84E-01	2.11E-01	1.01E+00	1.14E+01
1.5	3.08E-01	1.97E-01	2.45E-01	1.10E-01	5.52E-01	5.98E+00
2.0	0.00E+00	0.00E+00	1.64E-01	0.00E+00	1.64E-01	4.00E+00
3.0	1.78E-01	1.18E-01	1.17E-01	0.00E+00	2.33E-01	2.87E+00
4.0	1.28E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.29E+00
5.0	1.09E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.12E+00
7.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.58E+00
10.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.12E+00

Assessment Data Results Saved to File.
Grand Gulf 10Miles Monitored Release 04062017 075625.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

*** Classification: Validate against Emergency Action Levels ***

Release Rates (Ci / sec)	
Particulate	9.72E-04 (0.2%)
Iodine	2.38E-02 (4.8%)
Noble Gas	4.90E-01 (95.2%)

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

Radwaste Vent AXM 4– General Emergency

Dose Assessment

Grand Gulf **Thursday, April 6, 2017 07:58**

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <Aux Bldg> <Rad Waste> <HVAC Filters> <Env> PRF: 3.20E-02

Containment HUT: = N/A	Containment Sprays: = N/A	Supp Pool Status: = N/A	Safety Filters: = N/A
HVAC Filters: = Working	Aux Bldg HUT: = < 2 Hours	Turbine Bldg HUT: = N/A	RadWaste Bldg HUT: = < 2 Hours

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 0:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

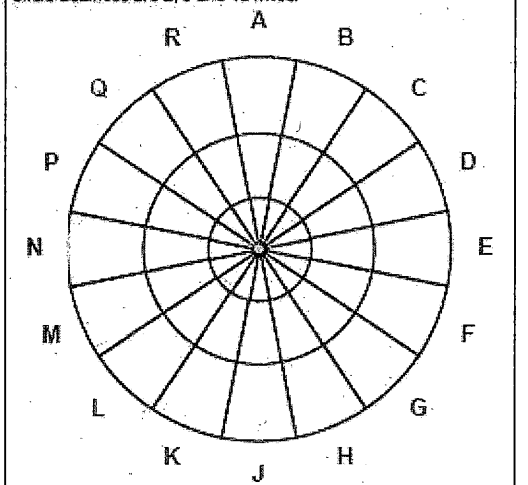
Monitor: AXM ch 4 Readings: 6.50E+05 cpm Flowrate: 52495 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.70E+02	1.12E+02	1.99E+02	8.34E+01	3.94E+02	5.00E+03
0.5	1.35E+02	8.80E+01	1.40E+02	6.04E+01	2.89E+02	3.51E+03
0.7	8.86E+01	5.72E+01	8.36E+01	3.71E+01	1.78E+02	2.08E+03
1.0	5.20E+01	3.30E+01	4.84E+01	2.11E+01	1.01E+02	1.14E+03
1.5	3.06E+01	1.97E+01	2.45E+01	1.10E+01	5.52E+01	5.96E+02
2.0	9.68E+00	5.80E+00	1.64E+01	7.02E+00	2.92E+01	4.00E+02
3.0	1.78E+01	1.16E+01	1.17E+01	5.03E+00	2.83E+01	2.87E+02
4.0	1.28E+01	8.34E+00	9.38E+00	3.90E+00	2.16E+01	2.29E+02
5.0	1.09E+01	7.01E+00	8.65E+00	3.48E+00	1.91E+01	2.12E+02
7.0	6.96E+00	4.34E+00	6.42E+00	2.42E+00	1.32E+01	1.58E+02
10.0	3.64E+00	2.43E+00	4.49E+00	1.56E+00	8.49E+00	1.12E+02

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 04062017 075856.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Classification: Site Area Emergency

Release Rates (Ci / sec)	
Particulate	8.72E-02 (0.2%)
Iodine	2.38E+00 (4.6%)
Noble Gas	4.89E+01 (95.2%)

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

Radwaste Vent AXM 4 – Site Area Emergency

Dose Assessment

Grand Gulf — Thursday, April 6, 2017 07:59

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <Aux Bldg> <Rad Waste> <HVAC Filters> <Env> — PRF: 3.20E-02

Containment HUT: = N/A — Containment Sprays: = N/A — Supp Pool Status: = N/A — Safety Filters: = N/A

HVAC Filters = Working — Aux Bldg HUT: = < 2 Hours — Turbine Bldg HUT: = N/A — RadWaste Bldg HUT: = < 2 Hours

Source Term: Reactor Core Accident - Clad — OnSite Lower

Time After S/D (hh:mm): 0:00 — Wind: From 270° @ 4.4 mph

Release Duration (hh:mm): 1:00 — ETE (hh:mm): [N/A] — Stability Class: D

Precipitation: None

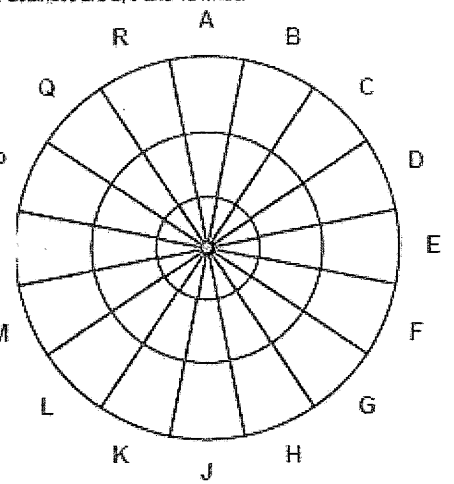
Monitor: AXM ch 4 — Readings: 6.50E+04 cpm — Flowrate: 52495 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.70E+01	1.12E+01	1.99E+01	8.34E+00	3.94E+01	5.00E+02
0.5	1.35E+01	8.80E+00	1.40E+01	6.04E+00	2.89E+01	3.51E+02
0.7	8.88E+00	5.72E+00	8.36E+00	3.71E+00	1.78E+01	2.08E+02
1.0	5.20E+00	3.30E+00	4.84E+00	2.11E+00	1.01E+01	1.14E+02
1.5	3.08E+00	1.97E+00	2.45E+00	1.10E+00	5.52E+00	5.96E+01
2.0	9.88E-01	5.90E-01	1.64E+00	7.02E-01	2.92E+00	4.00E+01
3.0	1.78E+00	1.16E+00	1.17E+00	5.03E-01	2.83E+00	2.87E+01
4.0	1.28E+00	8.34E-01	9.38E-01	3.90E-01	2.16E+00	2.29E+01
5.0	1.09E+00	7.01E-01	8.65E-01	3.48E-01	1.91E+00	2.12E+01
7.0	8.96E-01	4.34E-01	6.42E-01	2.42E-01	1.32E+00	1.58E+01
10.0	3.84E-01	2.43E-01	4.49E-01	1.56E-01	8.49E-01	1.12E+01

Assessment Data Results Saved to File.
Grand Gulf 10Miles Monitored Release 04062017 075921.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

*** Classification: Validate against Emergency Action Levels ***

Release Rates (Ci / sec)	
Particulate	9.72E-03 (0.2%)
Iodine	2.38E-01 (4.8%)
Noble Gas	4.89E+00 (95.2%)

Reviewed By: _____

Page 1 of 3 — Grand Gulf / 2.0.1.0

Radwaste Vent AXM 4 – Alert

Dose Assessment

Grand Gulf **Thursday, April 6, 2017 07:59**

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <Aux Bldg> <Rad Waste> <HVAC Filters> <Env> PRF: 3.20E-02

Containment HUT: = N/A | Containment Sprays: = N/A | Supp Pool Status: = N/A | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = < 2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = < 2 Hours

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 0:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

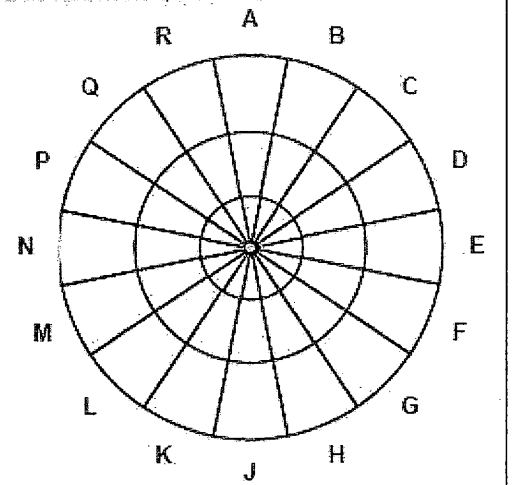
Monitor: AXM ch 4 Readings: 6.50E+03 cpm Flowrate: 52495 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.70E+00	1.12E+00	1.99E+00	8.34E-01	3.94E+00	5.00E+01
0.5	1.35E+00	8.80E-01	1.40E+00	6.04E-01	2.89E+00	3.51E+01
0.7	8.88E-01	5.72E-01	8.38E-01	3.71E-01	1.78E+00	2.08E+01
1.0	5.20E-01	3.30E-01	4.84E-01	2.11E-01	1.01E+00	1.14E+01
1.5	3.08E-01	1.97E-01	2.45E-01	1.10E-01	5.52E-01	5.96E+00
2.0	0.00E+00	0.00E+00	1.64E-01	0.00E+00	1.64E-01	4.00E+00
3.0	1.79E-01	1.18E-01	1.17E-01	0.00E+00	2.33E-01	2.87E+00
4.0	1.28E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.29E+00
5.0	1.09E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.12E+00
7.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.58E+00
10.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.12E+00

Assessment Data Results Saved to File.
Grand Gulf 10Miles Monitored Release 04062017 075949.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

*** Classification: Validate against Emergency Action Levels ***

Release Rates (Ci / sec)	
Particulate	9.72E-04 (0.2%)
Iodine	2.38E-02 (4.8%)
Noble Gas	4.89E-01 (95.2%)

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

Turbine Building Vent SPING 7 – General Emergency

Dose Assessment

Grand Gulf **Wednesday, April 5, 2017 14:40**

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <TB Bldg> <HVAC Filters> <Env> PRF: 8.00E-02

Containment HUT: = N/A | Containment Sprays: = N/A | Supp Pool Status: = N/A | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = N/A | Turbine Bldg HUT: = <2 Hours | RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 1:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

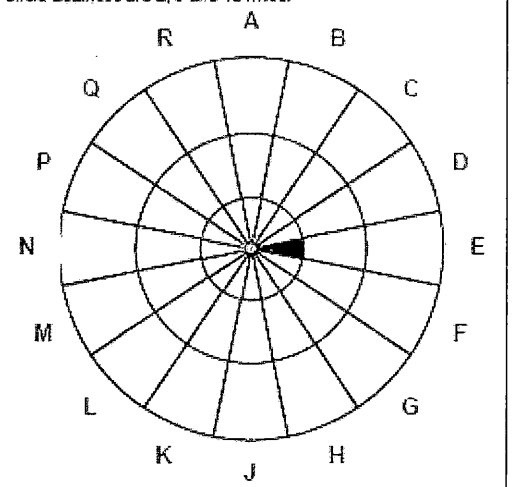
Monitor: SPING ch 7 Readings: 4.26E+04 cpm Flowrate: 5000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	3.78E+01	2.24E+01	1.98E+02	7.70E+01	2.97E+02	5.02E+03
0.5	3.05E+01	1.84E+01	1.41E+02	5.48E+01	2.14E+02	3.58E+03
0.7	2.06E+01	1.26E+01	8.55E+01	3.31E+01	1.31E+02	2.16E+03
1.0	1.29E+01	7.86E+00	4.80E+01	1.85E+01	7.44E+01	1.21E+03
1.5	7.12E+00	4.44E+00	2.51E+01	9.60E+00	3.92E+01	6.32E+02
2.0	5.16E+00	3.30E+00	1.57E+01	5.91E+00	2.49E+01	3.94E+02
3.0	3.95E+00	2.63E+00	9.76E+00	3.62E+00	1.60E+01	2.46E+02
4.0	3.31E+00	2.11E+00	6.85E+00	2.94E+00	1.31E+01	2.03E+02
5.0	2.83E+00	1.79E+00	7.03E+00	2.53E+00	1.14E+01	1.77E+02
7.0	2.01E+00	1.26E+00	5.30E+00	1.86E+00	8.41E+00	1.33E+02
10.0	1.35E+00	8.79E-01	4.10E+00	1.39E+00	6.37E+00	1.03E+02

Assessment Data Results Saved to File.
Grand Gulf 10Miles Monitored Release 04052017 144025.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



PAGs Exceeded in Designated Areas

***** Classification: General Emergency *****

Release Rates (Ci / sec)	
Particulate	8.92E-02 (0.7%)
Iodine	1.72E+00 (12.8%)
Noble Gas	1.16E+01 (86.5%)

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

Turbine Building Vent SPING 7 – Site Area Emergency

Dose Assessment

Grand Gulf Wednesday, April 5, 2017 14:40

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <TB Bldg> <HVAC Filters> <Env> PRF: 8.00E-02

Containment HUT: = N/A | Containment Sprays: = N/A | Supp Pool Status: = N/A | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = N/A | Turbine Bldg HUT: = <2 Hours | RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 1:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

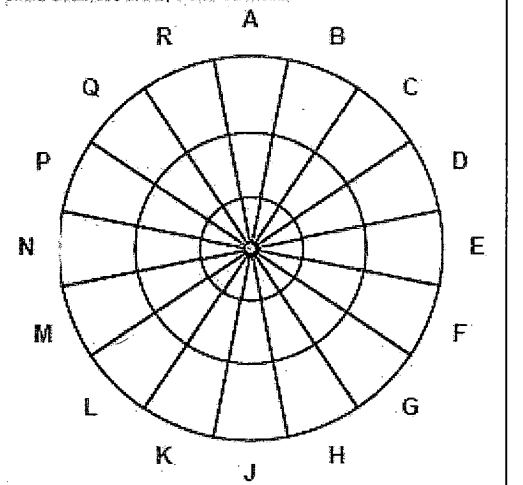
Monitor: SPING ch 7 Readings: 4.26E+03 cpm Flowrate: 5000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	3.78E+00	2.24E+00	1.98E+01	7.70E+00	2.97E+01	5.02E+02
0.5	3.05E+00	1.84E+00	1.41E+01	5.48E+00	2.14E+01	3.58E+02
0.7	2.06E+00	1.26E+00	8.55E+00	3.31E+00	1.31E+01	2.16E+02
1.0	1.29E+00	7.98E-01	4.80E+00	1.85E+00	7.44E+00	1.21E+02
1.5	7.12E-01	4.44E-01	2.51E+00	9.60E-01	3.92E+00	6.32E+01
2.0	5.16E-01	3.30E-01	1.57E+00	5.91E-01	2.49E+00	3.94E+01
3.0	3.95E-01	2.63E-01	9.76E-01	3.62E-01	1.60E+00	2.46E+01
4.0	3.31E-01	2.11E-01	8.05E-01	2.94E-01	1.31E+00	2.03E+01
5.0	2.83E-01	1.79E-01	7.03E-01	2.53E-01	1.14E+00	1.77E+01
7.0	2.01E-01	1.28E-01	5.30E-01	1.86E-01	8.41E-01	1.33E+01
10.0	1.35E-01	0.00E+00	4.10E-01	1.39E-01	5.49E-01	1.03E+01

Assessment Data Results Saved to File.
Grand Gulf 10Miles Monitored Release 04052017 144058.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Classification: Site Area Emergency

Release Rates (Ci / sec)	
Particulate	8.92E-03 (0.7%)
Iodine	1.72E-01 (12.8%)
Noble Gas	1.16E+00 (86.5%)

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

Turbine Building Vent SPING 7 – Alert

Dose Assessment

Grand Gulf — Wednesday, April 5, 2017 14:41

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <TB Bldg> <HVAC Filters> <Env> — PRF: 8.00E-02

Containment HUT: = N/A	Containment Sprays: = N/A	Supp Pool Status: = N/A	Safety Filters: = N/A
HVAC Filters: = Working	Aux Bldg HUT: = N/A	Turbine Bldg HUT: = <2 Hours	RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad — OnSite Lower

Time After S/D (hh:mm): 1:00 — Wind: From 270° @ 4.4 mph

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A] — Stability Class: D

Precipitation: None

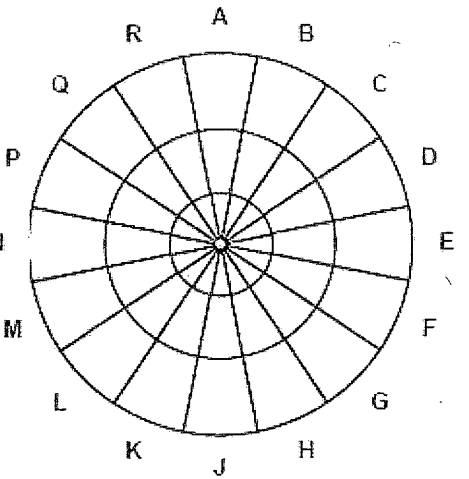
Monitor: SPING ch 7 Readings: 4.26E+02 cpm Flowrate: 5000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	3.78E-01	2.24E-01	1.98E+00	7.70E-01	2.97E+00	5.02E+01
0.5	3.05E-01	1.84E-01	1.41E+00	5.48E-01	2.14E+00	3.58E+01
0.7	2.06E-01	1.26E-01	8.58E-01	3.31E-01	1.31E+00	2.16E+01
1.0	1.29E-01	0.00E+00	4.80E-01	1.85E-01	6.65E-01	1.21E+01
1.5	0.00E+00	0.00E+00	2.51E-01	0.00E+00	2.51E-01	6.32E+00
2.0	0.00E+00	0.00E+00	1.57E-01	0.00E+00	1.57E-01	3.94E+00
3.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.46E+00
4.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.03E+00
5.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.77E+00
7.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.33E+00
10.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.03E+00

Assessment Data Results Saved to File.
Grand Gulf 10Miles Monitored Release 04052017 144124.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

*** Classification: Validate against Emergency Action Levels ***

Release Rates (Ci / sec)	
Particulate	8.92E-04 (0.7%)
Iodine	1.72E-02 (12.8%)
Noble Gas	1.16E-01 (86.5%)

Reviewed By: _____

Page 1 of 3 — Grand Gulf / 2.0.1.0

Turbine Building Vent AXM 3 – General Emergency

Dose Assessment

Grand Gulf **Wednesday, April 5, 2017 14:21**

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <TB Bldg> <HVAC Filters> <Env> PRF: 8.00E-02

Containment HUT: = N/A	Containment Sprays: = N/A	Supp Pool Status: = N/A	Safety Filters: = N/A
HVAC Filters: = Working	Aux Bldg HUT: = N/A	Turbine Bldg HUT: = <2 Hours	RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 1:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

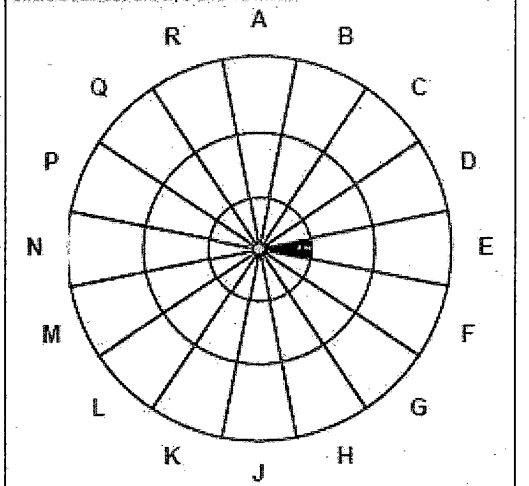
Monitor: AXM ch 3 Readings: 2.90E+03 cpm Flowrate: 5000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	3.78E+01	2.24E+01	1.98E+02	7.70E+01	2.97E+02	5.02E+03
0.5	3.05E+01	1.84E+01	1.41E+02	5.48E+01	2.14E+02	3.58E+03
0.7	2.06E+01	1.26E+01	8.56E+01	3.31E+01	1.31E+02	2.16E+03
1.0	1.29E+01	7.98E+00	4.80E+01	1.85E+01	7.44E+01	1.21E+03
1.5	7.12E+00	4.44E+00	2.51E+01	9.60E+00	3.92E+01	6.32E+02
2.0	5.16E+00	3.30E+00	1.57E+01	5.91E+00	2.49E+01	3.94E+02
3.0	3.95E+00	2.63E+00	9.76E+00	3.62E+00	1.60E+01	2.46E+02
4.0	3.31E+00	2.11E+00	8.05E+00	2.94E+00	1.31E+01	2.03E+02
5.0	2.83E+00	1.79E+00	7.03E+00	2.53E+00	1.14E+01	1.77E+02
7.0	2.01E+00	1.28E+00	5.30E+00	1.86E+00	8.41E+00	1.33E+02
10.0	1.35E+00	8.79E-01	4.10E+00	1.39E+00	6.37E+00	1.03E+02

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 04052017 142107.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



PAGs Exceeded in Designated Areas

Release Rates (Ci / sec)	
Particulate	8.93E-02 (0.7%)
Iodine	1.72E+00 (12.8%)
Noble Gas	1.16E+01 (86.5%)

*** Classification: General Emergency ***

Reviewed By: _____

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Turbine Building Vent AXM 3 – Site Area Emergency

Dose Assessment

Grand Gulf Thursday, April 6, 2017 08:07

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <TB Bldg> <HVAC Filters> <Env> PRF: 8.00E-02

Containment HUT: = N/A	Containment Sprays: = N/A	Supp Pool Status: = N/A	Safety Filters: = N/A
HVAC Filters: = Working	Aux Bldg HUT: = N/A	Turbine Bldg HUT: = <2 Hours	RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 1:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

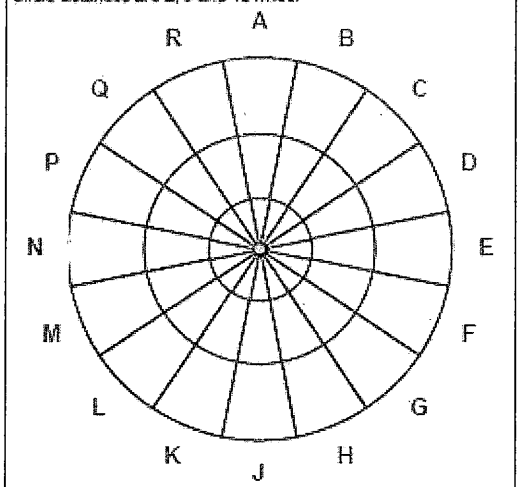
Monitor: AXM ch 3 Readings: 2.90E+02 cpm Flowrate: 5000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	3.78E+00	2.24E+00	1.98E+01	7.70E+00	2.97E+01	5.02E+02
0.5	3.05E+00	1.84E+00	1.41E+01	5.48E+00	2.14E+01	3.58E+02
0.7	2.06E+00	1.26E+00	8.56E+00	3.31E+00	1.31E+01	2.16E+02
1.0	1.29E+00	7.96E-01	4.80E+00	1.85E+00	7.44E+00	1.21E+02
1.5	7.12E-01	4.44E-01	2.51E+00	9.60E-01	3.92E+00	6.32E+01
2.0	5.16E-01	3.30E-01	1.57E+00	5.91E-01	2.49E+00	3.84E+01
3.0	3.95E-01	2.63E-01	9.76E-01	3.62E-01	1.60E+00	2.46E+01
4.0	3.31E-01	2.11E-01	8.05E-01	2.94E-01	1.31E+00	2.03E+01
5.0	2.83E-01	1.79E-01	7.03E-01	2.53E-01	1.14E+00	1.77E+01
7.0	2.01E-01	1.28E-01	5.30E-01	1.88E-01	8.41E-01	1.33E+01
10.0	1.35E-01	0.00E+00	4.10E-01	1.39E-01	5.49E-01	1.03E+01

Assessment Data Results Saved to File.
Grand Gulf 10Miles Monitored Release 04062017 080756.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Classification: Site Area Emergency

Release Rates (Ci / sec)	
Particulate	8.93E-03 (0.7%)
Iodine	1.72E-01 (12.8%)
Noble Gas	1.16E+00 (86.5%)

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

Turbine Building Vent AXM 3 – Alert

Dose Assessment

Grand Gulf **Wednesday, April 5, 2017 14:22**

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <TB Bldg> <HVAC Filters> <Env> PRF: 8.00E-02

Containment HUT: = N/A | Containment Sprays: = N/A | Supp Pool Status: = N/A | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = N/A | Turbine Bldg HUT: = <2 Hours | RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 1:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

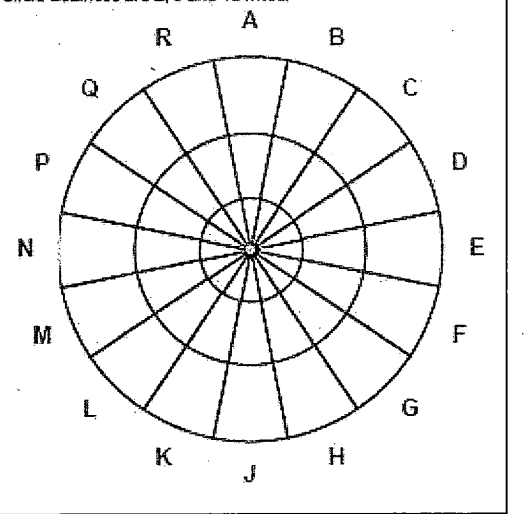
Monitor: AXM ch 3 Readings: 2.90E+01 cpm Flowrate: 5000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	3.78E-01	2.24E-01	1.98E+00	7.70E-01	2.97E+00	5.02E+01
0.5	3.05E-01	1.84E-01	1.41E+00	5.48E-01	2.14E+00	3.58E+01
0.7	2.06E-01	1.20E-01	8.56E-01	3.31E-01	1.31E+00	2.16E+01
1.0	1.29E-01	0.00E+00	4.80E-01	1.85E-01	6.65E-01	1.21E+01
1.5	0.00E+00	0.00E+00	2.51E-01	0.00E+00	2.51E-01	6.32E+00
2.0	0.00E+00	0.00E+00	1.57E-01	0.00E+00	1.57E-01	3.94E+00
3.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.46E+00
4.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.03E+00
5.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.77E+00
7.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.33E+00
10.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.03E+00

Assessment Data Results Saved to File.
Grand Gulf 10Miles Monitored Release 04052017 142237.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

*** Classification: Validate against Emergency Action Levels ***

Release Rates (Ci / sec)	
Particulate	8.93E-04 (0.7%)
Iodine	1.72E-02 (12.8%)
Noble Gas	1.16E-01 (86.5%)

Reviewed By: _____

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Turbine Building Vent AXM 4 – General Emergency

Dose Assessment

Grand Gulf — Wednesday, April 5, 2017 14:17

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <TB Bldg> <HVAC Filters> <Env> — PRF: 8.00E-02

Containment HUT: = N/A — Containment Sprays: = N/A — Supp Pool Status: = N/A — Safety Filters: = N/A

HVAC Filters: = Working — Aux Bldg HUT: = N/A — Turbine Bldg HUT: = <2 Hours — RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad — OnSite Lower

Time After S/D (hh:mm): 1:00 — Wind: From 270° @ 4.4 mph

Release Duration (hh:mm): 1:00 — ETE (hh:mm): [N/A] — Stability Class: D

Precipitation: None

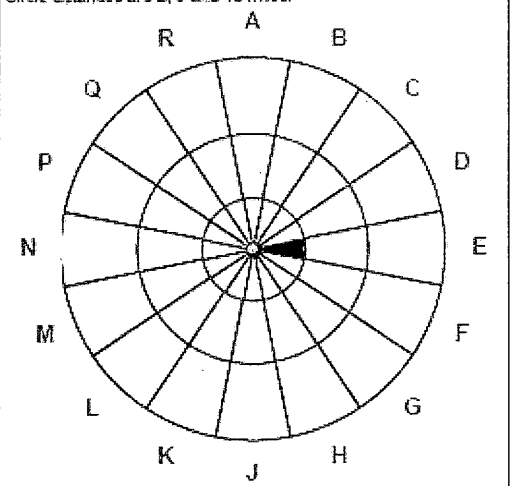
Monitor: AXM ch 4 — Readings: 1.61E+06 cpm — Flowrate: 5000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	3.77E+01	2.23E+01	1.98E+02	7.70E+01	2.97E+02	5.01E+03
0.5	3.04E+01	1.84E+01	1.41E+02	5.48E+01	2.14E+02	3.67E+03
0.7	2.06E+01	1.26E+01	8.52E+01	3.30E+01	1.31E+02	2.18E+03
1.0	1.28E+01	7.86E+00	4.80E+01	1.85E+01	7.44E+01	1.21E+03
1.5	7.08E+00	4.44E+00	2.51E+01	9.60E+00	3.92E+01	6.32E+02
2.0	5.16E+00	3.28E+00	1.56E+01	5.91E+00	2.48E+01	3.94E+02
3.0	3.93E+00	2.62E+00	9.76E+00	3.62E+00	1.60E+01	2.46E+02
4.0	3.30E+00	2.11E+00	8.05E+00	2.94E+00	1.31E+01	2.03E+02
5.0	2.82E+00	1.79E+00	7.03E+00	2.53E+00	1.13E+01	1.77E+02
7.0	2.00E+00	1.25E+00	5.29E+00	1.86E+00	8.40E+00	1.33E+02
10.0	1.35E+00	8.78E-01	4.10E+00	1.39E+00	6.37E+00	1.03E+02

Assessment Data Results Saved to File.
Grand Gulf 10Miles Monitored Release 04052017 141715.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



PAGs Exceeded in Designated Areas

***** Classification: General Emergency *****

Release Rates (Ci / sec)	
Particulate	8.92E-02 (0.7%)
Iodine	1.72E+00 (12.9%)
Noble Gas	1.15E+01 (86.4%)

Reviewed By: _____

Page 1 of 3 — Grand Gulf / 2.0.1.0

Turbine Building Vent AXM 4 – Site Area Emergency

Dose Assessment

Grand Gulf Wednesday, April 5, 2017 14:18

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <TB Bldg> <HVAC Filters> <Env> PRF: 8.00E-02

Containment HUT: = N/A | Containment Sprays: = N/A | Supp Pool Status: = N/A | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = N/A | Turbine Bldg HUT: = <2 Hours | RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 1:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

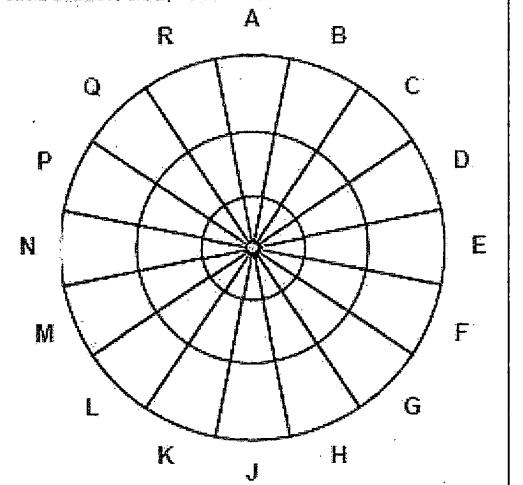
Monitor: AXM ch 4 Readings: 1.61E+05 cpm Flowrate: 5000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	3.77E+00	2.33E+00	1.98E+01	7.70E+00	2.97E+01	5.01E+02
0.5	3.04E+00	1.84E+00	1.41E+01	5.48E+00	2.14E+01	3.57E+02
0.7	2.06E+00	1.26E+00	8.62E+00	3.30E+00	1.31E+01	2.16E+02
1.0	1.28E+00	7.98E-01	4.80E+00	1.85E+00	7.44E+00	1.21E+02
1.5	7.08E-01	4.44E-01	2.51E+00	9.60E-01	3.92E+00	6.32E+01
2.0	6.16E-01	3.28E-01	1.56E+00	5.91E-01	2.48E+00	3.94E+01
3.0	3.93E-01	2.62E-01	9.76E-01	3.62E-01	1.60E+00	2.46E+01
4.0	3.30E-01	2.11E-01	8.05E-01	2.94E-01	1.31E+00	2.03E+01
5.0	2.82E-01	1.70E-01	7.03E-01	2.53E-01	1.13E+00	1.77E+01
7.0	2.00E-01	1.25E-01	5.39E-01	1.86E-01	8.40E-01	1.33E+01
10.0	1.35E-01	0.00E+00	4.10E-01	1.39E-01	5.49E-01	1.03E+01

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 04052017 141800.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Classification: Site Area Emergency

Release Rates (Ci / sec)	
Particulate	8.92E-03 (0.7%)
Iodine	1.72E-01 (12.9%)
Noble Gas	1.15E+00 (86.4%)

Reviewed By: _____

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Turbine Building Vent AXM 4 – Alert

Dose Assessment

Grand Gulf Wednesday, April 5, 2017 14:18

Method: Detailed Assessment - Monitored Release

Release Pathway: <RCS> <TB Bldg> <HVAC Filters> <Env> PRF: 8.00E-02

Containment HUT: = N/A	Containment Sprays: = N/A	Supp Pool Status: = N/A	Safety Filters: = N/A
HVAC Filters: = Working	Aux Bldg HUT: = N/A	Turbine Bldg HUT: = <2 Hours	RadWaste Bldg HUT: = N/A

Source Term: Reactor Core Accident - Clad

Time After S/D (hh:mm): 1:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

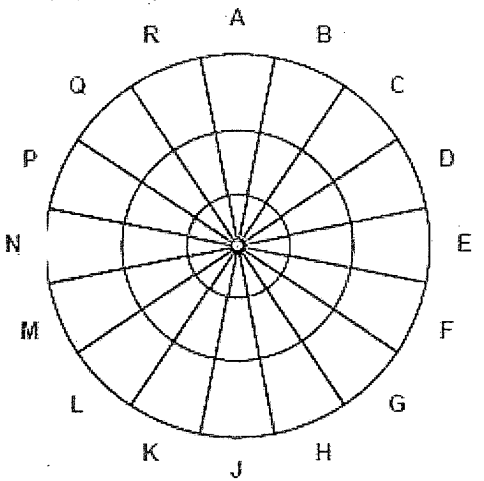
Monitor: AXM ch 4 Readings: 1.61E+04 cpm Flowrate: 5000 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	3.77E-01	2.23E-01	1.98E+00	7.70E-01	2.97E+00	5.01E+01
0.5	3.04E-01	1.84E-01	1.41E+00	5.48E-01	2.14E+00	3.57E+01
0.7	2.08E-01	1.28E-01	8.52E-01	3.30E-01	1.31E+00	2.16E+01
1.0	1.28E-01	0.00E+00	4.80E-01	1.85E-01	6.65E-01	1.21E+01
1.5	0.00E+00	0.00E+00	2.51E-01	0.00E+00	2.51E-01	6.32E+00
2.0	0.00E+00	0.00E+00	1.56E-01	0.00E+00	1.56E-01	3.94E+00
3.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.46E+00
4.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.03E+00
5.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.77E+00
7.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.33E+00
10.0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.03E+00

Assessment Data Results Saved to File.
Grand Gulf 10Miles Monitored Release 04052017 141852.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

*** Classification: Validate against Emergency Action Levels ***

Release Rates (Ci / sec)	
Particulate	8.92E-04 (0.7%)
Iodine	1.72E-02 (12.9%)
Noble Gas	1.15E-01 (86.4%)

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

Fuel Handling Event via Aux Building Vent SPING 7 – General Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 21:12

Method: Detailed Assessment - Monitored Release

Release Pathway: <SF> <Under Water> <AUX Bldg> < HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = N/A	Containment Sprays: = N/A	Supp Pool Status: = N/A	Safety Filters: = N/A
HVAC Filters: = Working	Aux Bldg HUT: = < 2 Hours	Turbine Bldg HUT: = N/A	RadWaste Bldg HUT: = N/A

Source Term: Spent Fuel Accident - Under Water Damage: 0.250 %
Time Since Irradiated (hh:mm): 80:00
Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower
Wind: From 270° @ 4.4 mph
Stability Class: D
Precipitation: None

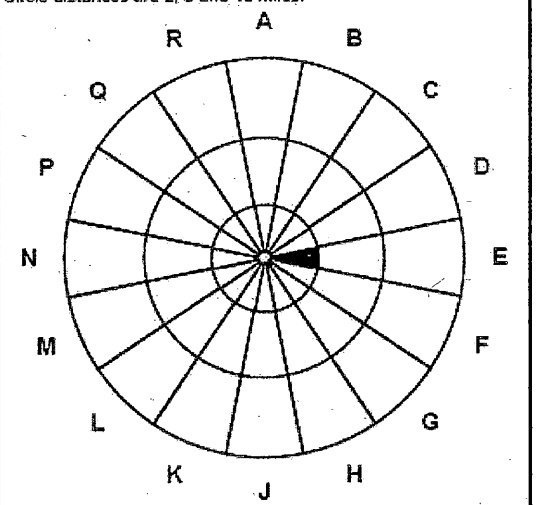
Monitor: SPING ch 7 Readings: 6.43E+06 cpm Flowrate: 24720 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.40E+03	9.80E+02	1.90E+01	5.88E+00	1.00E+03	4.22E+02
0.5	1.13E+03	7.92E+02	1.38E+01	4.27E+00	8.10E+02	3.07E+02
0.7	7.40E+02	5.20E+02	8.48E+00	2.62E+00	5.31E+02	1.88E+02
1.0	4.52E+02	3.16E+02	4.84E+00	1.49E+00	3.23E+02	1.08E+02
1.5	2.44E+02	1.71E+02	2.56E+00	7.93E-01	1.74E+02	5.68E+01
2.0	1.98E+02	1.39E+02	1.60E+00	4.94E-01	1.41E+02	3.55E+01
3.0	1.39E+02	9.56E+01	9.22E-01	2.76E-01	9.68E+01	2.06E+01
4.0	1.17E+02	8.13E+01	7.72E-01	2.30E-01	8.23E+01	1.72E+01
5.0	1.05E+02	7.09E+01	6.70E-01	1.97E-01	7.18E+01	1.49E+01
7.0	8.36E+01	5.83E+01	5.47E-01	1.59E-01	5.90E+01	1.22E+01
10.0	6.44E+01	4.41E+01	4.09E-01	1.17E-01	4.46E+01	9.12E+00

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 211259.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



PAGs Exceeded in Designated Areas

Release Rates (Ci / sec)	
Particulate	1.43E-02 (0.0%)
Iodine	3.41E-02 (0.0%)
Noble Gas	8.62E+03 (100.0%)

***** Classification: General Emergency *****

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

Fuel Handling Event via Aux Building Vent SPING 7 – Site Area Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 21:15

Method: Detailed Assessment - Monitored Release

Release Pathway: <SF> <Under Water> <AUX Bldg> < HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = N/A | Containment Sprays: = N/A | Supp Pool Status: = N/A | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = < 2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = N/A

Source Term: Spent Fuel Accident - Under Water Damage: 0.250 %

Time Since Irradiated (hh:mm): 80:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

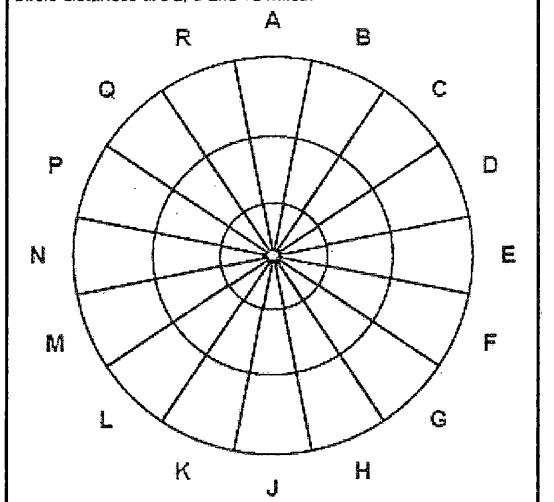
Monitor: SPING ch 7 Readings: 6.43E+05 cpm Flowrate: 24720 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.40E+02	9.80E+01	1.90E+00	5.88E-01	1.00E+02	4.22E+01
0.5	1.13E+02	7.92E+01	1.38E+00	4.27E-01	8.10E+01	3.07E+01
0.7	7.40E+01	5.20E+01	8.48E-01	2.62E-01	5.31E+01	1.88E+01
1.0	4.52E+01	3.16E+01	4.84E-01	1.49E-01	3.23E+01	1.08E+01
1.5	2.44E+01	1.71E+01	2.56E-01	0.00E+00	1.73E+01	5.68E+00
2.0	1.98E+01	1.39E+01	1.60E-01	0.00E+00	1.40E+01	3.55E+00
3.0	1.39E+01	9.56E+00	0.00E+00	0.00E+00	9.56E+00	2.06E+00
4.0	1.17E+01	8.13E+00	0.00E+00	0.00E+00	8.13E+00	1.72E+00
5.0	1.05E+01	7.09E+00	0.00E+00	0.00E+00	7.09E+00	1.49E+00
7.0	8.36E+00	5.83E+00	0.00E+00	0.00E+00	5.83E+00	1.22E+00
10.0	6.44E+00	4.41E+00	0.00E+00	0.00E+00	4.41E+00	9.12E-01

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 211527.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Classification: Site Area Emergency

Release Rates (Ci / sec)	
Particulate	1.43E-03 (0.0%)
Iodine	3.41E-03 (0.0%)
Noble Gas	8.62E+02 (100.0%)

Reviewed By: _____

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Fuel Handling Event via Aux Building Vent SPING 7 – Alert

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 21:16

Method: Detailed Assessment - Monitored Release

Release Pathway: <SF> <Under Water> <AUX Bldg> < HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = N/A	Containment Sprays: = N/A	Supp Pool Status: = N/A	Safety Filters: = N/A
HVAC Filters: = Working	Aux Bldg HUT: = < 2 Hours	Turbine Bldg HUT: = N/A	RadWaste Bldg HUT: = N/A

Source Term: Spent Fuel Accident - Under Water Damage: 0.250 %
Time Since Irradiated (hh:mm): 80:00
Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

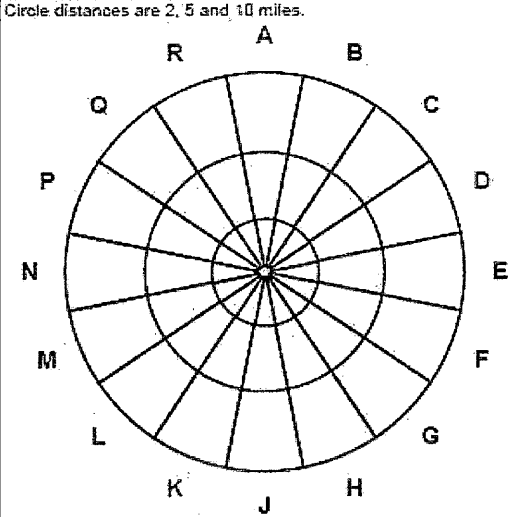
OnSite Lower
Wind: From 270° @ 4.4 mph
Stability Class: D
Precipitation: None

Monitor: SPING ch 7 Readings: 6.44E+04 cpm Flowrate: 24720 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.40E+01	9.81E+00	1.90E-01	0.00E+00	1.00E+01	4.23E+00
0.5	1.13E+01	7.92E+00	1.38E-01	0.00E+00	8.06E+00	3.08E+00
0.7	7.44E+00	5.20E+00	0.00E+00	0.00E+00	5.20E+00	1.89E+00
1.0	4.52E+00	3.17E+00	0.00E+00	0.00E+00	3.17E+00	1.08E+00
1.5	2.45E+00	1.71E+00	0.00E+00	0.00E+00	1.71E+00	5.72E-01
2.0	1.99E+00	1.39E+00	0.00E+00	0.00E+00	1.39E+00	3.56E-01
3.0	1.39E+00	9.60E-01	0.00E+00	0.00E+00	9.60E-01	2.06E-01
4.0	1.18E+00	8.17E-01	0.00E+00	0.00E+00	8.17E-01	1.73E-01
5.0	1.05E+00	7.10E-01	0.00E+00	0.00E+00	7.10E-01	1.50E-01
7.0	8.40E-01	5.83E-01	0.00E+00	0.00E+00	5.83E-01	1.22E-01
10.0	6.48E-01	4.41E-01	0.00E+00	0.00E+00	4.41E-01	0.00E+00

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 211639.URI7

Evacuation Areas From 0 to 10 Miles
Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Release Rates (Ci / sec)	
Particulate	1.43E-04 (0.0%)
Iodine	3.41E-04 (0.0%)
Noble Gas	8.64E+01 (100.0%)

* * * Classification: Validate against Emergency Action Levels * * *

Reviewed By: _____

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Fuel Handling Event via Aux Building Vent AXM 3 – General Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 21:19

Method: Detailed Assessment - Monitored Release

Release Pathway: <SF> <Under Water> <AUX Bldg> < HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = N/A | Containment Sprays: = N/A | Supp Pool Status: = N/A | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = <.2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = N/A

Source Term: Spent Fuel Accident - Under Water Damage: 0.250 %

Time Since Irradiated (hh:mm): 80:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

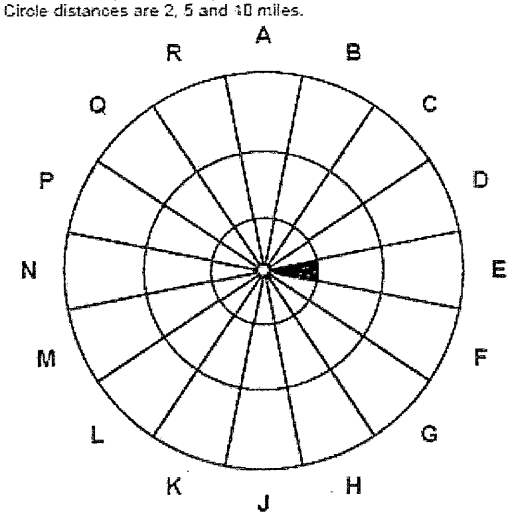
Monitor: AXM ch 3 Readings: 4.38E+05 cpm Flowrate: 24720 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.40E+03	9.80E+02	1.90E+01	5.88E+00	1.00E+03	4.23E+02
0.5	1.13E+03	7.92E+02	1.38E+01	4.27E+00	8.10E+02	3.08E+02
0.7	7.40E+02	5.20E+02	8.48E+00	2.62E+00	5.31E+02	1.89E+02
1.0	4.52E+02	3.17E+02	4.84E+00	1.50E+00	3.23E+02	1.08E+02
1.5	2.44E+02	1.71E+02	2.56E+00	7.93E-01	1.75E+02	5.72E+01
2.0	1.98E+02	1.39E+02	1.60E+00	4.94E-01	1.41E+02	3.56E+01
3.0	1.39E+02	9.57E+01	9.27E-01	2.76E-01	9.70E+01	2.06E+01
4.0	1.17E+02	8.15E+01	7.75E-01	2.30E-01	8.25E+01	1.73E+01
5.0	1.05E+02	7.09E+01	6.72E-01	1.97E-01	7.18E+01	1.50E+01
7.0	8.40E+01	5.83E+01	5.47E-01	1.59E-01	5.90E+01	1.22E+01
10.0	6.48E+01	4.41E+01	4.10E-01	1.17E-01	4.46E+01	9.14E+00

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 211918.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



PAGs Exceeded in Designated Areas

Release Rates (Ci / sec)	
Particulate	1.43E-02 (0.0%)
Iodine	3.41E-02 (0.0%)
Noble Gas	8.63E+03 (100.0%)

***** Classification: General Emergency *****

Reviewed By: _____

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Fuel Handling Event via Aux Building Vent AXM 3 – Site Area Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 21:20

Method: Detailed Assessment - Monitored Release

Release Pathway: <SF> <Under Water> <AUX Bldg> < HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = N/A | Containment Sprays: = N/A | Supp Pool Status: = N/A | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = < 2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = N/A

Source Term: Spent Fuel Accident - Under Water Damage: 0.250 %

Time Since Irradiated (hh:mm): 80:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

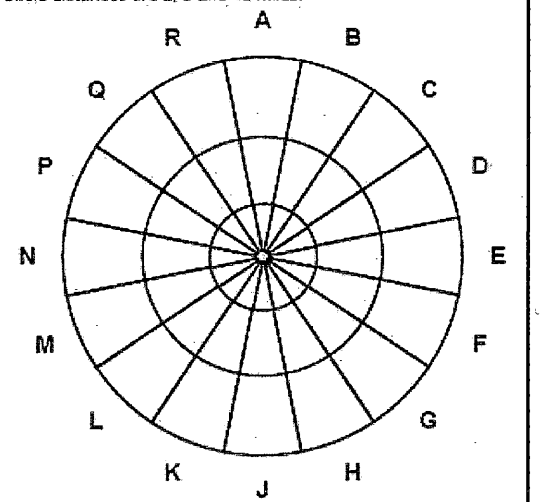
Monitor: AXM ch 3 Readings: 4.38E+04 cpm Flowrate: 24720 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.40E+02	9.80E+01	1.90E+00	5.88E-01	1.00E+02	4.23E+01
0.5	1.13E+02	7.92E+01	1.38E+00	4.27E-01	8.10E+01	3.08E+01
0.7	7.40E+01	5.20E+01	8.48E-01	2.62E-01	5.31E+01	1.89E+01
1.0	4.52E+01	3.17E+01	4.84E-01	1.50E-01	3.23E+01	1.08E+01
1.5	2.44E+01	1.71E+01	2.56E-01	0.00E+00	1.74E+01	5.72E+00
2.0	1.98E+01	1.39E+01	1.60E-01	0.00E+00	1.40E+01	3.56E+00
3.0	1.39E+01	9.57E+00	0.00E+00	0.00E+00	9.57E+00	2.06E+00
4.0	1.17E+01	8.15E+00	0.00E+00	0.00E+00	8.15E+00	1.73E+00
5.0	1.05E+01	7.09E+00	0.00E+00	0.00E+00	7.09E+00	1.50E+00
7.0	8.40E+00	5.83E+00	0.00E+00	0.00E+00	5.83E+00	1.22E+00
10.0	6.48E+00	4.41E+00	0.00E+00	0.00E+00	4.41E+00	9.14E-01

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 212035.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Classification: Site Area Emergency

Release Rates (Ci / sec)	
Particulate	1.43E-03 (0.0%)
Iodine	3.41E-03 (0.0%)
Noble Gas	8.63E+02 (100.0%)

Reviewed By: _____

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Fuel Handling Event via Aux Building Vent AXM 3 – Alert

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 21:21

Method: Detailed Assessment - Monitored Release

Release Pathway: <SF> <Under Water> <AUX Bldg> < HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = N/A	Containment Sprays: = N/A	Supp Pool Status: = N/A	Safety Filters: = N/A
HVAC Filters: = Working	Aux Bldg HUT: = < 2 Hours	Turbine Bldg HUT: = N/A	RadWaste Bldg HUT: = N/A

Source Term: Spent Fuel Accident - Under Water Damage: 0.250 %
Time Since Irradiated (hh:mm): 80:00
Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower
Wind: From 270° @ 4.4 mph
Stability Class: D
Precipitation: None

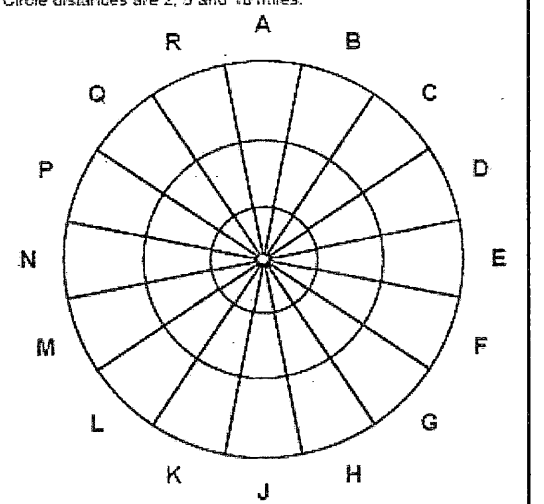
Monitor: AXM ch 3 Readings: 4.39E+03 cpm Flowrate: 24720 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.40E+01	9.81E+00	1.90E-01	0.00E+00	1.00E+01	4.23E+00
0.5	1.13E+01	7.92E+00	1.38E-01	0.00E+00	8.06E+00	3.08E+00
0.7	7.44E+00	5.20E+00	0.00E+00	0.00E+00	5.20E+00	1.89E+00
1.0	4.52E+00	3.17E+00	0.00E+00	0.00E+00	3.17E+00	1.08E+00
1.5	2.45E+00	1.72E+00	0.00E+00	0.00E+00	1.72E+00	5.72E-01
2.0	1.99E+00	1.39E+00	0.00E+00	0.00E+00	1.39E+00	3.56E-01
3.0	1.40E+00	9.60E-01	0.00E+00	0.00E+00	9.60E-01	2.06E-01
4.0	1.18E+00	8.17E-01	0.00E+00	0.00E+00	8.17E-01	1.73E-01
5.0	1.05E+00	7.10E-01	0.00E+00	0.00E+00	7.10E-01	1.50E-01
7.0	8.40E-01	5.83E-01	0.00E+00	0.00E+00	5.83E-01	1.22E-01
10.0	6.48E-01	4.42E-01	0.00E+00	0.00E+00	4.42E-01	0.00E+00

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 212146.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Release Rates (Ci / sec)	
Particulate	1.44E-04 (0.0%)
Iodine	3.42E-04 (0.0%)
Noble Gas	8.65E+01 (100.0%)

*** Classification: Validate against Emergency Action Levels ***

Reviewed By: _____

Page 1 of 3 — Grand Gulf / 2.0.1.0

Fuel Handling Event via Aux Building Vent AXM 4 – Site Area Emergency

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 21:24

Method: Detailed Assessment - Monitored Release

Release Pathway: <SF> <Under Water> <AUX Bldg> < HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = N/A — Containment Sprays: = N/A — Supp Pool Status: = N/A — Safety Filters: = N/A

HVAC Filters: = Working — Aux Bldg HUT: = < 2 Hours — Turbine Bldg HUT: = N/A — RadWaste Bldg HUT: = N/A

Source Term: Spent Fuel Accident - Under Water Damage: 0.250 %

Time Since Irradiated (hh:mm): 80:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

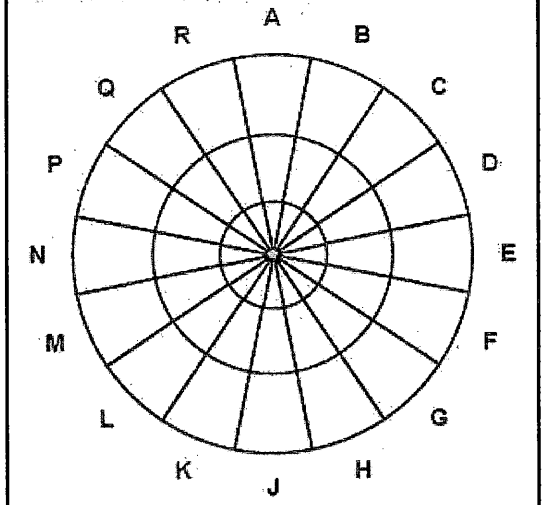
Monitor: AXM ch 4 Readings: 2.43E+07 cpm Flowrate: 24720 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.40E+02	9.80E+01	1.90E+00	5.86E-01	1.00E+02	4.22E+01
0.5	1.13E+02	7.92E+01	1.38E+00	4.26E-01	8.10E+01	3.07E+01
0.7	7.40E+01	5.20E+01	8.48E-01	2.62E-01	5.31E+01	1.88E+01
1.0	4.52E+01	3.16E+01	4.84E-01	1.49E-01	3.23E+01	1.08E+01
1.5	2.44E+01	1.71E+01	2.56E-01	0.00E+00	1.73E+01	5.68E+00
2.0	1.98E+01	1.39E+01	1.59E-01	0.00E+00	1.40E+01	3.55E+00
3.0	1.39E+01	9.56E+00	0.00E+00	0.00E+00	9.56E+00	2.06E+00
4.0	1.17E+01	8.13E+00	0.00E+00	0.00E+00	8.13E+00	1.72E+00
5.0	1.05E+01	7.09E+00	0.00E+00	0.00E+00	7.09E+00	1.49E+00
7.0	8.36E+00	5.83E+00	0.00E+00	0.00E+00	5.83E+00	1.22E+00
10.0	6.44E+00	4.41E+00	0.00E+00	0.00E+00	4.41E+00	9.12E-01

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 212400.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

Release Rates (Ci / sec)	
Particulate	1.43E-03 (0.0%)
Iodine	3.40E-03 (0.0%)
Noble Gas	8.62E+02 (100.0%)

Classification: Site Area Emergency

Reviewed By: _____

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Fuel Handling Event via Aux Building Vent AXM 4 – Alert

Dose Assessment

Grand Gulf — Tuesday, February 6, 2018 21:25

Method: Detailed Assessment - Monitored Release

Release Pathway: <SF> <Under Water> <AUX Bldg> < HVAC Filters> <Env> — PRF: 8.00E-04

Containment HUT: = N/A | Containment Sprays: = N/A | Supp Pool Status: = N/A | Safety Filters: = N/A

HVAC Filters: = Working | Aux Bldg HUT: = < 2 Hours | Turbine Bldg HUT: = N/A | RadWaste Bldg HUT: = N/A

Source Term: Spent Fuel Accident - Under Water Damage: 0.250 %

Time Since Irradiated (hh:mm): 80:00

Release Duration (hh:mm): 1:00 ETE (hh:mm): [N/A]

OnSite Lower

Wind: From 270° @ 4.4 mph

Stability Class: D

Precipitation: None

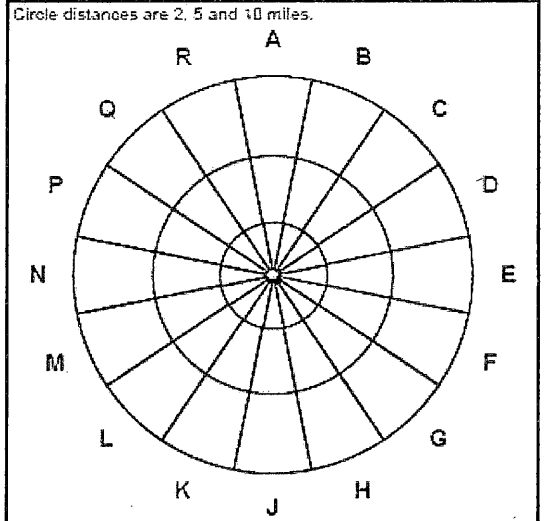
Monitor: AXM ch 4 Readings: 2.44E+06 cpm Flowrate: 24720 CFM

Distance (Miles)	Exposure Rate (mR/hr)	External Plume DDE (mRem)	Inhalation CEDE (mRem)	Deposition Ground DDE (mRem)	TEDE (mRem)	CDE Thyroid (mRem)
S.B.	1.40E+01	9.81E+00	1.90E-01	0.00E+00	1.00E+01	4.23E+00
0.5	1.13E+01	7.92E+00	1.38E-01	0.00E+00	8.06E+00	3.08E+00
0.7	7.44E+00	5.20E+00	0.00E+00	0.00E+00	5.20E+00	1.89E+00
1.0	4.52E+00	3.17E+00	0.00E+00	0.00E+00	3.17E+00	1.08E+00
1.5	2.45E+00	1.72E+00	0.00E+00	0.00E+00	1.72E+00	5.72E-01
2.0	1.99E+00	1.39E+00	0.00E+00	0.00E+00	1.39E+00	3.56E-01
3.0	1.40E+00	9.60E-01	0.00E+00	0.00E+00	9.60E-01	2.06E-01
4.0	1.18E+00	8.17E-01	0.00E+00	0.00E+00	8.17E-01	1.73E-01
5.0	1.05E+00	7.10E-01	0.00E+00	0.00E+00	7.10E-01	1.50E-01
7.0	8.40E-01	5.83E-01	0.00E+00	0.00E+00	5.83E-01	1.22E-01
10.0	6.48E-01	4.42E-01	0.00E+00	0.00E+00	4.42E-01	0.00E+00

Assessment Data Results Saved to File:
Grand Gulf 10Miles Monitored Release 02062018 212509.URI7

Evacuation Areas From 0 to 10 Miles

Circle distances are 2, 5 and 10 miles.



No PAGs Exceeded

* * * Classification: Validate against Emergency Action Levels * * *

Release Rates (Ci / sec)	
Particulate	1.44E-04 (0.0%)
Iodine	3.42E-04 (0.0%)
Noble Gas	8.65E+01 (100.0%)

Reviewed By: _____

Page 1 of 3 Grand Gulf / 2.0.1.0

GNRO-2018/00048, ENCLOSURE

ATTACHMENT 2

GGNS EAL BASIS DOCUMENT

Grand Gulf Nuclear Station
EAL Technical Basis

Table of Contents

1.0 INTRODUCTION

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Grand Gulf Nuclear Station (GGNS). It should be used to facilitate review of the GGNS EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of 10-S-01-1, Activation of the Emergency Plan, 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 or less in all cases of conditions present. Use of this document for assistance is not intended to delay the emergency classification.

Because the information in a basis document can affect emergency classification decision-making (e.g., the Emergency 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).

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 GGNS Emergency Plan.

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

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

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

Subsequently, Revision 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels

for Non-Passive Reactors," November 2012 (ref. 4.1.1), GGNS conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

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

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

The primary fission product barriers are:

A. Fuel Clad Barrier (FCB): The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.

B. Reactor Coolant System Barrier (RCB): The RCS Barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping out to and including the isolation valves.

C. Containment Barrier (CNB): The Containment Barrier includes the drywell, the containment, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. 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 GGNS EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
  - EALs applicable under <u>any</u> plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
  - EALs applicable only under <u>hot</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Startup, or Power Operation mode.
  - EALs applicable only under <u>cold</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling 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 and subcategories:

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

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

EAL Groups, Categories and Subcategories

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

2.5 Technical Bases Information

EAL technical bases are provided in Attachment 1 for each EAL according to EAL group (Any, Hot, Cold), EAL category (A, C, E, F, H and S) 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:

Category Letter & Title

Subcategory Number & Title

Initiating Condition (IC)

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

EAL Identifier (enclosed in rectangle)

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

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

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

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

Classification (enclosed in rectangle):

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

EAL (enclosed in rectangle)

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

Mode Applicability

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

Definitions:

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

Basis:

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

Reference(s):

Source documentation from which the EAL is derived

2.6 Operating Mode Applicability

1 Power Operation
Reactor is critical and the mode switch is in RUN

2 Startup
The mode switch is in REFUEL (with all reactor vessel head closure bolts fully tensioned) or STARTUP/HOT STANDBY

3 Hot Shutdown
The mode switch is in SHUTDOWN and average reactor coolant temperature is >200°F

4 Cold Shutdown
The mode switch is in SHUTDOWN and average reactor coolant temperature is ≤ 200°F

5 Refueling
The mode switch is in REFUEL or SHUTDOWN with one or more reactor vessel head closure bolts are less than fully tensioned

DEF Defueled
RPV contains no irradiated fuel

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

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Emergency 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.

EAL matrices should be read from left to right, from General Emergency to Unusual Event, and top to bottom. Declaration decisions should be independently verified before declaration is made except when gaining this verification would exceed the 15 minute declaration requirement. Place keeping should be used on all EAL matrices.

3.1.1 Classification Timeliness

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, "Interim Staff Guidance, Emergency Planning for Nuclear Power Plants" (ref. 4.1.8).

3.1.2 Valid Indications

All emergency classification assessments shall be based upon valid indications, reports or conditions. A valid indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy. For example, verification could be accomplished through an instrument channel check, response on related or redundant indicators, or direct observation by plant personnel.

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 10 § CFR 50.72 (ref. 4.1.4).

3.1.5 Classification Based on Analysis

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

3.1.6 Emergency 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. The NEI 99-01 EAL scheme provides the Emergency Director with the ability to classify events and conditions based upon judgment using EALs that are consistent with the Emergency Classification Level (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 in the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

3.2 Classification Methodology

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

When assessing an EAL that specifies a time duration for the off-normal condition, the "clock" for the EAL time duration runs concurrently with the emergency classification process "clock." For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01 (ref. 4.1.8).

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 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 units, an Alert should be declared.

3.2.2 Consideration of Mode Changes During Classification

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

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

3.2.3 Classification of Imminent Conditions

Although EALs provide specific thresholds, the Emergency 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 emergency classification levels, this approach is particularly important at the higher emergency classification levels since it provides additional time for implementation of protective measures.

3.2.4 Emergency Classification Level Upgrading and Termination

An ECL may be terminated when the event or condition that meets the classified IC and EAL no longer exists, and other site-specific termination requirements are met.

3.2.5 Classification of Short-Lived Events

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

3.2.6 Classification of Transient Conditions

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

<u>EAL momentarily met during expected plant response</u> - In instances in which 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 the high pressure ECCS systems fail to automatically start. The plant enters an inadequate core cooling condition (a potential loss of both the Fuel Clad and RCS Barriers). If an operator manually starts a high pressure ECCS system in accordance with an EOP step and clears the inadequate core cooling condition prior to an emergency declaration, then the classification should be based on the ATWS only.

It is important to stress that the 15-minute emergency classification assessment period (process clock) is not a "grace period" during which a classification may be delayed to allow the performance of a corrective action that would obviate the need to classify the event. Emergency classification assessments must be deliberate and timely, with no undue delays. The provision discussed above addresses only those rapidly evolving situations when an operator is able to take a successful corrective action prior to the Emergency 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 NEI 99-01 Revision 6, Methodology for the Development of Emergency Action Levels for Non-Passive Reactors, ADAMS Accession Number ML12326A805

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

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

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

4.1.5 10 § CFR 50.73 License Event Report System

4.1.6 GGNS Technical Specifications Table 1.1-1, Modes

4.1.7 GGNS Offsite Dose Calculation Manual (ODCM)

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

4.1.9 GGNS Emergency Plan

4.1.10 GGNS UFSAR 9.1.4.2.10.4 Storage of Fuel at the Independent Spent Fuel Storage Installation (ISFSI)

4.1.11 GGNS UFSAR 9.1.4.2.10 Description of Fuel Transfer

4.1.12 SOPP-018-1 Shutdown Operations Protection Plan

4.1.13 10-S-01-12 Radiological Assessment and Protective Action Recommendations

4.2 Implementing

4.2.1 10-S-01-1 Activation of the Emergency Plan

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

4.2.3 GGNS EAL Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

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

Alert

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

Confinement Boundary

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the GGNS ISFSI, Confinement Boundary is defined as the Holtec System Multi-Purpose Canister (MPC) (ref. 4.1.10).

Containment Closure

The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Containment Closure is established when either Primary or Secondary Containment integrity is established (ref. 4.1.12).

Emergency Action Level (EAL)

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

Emergency Classification Level (ECL)

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

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

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.

Fission Product Barrier Threshold

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

Flooding

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

General Emergency

Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG 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 GGNS 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 GGNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the SECURITY OWNER CONTROLLED AREA(SOCA)).

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).

Independent Spent Fuel Storage Installation (ISFSI)

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

Initiating Condition (IC)

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

Projectile

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

Protected Area

An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled. (ref. 4.1.9).

RCS Intact

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

Refueling Pathway

Reactor cavity (well), upper containment pool, fuel transfer canal, and auxiliary building fuel pools, but **not** including the reactor vessel, comprise the refueling pathway (ref. 4.1.11).

Restore

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

Safety System

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

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

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

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

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

Security Condition

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

Security Owner Controlled Area (SOCA)

The SOCA is the area demarcated as a Vehicle Barrier System (VBS) consisting of passive elements including a series of large concrete blocks on the inside of a delay fence with early warning capabilities. The SOCA is the area between the SOCA Fence and the PROTECTED AREA Boundary.

Site Area Emergency

Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE 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 PAG exposure levels beyond the SITE BOUNDARY.

Site Boundary

That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor (ref. 4.1.13)

Unisolable

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

Unplanned

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

Unusual Event

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

Valid

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

Visible Damage

Damage to a SAFETY SYSTEM train 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 SAFETY SYSTEM train.

5.2 Abbreviations/Acronyms

°F ..... Degrees Fahrenheit
° ..... Degrees
AB ..... Auxiliary Building
AC ..... Alternating Current
AOP ..... Abnormal Operating Procedure
APRM ..... Average Power Range Meter
ARI ..... Alternate Rod Insertion
ATWS ..... Anticipated Transient Without Scram
BWR ..... Boiling Water Reactor
BWROG ..... Boiling Water Reactor Owners Group
CDE ..... Committed Dose Equivalent
CFR ..... Code of Federal Regulations
CNB ..... Containment Barrier
CS ..... Core Spray
CTMT ..... Containment
DEF ..... Defueled
DBA ..... Design Basis Accident
DC ..... Direct Current
D/G ..... Diesel Generator
EAL ..... Emergency Action Level
ECCS ..... Emergency Core Cooling System
ECL ..... Emergency Classification Level
EOF ..... Emergency Operations Facility
EOP ..... Emergency Operating Procedure
EPA ..... Environmental Protection Agency
EPG ..... Emergency Procedure Guideline
EPP ..... Emergency Plan Procedure
ERO ..... Emergency Response Organization
ESF ..... Engineered Safety Feature
FAA ..... Federal Aviation Administration
FBI ..... Federal Bureau of Investigation
FCB ..... Fuel Clad Barrier
FEMA ..... Federal Emergency Management Agency
FSAR ..... Final Safety Analysis Report
GE ..... General Emergency

HCTL ..... Heat Capacity Temperature Limit
HPCS ..... High Pressure Core Spray
IC ..... Initiating Condition
IPEEE ..... Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI ..... Independent Spent Fuel Storage Installation
K_{eff} ..... Effective Neutron Multiplication Factor
LCO ..... Limiting Condition of Operation
LER ..... Licensee Event Report
LOCA ..... Loss of Coolant Accident
LPCS ..... Low Pressure Core Spray
LRW ..... Liquid Radwaste
LWR ..... Light Water Reactor
MPC ..... Maximum Permissible Concentration/Multi-Purpose Canister
MPH ..... Miles Per Hour
mR, mRem, mrem, mREM ..... milli-Roentgen Equivalent Man
MSCRWL ..... Minimum Steam Cooling RPV Water Level
MSIV ..... Main Steam Isolation Valve
MSL ..... Main Steam Line
MW ..... Megawatt
NEI ..... Nuclear Energy Institute
NEIC ..... National Earthquake Information Center
NESP ..... National Environmental Studies Project
NORAD ..... North American Aerospace Defense Command
(NO)UE ..... Notification of Unusual Event
NPP ..... Nuclear Power Plant
NRC ..... Nuclear Regulatory Commission
NSSS ..... Nuclear Steam Supply System
OBE ..... Operating Basis Earthquake
ODCM ..... Offsite Dose Calculation Manual
ONEP ..... Off-Normal Event Procedure
ORO ..... Offsite Response Organization
PA ..... Protected Area
PAG ..... Protective Action Guideline
PB ..... Pushbutton
PCIS ..... Primary Containment Isolation System
PRA/PSA ..... Probabilistic Risk Assessment / Probabilistic Safety Assessment

PSIG ..... Pounds per Square Inch Gauge
R ..... Roentgen
RCB ..... RCS Barrier
RCIC ..... Reactor Core Isolation Cooling
RCS ..... Reactor Coolant System
Rem, rem, REM ..... Roentgen Equivalent Man
RETS ..... Radiological Effluent Technical Specifications
RHR ..... Residual Heat Removal
RPS ..... Reactor Protection System
RPT ..... Recirculation Pump Trip
RPV ..... Reactor Pressure Vessel
RWCU ..... Reactor Water Cleanup
SAP ..... Severe Accident Procedure
SAR ..... Safety Analysis Report
SBO ..... Station Blackout
SCBA ..... Self-Contained Breathing Apparatus
SOCA ..... Security Owner Controlled Area
SPDS ..... Safety Parameter Display System
SRO ..... Senior Reactor Operator
SRV ..... Safety Relief Valve
SSE ..... Safe Shutdown Earthquake
TEDE ..... Total Effective Dose Equivalent
TAF ..... Top of Active Fuel
TSC ..... Technical Support Center
UFSAR ..... Updated Final Safety Analysis Report
USGS ..... United States Geological Survey

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

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

GGNS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
AU1.1	AU1	1, 2
AU1.2	AU1	3
AU2.1	AU2	1
AA1.1	AA1	1
AA1.2	AA1	2
AA1.3	AA1	3
AA1.4	AA1	4
AA2.1	AA2	1
AA2.2	AA2	2
AA2.3	AA2	3
AA3.1	AA3	1
AA3.2	AA3	2
AS1.1	AS1	1
AS1.2	AS1	2
AS1.3	AS1	3
AS2.1	AS2	1
AG1.1	AG1	1
AG1.2	AG1	2
AG1.3	AG1	3

GGNS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
AG2.1	AG2	1
CU1.1	CU1	1
CU1.2	CU1	2
CU2.1	CU2	1
CU3.1	CU3	1
CU3.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1, 2, 3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA3.1	CA3	1, 2
CA6.1	CA6	1
CS1.1	CS1	1
CS1.2	CS1	2
CS1.3	CS1	3
CG1.1	CG1	2
EU1.1	EU1	1
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1, 2, 3
HU2.1	HU2	1

GGNS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
HU3.1	HU3	1
HU3.2	HU3	2
HU3.3	HU3	3
HU3.4	HU3	4
HU4.1	HU4	1
HU4.2	HU4	2
HU4.3	HU4	3
HU4.4	HU4	4
HU7.1	HU7	1
HA1.1	HA1	1, 2
HA5.1	HA5	1
HA6.1	HA6	1
HA7.1	HA7	1
HS1.1	HS1	1
HS6.1	HS6	1
HS7.1	HS7	1
HG7.1	HG7	1
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	1
SU4.2	SU3	2
SU5.1	SU4	1, 2, 3
SU6.1	SU5	1

GGNS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
SU6.2	SU5	2
SU7.1	SU6	1, 2, 3
N/A	SU7	1, 2
SA1.1	SA1	1
SA3.1	SA2	1
SA6.1	SA5	1
SA8.1	SA9	1
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1
SG1.1	SG1	1
SG1.2	SG8	1

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

Category A – Abnormal Rad Levels / Rad Effluent

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

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

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

Events of this category pertain to the following subcategories:

1. Radiological Effluent

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

2. Irradiated Fuel Event

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

3. Area Radiation Levels

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

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AU1.1 Unusual Event

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is 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 due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

Table A-1 Effluent Monitor Classification Thresholds

Release Point		GE	SAE	Alert	UE
Gaseous	SBGT A/B	8.1E+2 Ci/sec	8.1E+1 Ci/sec	8.1E+0 Ci/sec	6.7E-2 Ci/sec
	CTMT Vent	6.4E+2 Ci/sec	6.4E+1 Ci/sec	6.4E+0 Ci/sec	6.7E-2 Ci/sec
	Radwaste Building Vent	5.1E+1 Ci/sec	5.1E+0 Ci/sec	5.1E-1 Ci/sec	6.7E-2 Ci/sec
	Turbine Building Vent	1.3E+1 Ci/sec	1.3E+0 Ci/sec	1.3E-1 Ci/sec	6.7E-2 Ci/sec
	Fuel Handling (Aux BLDG) Vent	8.6E+3 Ci/sec	8.6E+2 Ci/sec	8.6E+1 Ci/sec	6.7E-2 Ci/sec
Liquid	Radwaste	----	----	----	7.33E+5 cpm

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

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

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

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

This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways as well as radioactivity releases that cause effluent radiation monitor readings to exceed 2 times the limit established by a radioactivity discharge permit. Such releases are typically associated with planned batch releases from non-continuous release pathways (e.g., radwaste, waste gas).

Escalation of the emergency classification level would be via IC AA1.

Reference(s):

1. IAS-04-1-01-D17-1 Process Radiation Monitoring
2. Offsite Dose Calculation Manual
3. XC-Q1D17-17001 Grand Gulf Nuclear Station (GGNS) Radiological Effluent EAL Threshold Values
4. NEI 99-01 AU1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AU1.2	**Unusual Event**
Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2)	

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

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

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

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

Escalation of the emergency classification level would be via IC AA1.

Reference(s):

1. Offsite Dose Calculation Manual
2. NEI 99-01 AU1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AA1.1 Alert

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is 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 due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

Note 4 The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table A-1 Effluent Monitor Classification Thresholds

Release Point		GE	SAE	Alert	UE
Gaseous	SBGT A/B	8.1E+2 Ci/sec	8.1E+1 Ci/sec	8.1E+0 Ci/sec	6.7E-2 Ci/sec
	CTMT Vent	6.4E+2 Ci/sec	6.4E+1 Ci/sec	6.4E+0 Ci/sec	6.7E-2 Ci/sec
	Radwaste Building Vent	5.1E+1 Ci/sec	5.1E+0 Ci/sec	5.1E-1 Ci/sec	6.7E-2 Ci/sec
	Turbine Building Vent	1.3E+1 Ci/sec	1.3E+0 Ci/sec	1.3E-1 Ci/sec	6.7E-2 Ci/sec
	Fuel Handling (Aux BLDG) Vent	8.6E+3 Ci/sec	8.6E+2 Ci/sec	8.6E+1 Ci/sec	6.7E-2 Ci/sec
Liquid	Radwaste	----	----	----	7.33E+5 cpm

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

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

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

Escalation of the emergency classification level would be via IC AS1.

Reference(s):

1. IAS-04-1-01-D17-1 Process Radiation Monitoring
2. XC-Q1D17-17001 Grand Gulf Nuclear Station (GGNS) Radiological Effluent EAL Threshold Values
3. NEI 99-01 AA1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AA1.2 Alert

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

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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

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

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

Escalation of the emergency classification level would be via IC AS1.

Reference(s):

1. 10-S-01-12 Radiological Assessment and Protective Action Recommendations
2. NEI 99-01 AA1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AA1.3 Alert

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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

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

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

This EAL is assessed per the ODCM (ref. 2)

Escalation of the emergency classification level would be via IC AS1.

Reference(s):

1. 10-S-01-12 Radiological Assessment and Protective Action Recommendations
2. Offsite Dose Calculation Manual
3. NEI 99-01 AA1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AA1.4 Alert

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

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

(Notes 1, 2)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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

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

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

Escalation of the emergency classification level would be via IC AS1.

Reference(s):

1. 10-S-01-12 Radiological Assessment and Protective Action Recommendations
2. NEI 99-01 AA1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AS1.1 Site Area Emergency

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is 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 due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table A-1 Effluent Monitor Classification Thresholds

Release Point		GE	SAE	Alert	UE
Gaseous	SBGT A/B	8.1E+2 Ci/sec	8.1E+1 Ci/sec	8.1E+0 Ci/sec	6.7E-2 Ci/sec
	CTMT Vent	6.4E+2 Ci/sec	6.4E+1 Ci/sec	6.4E+0 Ci/sec	6.7E-2 Ci/sec
	Radwaste Building Vent	5.1E+1 Ci/sec	5.1E+0 Ci/sec	5.1E-1 Ci/sec	6.7E-2 Ci/sec
	Turbine Building Vent	1.3E+1 Ci/sec	1.3E+0 Ci/sec	1.3E-1 Ci/sec	6.7E-2 Ci/sec
	Fuel Handling (Aux BLDG) Vent	8.6E+3 Ci/sec	8.6E+2 Ci/sec	8.6E+1 Ci/sec	6.7E-2 Ci/sec
Liquid	Radwaste	----	----	----	7.33E+5 cpm

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

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

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

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

Reference(s):

1. IAS-04-1-01-D17-1 Process Radiation Monitoring
2. XC-Q1D17-17001 Grand Gulf Nuclear Station (GGNS) Radiological Effluent EAL Threshold Values
3. NEI 99-01 AS1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AS1.2	**Site Area Emergency**
Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)	

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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

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

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

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

Reference(s):

1. 10-S-01-12 Radiological Assessment and Protective Action Recommendations
2. NEI 99-01 AS1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AS1.3 Site Area Emergency

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

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

(Notes 1, 2)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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

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

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

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

Reference(s):

1. 10-S-01-12 Radiological Assessment and Protective Action Recommendations
2. NEI 99-01 AS1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AG1.1 General Emergency

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is 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 due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table A-1 Effluent Monitor Classification Thresholds

Release Point		GE	SAE	Alert	UE
Gaseous	SBGT A/B	8.1E+2 Ci/sec	8.1E+1 Ci/sec	8.1E+0 Ci/sec	6.7E-2 Ci/sec
	CTMT Vent	6.4E+2 Ci/sec	6.4E+1 Ci/sec	6.4E+0 Ci/sec	6.7E-2 Ci/sec
	Radwaste Building Vent	5.1E+1 Ci/sec	5.1E+0 Ci/sec	5.1E-1 Ci/sec	6.7E-2 Ci/sec
	Turbine Building Vent	1.3E+1 Ci/sec	1.3E+0 Ci/sec	1.3E-1 Ci/sec	6.7E-2 Ci/sec
	Fuel Handling (Aux BLDG) Vent	8.6E+3 Ci/sec	8.6E+2 Ci/sec	8.6E+1 Ci/sec	6.7E-2 Ci/sec
Liquid	Radwaste	----	----	----	7.33E+5 cpm

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

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

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

Reference(s):

1. IAS-04-1-01-D17-1 Process Radiation Monitoring
2. XC-Q1D17-17001 Grand Gulf Nuclear Station (GGNS) Radiological Effluent EAL Threshold Values
3. NEI 99-01 AG1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AG1.2	**General Emergency**
Dose assessment using actual meteorology indicates doses > 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)	

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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

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

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

Reference(s):

1. 10-S-01-12 Radiological Assessment and Protective Action Recommendations
2. NEI 99-01 AG1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AG1.3 General Emergency

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

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

(Notes 1, 2)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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

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

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

Reference(s):

1. 10-S-01-12 Radiological Assessment and Protective Action Recommendations
2. NEI 99-01 AG1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: UNPLANNED loss of water level above irradiated fuel

EAL:

AU2.1 Unusual Event

UNPLANNED water level drop in the REFUELING PATHWAY as indicated by Fuel Pool Drain Tank low water level alarm, visual observation, water level drop in Upper Ctmt Pools, Aux Bldg Fuel Pools or the Fuel Transfer Canal

AND

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

- Ctmt 209 Airlock (1D21K630)
- Ctmt Fuel Hdlg Area (1D21K626)
- Aux Bldg Fuel Hdlg Area(1D21K622)

Mode Applicability:

All

Definition(s):

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

REFUELING PATHWAY- Reactor cavity (well), upper containment pool, fuel transfer canal, and auxiliary building fuel pools, but **not** including the reactor vessel, comprise the refueling pathway.

Basis:

This IC addresses a drop 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 drop 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 a rise 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 rise due to planned evolutions such as lifting of the reactor vessel head or movement of a fuel assembly. Note that this EAL is applicable only in cases where the elevated reading is due to an UNPLANNED loss of water level.

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

Escalation of the emergency classification level would be via IC AA2.

Reference(s):

1. 05-1-02-II-8 High Radiation During Fuel Handling
2. 04-1-01-D21-1 Area Radiation Monitoring System
3. UFSAR 12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation
4. NEI 99-01 AU2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AA2.1 Alert

IMMINENT uncovery of irradiated fuel in the REFUELING PATHWAY

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY- The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the GGNS ISFSI, Confinement Boundary is defined as the Holtec System Multi-Purpose Canister (MPC).

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.

REFUELING PATHWAY- Reactor cavity (well), upper containment pool, fuel transfer canal, and auxiliary building fuel pools, but **not** including the reactor vessel, comprise the refueling pathway.

Basis:

This IC addresses events that have caused IMMINENT or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the REFUELING PATHWAY. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant. This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC EU1.

This EAL escalates from AU2.1 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 a rise 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 with Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC AS1.

Reference(s):

1. NEI 99-01 AA2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity

AND

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

- Ctmt Vent (P601-19A-G9)
- FH Area Vent (P601-19A-C11)
- Ctmt 209 Airlock (P844-1A-A1)
- Ctmt Fuel Hdlg Area (P844-1A-A3)
- Aux Bldg Fuel Hdlg Area (P844-1A-A4)

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY- The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the GGNS ISFSI, Confinement Boundary is defined as the Holtec System Multi-Purpose Canister (MPC).

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

Basis:

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

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

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

Escalation of the emergency classification level would be via IC AS1.

Reference(s):

1. 05-1-02-II-8 High Radiation During Fuel Handling
2. 04-1-01-D21-1 Area Radiation Monitoring System
3. UFSAR 12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation
4. Offsite Dose Calculation Manual
5. NEI 99-01 AA2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AA2.3 **Alert**
Lowering of spent fuel pool level to 193 ft. (Level 2) on G41R040A(B)

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL addresses events that have caused 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.

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

Escalation of the emergency classification level would be via IC AS1 or AS2.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2 – 193 ft. 2 $^{1}/_{8}$ in. rounded to 193 ft. for readability) and SFP level at the top of the fuel racks (Level 3 – 183 ft. 2 $^{1}/_{8}$ in. rounded to 183 ft. for readability) (ref. 1).

G41R040A(B) Spent Fuel Pool Level Instrument is not located in the Control Room. The display cabinets are located in the 148' Control Building in the Lower Cable Spreading Room.

Reference(s):

1. 05-S-01-FSG-011 Alternate Spent Fuel Pool Makeup and Cooling
2. NEI 99-01 AA2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AS2.1 Site Area Emergency

Lowering of spent fuel pool level to 183 ft. (Level 3) on G41R040A(B)

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

Escalation of the emergency classification level would be via IC AG1 or AG2.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2 – 193 ft. 2 $^1/_8$ in. rounded to 193 ft. for readability) and SFP level at the top of the fuel racks (Level 3 – 183 ft. 2 $^1/_8$ in. rounded to 183 ft. for readability) (ref. 1).

G41 R040A(B) Spent Fuel Pool Level Instrument is not located in the Control Room. The display cabinets are located in the148' Control Building in the Lower Cable Spreading Room.

Reference(s):

1. 05-S-01-FSG-011 Alternate Spent Fuel Pool Makeup and Cooling
2. NEI 99-01 AS2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AG2.1	**General Emergency**
Spent fuel pool level **cannot** be restored to at least 183 ft. (Level 3) on G41R040A(B) for ≥ 60 min. (Note 1)	

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2 – 193 ft. 2 $^{1}/_{8}$ in. rounded to 193 ft. for readability) and SFP level at the top of the fuel racks (Level 3 – 183 ft. 2 $^{1}/_{8}$ in. rounded to 183 ft. for readability) (ref. 1).

G41 R040A(B) Spent Fuel Pool Level Instrument is not located in the Control Room. The display cabinets are located in the148' Control Building in the Lower Cable Spreading Room.

Reference(s):

1. 05-S-01-FSG-011 Alternate Spent Fuel Pool Makeup and Cooling
2. NEI 99-01 AG2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

AA3.1 Alert

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

- Control Room (SD21-K600)
- Central Alarm Station (by survey)

Mode Applicability:

All

Definition(s):

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

Basis:

Areas that meet this threshold include the Control Room (CR) and the Central Alarm Station (CAS). The Control Room is monitored for excessive radiation by one detector, SD21-K600 (ref. 1). The CAS is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations. While the CAS is in the Control Room Envelope, there are no permanently installed area radiation monitors in CAS that may be used to assess this EAL threshold. Therefore, this threshold is evaluated using local radiation survey for this area.

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 rise in radiation levels and determine if another IC may be applicable.

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

Reference(s):

1. 06-IC-1D21-R-1001 Area Radiation Monitoring Calibration
2. NEI 99-01 AA3

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

AA3.2 Alert

An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table A-3 room or area (Note 5)

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

Table A-3 Safe Operation & Shutdown Rooms/Areas

Room/Area	Mode
Control Building 111' SWGR Rms (0C202, 0C215)	3
Auxiliary Building 93' RHR A Pump Room (1A103)	3
Auxiliary Building 93' RHR B Pump Room (1A105)	3
Auxiliary Building 93' Corridor (1A101)	3
Auxiliary Building 119' Corridor (1A201)	3
Auxiliary Building 119' RHR A Pump Room (1A203)	3
Auxiliary Building 119' RHR B Pump Room (1A205)	3
Auxiliary Building 119' RCIC Room (1A204)	3
Auxiliary Building 139' RHR A Room (1A303, 1A304)	3
Auxiliary Building 139' RHR B Room (1A306, 1A307)	3
Radwaste Building 118' Radwaste Control Room (0R241)	3

Mode Applicability:

3 – Hot Shutdown

Definition(s):

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

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

Basis:

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 rise in radiation levels and determine if another IC may be applicable.

For AA3.2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the higher 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 rise occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 3.
- The higher 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 access control measures are of a conservative or precautionary nature, and would not actually prevent or IMPEDE a required action.

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

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

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

EAL AA3.2 mode applicability has been limited to the mode limitations of Table A-3 (Mode 3 **only**).

Reference(s):

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

Category C – Cold Shutdown / Refueling System Malfunction

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

Category C EALs are directly associated with cold shutdown or refueling system safety functions. Given the variability of plant configurations (e.g., systems out-of-service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The cold shutdown and refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, CONTAINMENT CLOSURE, and fuel clad integrity for the applicable operating modes (4 - Cold Shutdown, 5 - Refueling, DEF – Defueled).

The events of this category pertain to the following subcategories:

1. RPV Level

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

2. Loss of Emergency AC Power

Loss of vital plant electrical power can compromise plant SAFETY SYSTEM operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite power sources for 4.16 KV ESF buses.

3. RCS Temperature

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

4. Loss of Vital DC Power

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

5. Loss of Communications

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

6. Hazardous Event Affecting Safety Systems

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

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: UNPLANNED loss of RPV inventory

EAL:

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

Basis:

Grand Gulf is equipped with multiple RPV water level instruments including: Wide Range, Fuel Zone, Shutdown Range, Upset Range, and Narrow Range (ref. 1). Multiple instruments on different reference and variable legs should be monitored. The Upset Range and Shutdown Range instruments share a common reference leg; therefore, Narrow Range instruments should be routinely monitored when relying on Shutdown or Upset Range instrument as the primary indication.

With the plant in Cold Shutdown, RPV water level is normally maintained above the RPV low level scram setpoint of 11.4 in. (ref. 2). However, if RPV level is being controlled below the RPV low level scram setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

With the plant in Refueling mode, RPV water level is normally maintained at or above the reactor vessel flange. Technical Specifications require at least 22 ft 8 in. of water above the top of the reactor vessel flange in the refueling cavity during refueling operations (ref. 3). The RPV flange is at approximately 212 in. on the Shutdown Range. (ref. 4).

This EAL addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor RPV level concurrent

with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

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

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

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

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

Reference(s):

1. 02-S-01-40 EP Technical Bases
2. 05-S-01-EP-2 RPV Control
3. Technical Specifications 3.9.6
4. 07-S-14-413 RPV Disassembly
5. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: UNPLANNED loss of RPV inventory

EAL:

CU1.2 Unusual Event

RPV water level **cannot** be monitored

AND EITHER

- UNPLANNED rise in **any** Table C-1 sump or pool level due to a loss of RPV inventory
- Visual observation of UNISOLABLE RCS leakage

Table C-1 Sumps/Pool
• Drywell equipment drain sump • Drywell floor drain sump • CTMT equipment drain sump • CTMT floor drain sump • Suppression Pool • RHR A, B, C, HPCS, LPCS, RCIC room sumps • Auxiliary Building floor drain sump

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

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

UNPLANNED-. A parameter changes 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:

In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. RPV level in the Refueling mode is normally monitored using the Shutdown Range instrument which is re-spanned to indicate water level in the refuel cavity and the Core Plate d/p instrument which is re-spanned and re-scaled to indicate water level (ref. 1).

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Rise in drywell equipment drain sump level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 2, 3). An Auxiliary Building sump level rise may also be indicative of RCS inventory losses external to the Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

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

Refueling evolutions that lower RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level lowering 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 RPV level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels (Table C-1). Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RPV.

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

Reference(s):

1. 03-1-01-5 Refueling
2. 04-1-02-1H13-P601 Alarm Response Instruction Panel 1H13-P601
3. 04-1-02-1H13-P680 Alarm Response Instruction Panel 1H13-P680
4. 05-S-01-EP-4 Auxiliary Building Control
5. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Significant Loss of RPV inventory

EAL:

CA1.1 **Alert**
Loss of RPV inventory as indicated by RPV water level < -42 in. (Level 2)

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

None

Basis:

The threshold RPV water level of -42 in. is the Level 2 actuation setpoint for HPCS and RCIC. Although RCIC cannot restore RPV inventory in the cold condition, the Level 2 actuation setpoint is operationally significant and is indicative of a loss of RPV inventory significantly below the low RPV water level scram setpoint specified in CU1.1 (ref. 1, 2).

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

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

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

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

Reference(s):

1. Technical Specifications Table 3.3.5.1-1, Emergency Core Cooling System Instrumentation
2. 04-1-02-1H13-P601-16A-A4 Alarm Response Instruction Panel 1H13-P601 panel 16A-A4 RX LVL 2 (-42") LO
3. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Significant Loss of RPV inventory

EAL:

CA1.2 Alert

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

AND EITHER

- UNPLANNED rise in **any** Table C-1 sump or pool level due to a loss of RPV inventory
- Visual observation of UNISOLABLE RCS leakage

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table C-1 Sumps/Pool
• Drywell equipment drain sump • Drywell floor drain sump • CTMT equipment drain sump • CTMT floor drain sump • Suppression Pool • RHR A, B, C, HPCS, LPCS, RCIC room sumps • Auxiliary Building floor drain sump

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

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

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

Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. RPV level in the Refueling mode is normally monitored using the Shutdown Range instrument which is re-spanned to indicate water level in the refuel cavity and the Core Plate d/p instrument which is re-spanned and re-scaled to indicate water level. (ref. 1).

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Rise in drywell equipment drain sump level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 2, 3). An Auxiliary Building sump level rise may also be indicative of RCS inventory losses external to the Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

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

For this EAL, the inability to monitor RPV 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 RPV.

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

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

Reference(s):

1. 03-1-01-5 Refueling
2. 04-1-02-1H13-P601 Alarm Response Instruction Panel 1H13-P601
3. 04-1-02-1H13-P680 Alarm Response Instruction Panel 1H13-P680
4. 05-S-01-EP-4 Auxiliary Building Control
5. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.1 Site Area Emergency

CONTAINMENT CLOSURE **not** established

AND

RPV water level < -150 in. (Level 1)

Mode Applicability:

4 – Cold Shutdown, 5 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Containment Closure is established when either Primary or Secondary Containment integrity is established.

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

Basis:

The threshold RPV water level of -150 in. is the low-low-low ECCS actuation setpoint (Level 1). The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further lowering of RPV water level and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier (ref. 1, 2).

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

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

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control

functions. The difference in the specified RPV levels of EALs CS1.1 and CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

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

Escalation of the emergency classification level would be via IC CG1 or AG1.

Reference(s):

1. Technical Specifications Table 3.3.5.1-1, Emergency Core Cooling System Instrumentation
2. 04-1-02-1H13-P601-17A-E2 Alarm Response Instruction Panel 1H13-P601 panel 17A-E2 RX LVL 1 (-150") LO
3. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.2 Site Area Emergency
CONTAINMENT CLOSURE established
AND
RPV water level < -167 in. (TAF)

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Containment Closure is established when either Primary or Secondary Containment integrity is established.

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

Basis:

When RPV level drops to the top of active fuel (TAF) (an indicated RPV level of -167 in.), core uncovery starts to occur (ref. 1).

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

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

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RPV levels of EALs CS1.1 and CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

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

Escalation of the emergency classification level would be via IC CG1 or AG1.

Reference(s):

1. 02-S-01-40 EP Technical Bases
2. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.3 Site Area Emergency

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

AND

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

- UNPLANNED rise in Suppression Pool level of sufficient magnitude to indicate core uncovery
- Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery
- Containment/Drywell High Range Area Radiation Monitor (1D21-K648B-C) ≥ 100 R/hr

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 – Cold Shutdown, 5 – Refueling

Definition(s):

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.

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

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

Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. RPV level in the Refueling mode is normally monitored using the Shutdown Range instrument which is re-spanned to indicate water level in the refuel cavity and the Core Plate d/p instrument which is re-spanned and re-scaled to indicate water level (ref. 1).

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications. Level rises must be evaluated against other potential

sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in Suppression Pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

In the Refueling Mode, as water level in the RPV lowers, the dose rate above the core will rise, with corresponding indications on area radiation monitors. 100 R/hr is used for this indication on Containment High Range Radiation Monitors (1D21-K648B and C). These detectors are located on the containment wall in a position to monitor the containment radiation environment above the refueling cavity elevation.

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

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

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

The inability to monitor RPV 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 RPV.

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

Escalation of the emergency classification level would be via IC CG1 or AG1

Reference(s):

1. 03-1-01-5 Refueling
2. 04-1-02-1H13-P601 Alarm Response Instruction Panel 1H13-P601
3. 04-1-02-1H13-P680 Alarm Response Instruction Panel 1H13-P680
4. 05-S-01-EP-4 Auxiliary Building Control
5. 06-IC-1D21-R-1002 Containment/Drywell High Range Area Radiation Monitor Calibration
6. NEI 99-01 CS1
7. Calculation J-D21-1, Set Points Determination For High Range DW & Containment Radiation Monitors (D21 System)

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Loss of RPV inventory affecting fuel clad integrity with Containment challenged

EAL:

CG1.1 General Emergency

RPV level < -167 in. (TAF) for ≥ 30 min. (Note 1)

AND

Any Containment Challenge indication, Table C-2

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • Drywell or containment hydrogen concentration > 4% • UNPLANNED rise in containment pressure • Exceeding one or more Auxiliary Building Control MAX SAFE area radiation levels (EP-4)

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Containment Closure is established when either Primary or Secondary Containment integrity is established.

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.

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:

When RPV level drops below -167 in., core uncovery starts to occur (ref. 1).

Four conditions are associated with a challenge to Containment integrity:

- 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 the containment. However, containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that hydrogen concentration has exceeded the minimum necessary to support a hydrogen burn (4%). The Igniter System is designed to prevent hydrogen accumulation by locally burning hydrogen in a controlled manner as soon as the hydrogen enters the containment atmosphere and reaches the igniters. For high rates of hydrogen production, ignition occurs at the lowest concentration that can support ignition. Following ignition, hydrogen is consumed through formation of diffusion flames where the gas enters the containment, thus controlling hydrogen concentration at approximately 4% (ref. 2).
- Any UNPLANNED rise in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of CONTAINMENT CLOSURE capability. UNPLANNED containment pressure rise indicates CONTAINMENT CLOSURE cannot be assured and the containment cannot be relied upon as a barrier to fission product release.
- Secondary Containment radiation monitors should provide indication of a larger release that may be indicative of a challenge to CONTAINMENT CLOSURE. The MAX SAFE radiation levels are indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in EP-4, Auxiliary Building Control, (ref. 3).

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

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV 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 core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.

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

Reference(s):

1. 02-S-01-40 EP Technical Bases
2. BWROG Emergency Procedure and Severe Accident Guidelines, Revision 3, p. B-16-64
3. 05-S-01-EP-4, Auxiliary Building Control
4. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CG1.2 General Emergency

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

AND

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

- UNPLANNED rise in Suppression Pool level of sufficient magnitude to indicate core uncovery
- Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery

Containment/Drywell High Range Area Radiation Monitor (1D21-K648B-C) ≥ 100R/hr

AND

Any Containment Challenge indication, Table C-2

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • Drywell or containment hydrogen concentration > 4% • UNPLANNED rise in containment pressure • Exceeding one or more Auxiliary Building Control MAX SAFE area radiation levels (EP-4)

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Containment Closure is established when either Primary or Secondary Containment integrity is established.

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.

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

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

Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. RPV level in the Refueling mode is normally monitored using the Shutdown Range instrument which is re-spanned to indicate water level in the refuel cavity and the Core Plate d/p instrument which is re-spanned and re-scaled to indicate water level (ref. 1).

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications. Level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in Suppression Pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

In the Refueling Mode, as water level in the RPV lowers, the dose rate above the core will rise, with corresponding indications on area radiation monitors. 100 R/hr is used for this indication on Containment High Range Radiation Monitors (1D21-K648B and C). These detectors are located on the containment wall in a position to monitor the containment radiation environment above the refueling cavity elevation.

Four conditions are associated with a challenge to Containment integrity:

- 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 the containment. However, containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that hydrogen concentration has exceeded the minimum necessary to support a hydrogen burn (4%). The Igniter System is designed to prevent hydrogen accumulation by locally burning hydrogen in a controlled manner as soon as the hydrogen enters the containment atmosphere and reaches the igniters. For high rates of hydrogen production, ignition occurs at the lowest concentration that can support ignition. Following ignition, hydrogen is consumed through formation of diffusion flames where the gas enters the containment, thus controlling hydrogen concentration at approximately 4% (ref. 4).
- Any UNPLANNED rise in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of CONTAINMENT CLOSURE capability. UNPLANNED containment pressure rise indicates CONTAINMENT CLOSURE cannot be assured and the containment cannot be relied upon as a barrier to fission product release.
- Secondary Containment radiation monitors should provide indication of a larger release that may be indicative of a challenge to CONTAINMENT CLOSURE. The MAX SAFE radiation levels are indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in EP-4, Auxiliary Building Control, (ref. 5).

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

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV 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 core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.

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

The inability to monitor RPV 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 RPV.

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

Reference(s):

1. 03-1-01-5 Refueling
2. 04-1-02-1H13-P601 Alarm Response Instruction Panel 1H13-P601
3. 04-1-02-1H13-P680 Alarm Response Instruction Panel 1H13-P680
4. BWROG Emergency Procedure and Severe Accident Guidelines, Revision 3, p. B-16-64
5. 05-S-01-EP-4, Auxiliary Building Control
6. 06-IC-1D21-R-1002 Containment/Drywell High Range Area Radiation Monitor Calibration
7. NEI 99-01 CG1
8. Calculation J-D21-1, Set Points Determination For High Range DW & Containment Radiation Monitors (D21 System)

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of ESF AC Power

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

EAL:

CU2.1 Unusual Event

AC power capability, Table C-3, to DIV I and DIV II ESF 4.16 KV buses reduced to a single power source for ≥ 15 min. (Note 1)

AND

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table C-3 AC Power Sources
Offsite • ESF Transformer 11 • ESF Transformer 12 • ESF Transformer 21 **Onsite** • DIV I DG (DG 11) • DIV II DG (DG 12)

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, DEF - Defueled

Definition(s):

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

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

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

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

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

Basis:

The HPCS bus (DIV III) is not credited because it only supplies power to the HPCS pump and associated loads, not any long term decay heat removal systems. In particular, suppression pool cooling mechanisms would be essential subsequent to a station blackout.

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

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as an Alert because of the greater time available to restore another power source to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

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

- A loss of all offsite power with a concurrent failure of all but one emergency ESF power source (e.g., an onsite diesel generator).
- A loss of all offsite power and loss of all emergency ESF power sources (e.g., onsite diesel generators) with a single train of emergency ESF buses being back-fed from the unit main generator.
- A loss of emergency ESF power sources (e.g., onsite diesel generators) with a single train of emergency ESF buses being fed from an offsite power source.

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

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

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

Reference(s):

1. UFSAR Figure 8.1-001 Main One Line Diagram
2. UFSAR section 8.1 Electric Power Introduction
3. UFSAR section 8.3 Onsite Power
4. UFSAR section 8A Loss of all AC Power
5. 05-1-02-I-4 Loss of AC Power
6. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of ESF AC Power

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

EAL:

CA2.1	**Alert**
Loss of **all** offsite and **all** onsite AC power to DIV I and DIV II ESF 4.16 KV buses for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, DEF - Defueled

Definition(s):

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

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

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

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

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

Basis:

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. Mitigative strategies using other power sources (HPCS DIV III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as a Site Area Emergency because of the greater time available to restore an ESF bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition represents an actual or potential substantial degradation of the level of safety of the plant.

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

Escalation of the emergency classification level would be via IC CS1 or AS1.

This EAL is the cold condition equivalent of the hot condition EAL SS1.1.

Reference(s):

1. UFSAR Figure 8.1-001 Main One Line Diagram
2. UFSAR section 8.1 Electric Power Introduction
3. UFSAR section 8.3 Onsite Power
4. UFSAR section 8A Loss of all AC Power
5. 05-1-02-I-4 Loss of AC Power
6. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED rise in RCS temperature

EAL:

CU3.1	**Unusual Event**
UNPLANNED rise in RCS temperature to > 200°F	

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

*Containment Closure is established when either Primary or Secondary Containment integrity is establish*ed.

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

Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F) (ref. 1, 2). In the absence of reliable RCS temperature indication, classification is based on the concurrent loss of reactor vessel level indications per EAL CU3.2.

This IC addresses an UNPLANNED rise 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 also refer to EAL CA3.1.

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

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

During an outage, the level in the reactor vessel will normally be maintained at or above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel

flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid rise in reactor coolant temperature depending on the time after shutdown.

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

Reference(s):

1. Technical Specifications Table 1.1-1
2. 03-1-01-3 Plant Shutdown
3. 04-1-01-E12-2 Shutdown Cooling and Alternate Decay Heat Removal
4. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED rise in RCS temperature

EAL:

CU3.2	**Unusual Event**
Loss of **all** RCS temperature and RPV water level indication for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5- Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Containment Closure is established when either Primary or Secondary Containment integrity is established.

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:

This EAL addresses the inability to determine RCS temperature and level, and represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Director should also refer to EAL CA3.1.

This EAL reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

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

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

Reference(s):

1. 02-S-01-40 EP Technical Bases
2. Technical Specifications Table 1.1-1
3. 03-1-01-3 Plant Shutdown
4. 04-1-01-E12-2 Shutdown Cooling and Alternate Decay Heat Removal
5. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.1 Alert

UNPLANNED rise in RCS temperature to > 200°F for > Table C-4 duration (Note 1)

OR

UNPLANNED RPV pressure rise > 10 psig

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table C-4 RCS Heat-up Duration Thresholds

RCS Status	CONTAINMENT CLOSURE Status	Heat-up Duration
Intact	N/A	60 min.*
Not intact	established	20 min.*
	not established	0 min.

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Containment Closure is established when either Primary or Secondary Containment integrity is established.

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:

In the absence of reliable RCS temperature indication, classification should be based on the RCS pressure rise criteria when the RCS is intact in Mode 4 or based on time to boil data when in Mode 5 or the RCS is not intact in Mode 4.

This 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.

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

The RCS Heat-up Duration Thresholds table addresses a rise in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not intact. The 20-minute criterion was included to allow time for operator action to address the temperature rise.

The RCS Heat-up Duration Thresholds table also addresses a rise 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 rise without a substantial degradation in plant safety.

Finally, in the case where there is a rise in RCS temperature, the RCS is not intact and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because 1) the evaporated reactor coolant may be released directly into the containment atmosphere and subsequently to the environment, and 2) there is reduced reactor coolant inventory above the top of irradiated fuel.

The RCS pressure rise threshold provides a pressure-based indication of RCS heat-up in the absence of RCS temperature monitoring capability.

Escalation of the emergency classification level would be via IC CS1 or AS1.

Reference(s):

1. Technical Specifications Table 1.1-1
2. 03-1-01-3 Plant Shutdown
3. 04-1-01-E12-2 Shutdown Cooling and Alternate Decay Heat Removal
4. NEI 99-01 CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – Loss of Vital DC Power

Initiating Condition: Loss of Vital DC power for 15 minutes or longer

EAL:

CU4.1	**Unusual Event**
Indicated voltage is < 105 VDC on required vital 125 VDC buses 11DA and 11DB for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

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

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

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

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

Basis

Vital DC buses 11DA and 11DB feed the Division 1 and Division 2 loads respectively. The Division 1 and Division 2 batteries each have 61 cells with a design minimum of 1.72 volts/cell. These cell voltages yield minimum design bus voltages of 104.92 VDC (rounded to 105 VDC) (ref. 1, 2).

This IC addresses a loss of vital DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or refueling mode. In these modes, the core decay heat load has been significantly reduced, and coolant system temperatures and pressures are lower; these conditions raise the time available to restore a vital DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, "required" means the vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if

Train A is out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an Unusual Event. A loss of Vital DC power to Train A would not warrant an emergency classification.

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

Depending upon the event, escalation of the emergency classification level would be via IC CA1 or CA3, or an IC in Recognition Category A.

This EAL is the cold condition equivalent of the hot condition EAL SS2.1.

Reference(s):

1. Calculation No: EC-Q1111-14001 Station Division I Battery 1A3 and Division II Battery 1B3 Discharge Capacity during Extended Loss of AC Power
2. UFSAR 8.3.2.1.1 Station DC Power
3. NEI 99-01 CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.1 Unusual Event

Loss of **all** Table C-5 onsite communication methods

OR

Loss of **all** Table C-5 State and local agency communication methods

OR

Loss of **all** Table C-5 NRC communication methods

Table C-5 Communication Methods

System	Onsite	State/ Local	NRC
Station Radio System	X		
GGNS Plant Phone System	X		
Public Address System	X		
Emergency Notification System (ENS)			X
Commercial Telephone System		X	X
Satellite Phones		X	X
Operational Hotline		X	

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, DEF – Defueled

Definition(s):

None

Basis:

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 State and local agencies and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all State and local agencies of an emergency declaration. The State and local agencies referred to here are the Mississippi Emergency Management Agency, Claiborne County Civil Defense, Mississippi Highway Safety Patrol, Claiborne County Sheriff's Department, Louisiana Department of Environmental Quality, Tensas Parish Sheriff's Office, and the Louisiana Governor's Office of Homeland Security and Emergency Preparedness.

The third EAL condition addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

This EAL is the cold condition equivalent of the hot condition EAL SU7.1.

Reference(s):

1. GGNS Emergency Plan Section 7.5, Communications Systems
2. 04-S-01-R61-1 Plant Communications
3. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting Safety Systems

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

EAL:

CA6.1 Alert

The occurrence of **any** Table C-6 hazardous event

AND

Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode

AND EITHER:

- Event damage has caused indications of degraded performance to the second train of the SAFETY SYSTEM needed for the current operating mode
- Event damage has resulted in VISIBLE DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode

(Notes 9, 10)

Note 9: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then emergency classification is **not** warranted.

Note 10: If the hazardous event **only** resulted in VISIBLE DAMAGE, with **no** indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is **not** warranted.

Table C-6 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

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

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

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

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

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

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

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

VISIBLE DAMAGE - Damage to a SAFETY SYSTEM train 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 SAFETY SYSTEM train.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance;

commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. 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. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via IC CS1 or AS1.

This EAL is the cold condition equivalent of the hot condition EAL SA8.1.

Reference(s):

1. EP FAQ 2016-002
2. NEI 99-01 CA6

Category E – Independent Spent Fuel Storage Installation (ISFSI)

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

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

An Unusual Event is declared on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated.

The GGNS ISFSI is located wholly within the plant PROTECTED AREA. Therefore any security event related to the ISFSI is classified under Category H1 security event related EALs.

Category: E - ISFSI

Subcategory: Confinement Boundary

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Unusual Event

Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask (HI-STORM overpack) > **EITHER** of the following:

- 60 mrem/hr (gamma + neutron) on the top of the overpack
- 600 mrem/hr (gamma + neutron) on the side of the overpack (excluding inlet and outlet ducts)

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY- The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the GGNS ISFSI, Confinement Boundary is defined as the Holtec System Multi-Purpose Canister (MPC).

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

Basis:

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

The existence of "damage" is determined by radiological survey. The specified EAL threshold values correspond to 2 times the cask technical specification values. The technical specification (licensing bases document) multiple of "2 times", which is also used in Recognition Category A IC AU1, is used here to distinguish between non-emergency and emergency conditions (ref. 2). The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose

rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

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

Reference(s):

1. UFSAR 9.1.4.2.10.4 Storage of Fuel at the Independent Spent Fuel Storage Installation
2. GGNS HI-STORM 100 10 CFR 72.212 Evaluation Report Licensing Basis Document, Revision 10, Section 4.2.4 (Section 5.7) Radiation Protection Program
3. NEI 99-01 E-HU1

Category F – Fission Product Barrier Degradation

EAL Group: Hot Conditions (RCS temperature > 200°F); 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 (FCB): The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.

B. Reactor Coolant System Barrier (RCB): The RCS Barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping out to and including the isolation valves.

C. Containment Barrier (CNB): The Containment Barrier includes the drywell, the containment, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. Containment Barrier thresholds are used as criteria for escalation of the Emergency Classification Level (ECL) from an 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. "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

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

Site Area Emergency:
Loss or potential loss of any two barriers

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

The logic used for emergency classification based on fission product barrier monitoring should reflect the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier.

- Unusual Event ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.
- For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC AG1 has been exceeded.
- The fission product barrier thresholds specified within a scheme reflect plant-specific GGNS 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, an interfacing system, or outside of the containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered to be RCS leakage.
- At the Site Area Emergency level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

Category: F - Fission Product Barrier Degradation

Subcategory: N/A

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

EAL:

FA1.1 Alert

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds, bases and references.

At the Alert classification level, Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1

Reference(s):

1. NEI 99-01 FA1

Category: F - Fission Product Barrier Degradation

Subcategory: N/A

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

EAL:

FS1.1	**Site Area Emergency**
Loss or potential loss of **any** two barriers (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 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.

Reference(s):

1. NEI 99-01 FS1

Category: F - Fission Product Barrier Degradation

Subcategory: N/A

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

EAL:

FG1.1 General Emergency

Loss of **any** two barriers

AND

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 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

Reference(s):

1. NEI 99-01 FG1

Table F-1 Fission Product Barrier Threshold Matrix & Bases

Table F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Containment). The table is structured so that each of the three barriers occupies adjacent columns. Each fission product barrier column is further divided into two columns; one for Loss thresholds and one for Potential Loss thresholds.

The first column of the table (to the left of the Fuel Clad Loss column) lists the categories (types) of fission product barrier thresholds. The fission product barrier categories are:

A. RPV Water Level

B. RCS Leak Rate

C. Containment Conditions

D. Containment Radiation / RCS Activity

E. Containment Integrity or Bypass

F. Emergency Director Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word "None" is entered in the cell.

Thresholds are assigned sequential numbers within each barrier column beginning with number one (ex., FCB1, FCB2...FCB6).

If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of Table F-1, locates the likely category and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine if a threshold has been exceeded. If a threshold has not been exceeded, the EAL-user proceeds to the next likely category and continues review of the thresholds in the new category

If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded, only that one barrier is lost or potentially lost. The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if containment radiation is sufficiently high, a Loss of the Fuel

Clad and RCS Barriers and a Potential Loss of the Containment Barrier can occur. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, and FA1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad Barrier threshold bases appear first, followed by the RCS Barrier and finally the Containment Barrier threshold bases. In each barrier, the bases are given according to category Loss followed by category Potential Loss beginning with Category A, then B,..., F.

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad Barrier (FCB)		Reactor Coolant System Barrier (RCB)		Containment Barrier (CNB)	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A **RPV Water Level**	FCB1 SAP entry is required	FCB2 RPV water level **cannot** be restored and maintained > -167 in. (TAF) or **cannot** be determined	RCB1 RPV water level **cannot** be restored and maintained > -167 in. (TAF) or **cannot** be determined	None	None	CNB1 SAP entry is required
B **RCS Leak Rate**	None	None	RCB2 UNISOLABLE break in **any** of the following: • Main steam line • RCIC steam Line • RWCU • Feedwater • HPCS RCB3 Emergency Depressurization is required	RCB4 UNISOLABLE primary system leakage that results in exceeding **EITHER**: • One or more EP-4 radiation Operating Limits • One or more EP-4 area temperature Operating Limits	CNB2 UNISOLABLE primary system leakage that results in exceeding **EITHER**: • One or more EP-4 MAX SAFE area radiation levels • One or more EP-4 MAX SAFE area temperature levels	None
C **CTMT Conditions**	None	None	RCB5 Drywell pressure > 1.23 psig due to RCS leakage	None	CNB3 UNPLANNED rapid drop in containment pressure following containment pressure rise CNB4 Containment pressure response **not** consistent with LOCA conditions	CNB5 Containment pressure > 15 psig CNB6 Drywell or containment hydrogen concentration > 4% CNB7 Parameters **cannot** be restored and maintained within the safe zone of the HCTL curve (EP Figure 1)
D **CTMT Rad / RCS Activity**	FCB3 Containment radiation (RITS-K648B or C) > 400 R/hr FCB4 Primary coolant activity > 300 µCi/gm dose equivalent I-131	None	RCB6 Drywell radiation (RITS-K648A or D) > 100 R/hr	None	None	CNB8 Containment radiation (RITS-K648B or C) > 7000 R/hr
E **CTMT Integrity or Bypass**	None	None	None	None	CNB9 UNISOLABLE direct downstream pathway to the environment exists after Containment isolation signal CNB10 Intentional Containment venting per EPs	None
F **Emergency Director Judgment**	FCB5 **Any** condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier	FCB6 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier	RCB7 **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	RCB8 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	CNB11 **Any** condition in the opinion of the Emergency Director that indicates loss of the Containment barrier	CNB12 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Loss

Threshold:

FCB1
SAP entry is required

Definition(s):

None

Basis:

Emergency Procedures (EPs) specify entry to the Severe Accident Procedures (SAPs) when core cooling is severely challenged. These EPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined (ref. 1, 2).

The EP conditions represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

This threshold is also a Potential Loss of the Containment barrier (CNB1). Since SAP entry occurs after core uncovery has occurred a Loss of the RCS barrier exists (RCB1). SAP entry, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

The Loss threshold represents the EOP requirement for entry into the SAPs. This is identified in the BWROG EPGs/SAGs when adequate core cooling cannot be assured.

Reference(s):

1. 05-S-01-EP-2 RPV Control
2. 05-S-01-EP-5 RPV Flooding
3. EP FAQ 2015-004
4. NEI 99-01, RPV Water Level Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

FCB2

RPV water level **cannot** be restored and maintained > -167 in. (TAF) or **cannot** be determined

Definition(s):

None

Basis:

An RPV water level instrument reading of -167 in. indicates RPV level is at the top of active fuel (TAF) (ref. 1). When RPV level is at or above the TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EPs specify alternate, more extreme, RPV water level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops to TAF, the level is indicative of a challenge to core cooling and the Fuel Clad barrier.

When RPV water level cannot be determined, EPs require entry to EP-5, RPV Flooding. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained (ref. 2). When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EP-5 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in scram-failure events). If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the fuel clad barrier exists.

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, EALs SA6.1 or SS6.1 will dictate the need for emergency classification.

This water level corresponds to the top of the active fuel and is used in the EOPs to indicate a challenge to core cooling.

The RPV water level threshold is the same as RCS barrier Loss threshold RCB1. Thus, this threshold indicates a Potential Loss of the Fuel Clad barrier and a Loss of the RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

This threshold is considered to be exceeded when, as specified in the site-specific EOPs, RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this Fuel Clad barrier Potential Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

The term "cannot be restored and maintained above" means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value cannot be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active fuel, but does not permit extended operation below the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

Since the loss of ability to determine if adequate core cooling is being provided presents a significant challenge to the fuel clad barrier, a potential loss of the fuel clad barrier is specified.

Reference(s):

1. 05-S-01-EP-2 RPV Control
2. 05-S-01-EP-5 RPV Flooding
3. 05-S-01-EP-2A ATWS RPV Control
4 NEI 99-01 RPV Water Level Potential Loss 2.A

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

FCB3
Containment radiation (RITS-K648B or C) > 400 R/hr

Definition(s):

None

Basis:

The containment radiation monitor reading (425 R/hr rounded to 400 R/hr for readability) corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300 µCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to 1.6% fuel clad damage (ref. 1). 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 RCB6 since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the ECL to a Site Area Emergency.

There is no Fuel Clad barrier Potential Loss threshold associated with CTMT Radiation / RCS Activity.

Reference(s):

1. XC-Q1D21-17001 Grand Gulf Nuclear Station (GGNS) Containment Radiation EAL Threshold Values
2. 04-1-01-D21-1 SOI Area Radiation Monitoring System
3. NEI 99-01 Primary Containment Radiation Fuel Clad Loss 4.A

Barrier: Fuel Clad

Category: D. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

FCB4
Coolant activity > 300 µCi/gm dose equivalent I-131

Definition(s):

None

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.

It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications.

There is no Fuel Clad barrier Potential Loss threshold associated with CTMT Radiation / RCS Activity.

Reference(s):

1. NEI 99-01 RCS Activity Fuel Clad Loss 1.A

Barrier: Fuel Clad

Category: D. CTMT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

FCB5 **Any** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad Barrier

Definition(s):

None

Basis:

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

Reference(s):

1. NEI 99-01 Emergency Director Judgment Fuel Clad Loss 6.A

Barrier: Fuel Clad

Category: F. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

FCB6
Any condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad Barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that are to 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.

Reference(s):

1. NEI 99-01 Emergency Director Judgment Potential Fuel Clad Loss 6.A

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

RCB1

RPV water level **cannot** be restored and maintained > -167 in. (TAF) or **cannot** be determined

Definition(s):

None.

Basis:

An RPV water level instrument reading of -167 in. indicates level is at the top of active fuel (TAF) (ref. 1). TAF is significantly lower than the normal operating RPV level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Containment barriers, and initiation of all ECCS. If RPV water level cannot be maintained above TAF, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the lowering level trend. The cause of the loss of RPV inventory is therefore assumed to be a LOCA. By definition, a LOCA event is a Loss of the RCS barrier.

When RPV water level cannot be determined, EOPs require entry to EP-5, RPV Flooding. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained (ref. 2). The instructions in EP-5 specify emergency depressurization of the RPV, which is defined to be a Loss of the RCS barrier (RCS Loss RCB3).

Note that EP-2A, ATWS RPV Control, may require intentionally lowering RPV water level to -167 in. and control level between the Minimum Steam Cooling RPV Water Level (MSCRWL) and the top of active fuel (ref. 3). Under these conditions, a high-power ATWS event exists and requires at least a Site Area Emergency classification in accordance with the System Malfunction - RPS Failure EALs.

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, EALs SA6.1 or SS6.1 will dictate the need for emergency classification.

This water level corresponds to the top of active fuel and is used in the EOPs to indicate a challenge to core cooling.

The RPV water level threshold is the same as Fuel Clad barrier Potential Loss threshold FCB2. Thus, this threshold indicates a Loss of the RCS barrier and Potential Loss of the Fuel Clad barrier and that appropriately escalates the emergency classification level to a Site Area Emergency.

This threshold is considered to be exceeded when, as specified in the site-specific EOPs, RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this RCS barrier Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

The term, "cannot be restored and maintained above," means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value cannot be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active fuel, but does not permit extended operation beyond the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

There is no RCS barrier Potential Loss threshold associated with RPV Water Level.

Reference(s):

1. 05-S-01-EP-2 RPV Control
2. 05-S-01-EP-5 RPV Flooding
3. 05-S-01-EP-2A ATWS RPV Control
4. NEI 99-01 RPV Water Level RCS Loss 2.A

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

RCB2

UNISOLABLE break in **any** of the following:

- Main steam line
- RCIC steam line
- RWCU
- Feedwater
- HPCS

Definition(s):

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

Basis:

Failure to isolate the leak, within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

The conditions of this threshold include required containment isolation failures allowing a flow path to the environment. A release pathway outside containment exists when flow is not prevented by downstream isolations. In the case of a failure of both isolation valves to close but in which no downstream flowpath exists, emergency declaration under this threshold would not be required. Similarly, if the emergency response requires the normal process flow of a system outside containment (e.g., EP requirement to bypass MSIV low RPV water level interlocks and maintain the main condenser as a heat sink using main turbine bypass valves), the threshold is not met. The combination of these threshold conditions represent the loss of both the RCS and Containment (see Loss CNB9) barriers and justifies declaration of a Site Area Emergency (i.e., Loss or Potential Loss of any two barriers).

Even though RWCU and Feedwater systems do not contain steam, they are included in the list because an UNISOLABLE break could result in the high-pressure discharge of fluid that is flashed to steam from relatively large volume systems directly connected to the RCS.

Even though the High Pressure Core Spray (HPCS) injects into the RCS, it is included in this EAL due to the potential for an inter-system loss of coolant back flowing from the discharge

lines (via failed isolation valves and check valves) and out through a break in the piping. A HPCS failure that does not result in back flow of RCS and out through a break should not be considered as meeting the EAL threshold.

Large high-energy lines that rupture outside containment can discharge significant amounts of inventory and jeopardize the pressure-retaining capability of the RCS until they are isolated. If it is determined that the ruptured line cannot be promptly isolated, remotely or locally,, the RCS barrier Loss threshold is met.

Reference(s):

1. NEI 99-01 RCS Leak Rate RCS Loss 3.A

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

RCB3 Emergency Depressurization is required

Definition(s):

None

Basis:

Emergency Depressurization in accordance with the EOPs (ref. 1, 2) is indicative of a loss of the RCS barrier. If Emergency RPV Depressurization is performed, the plant operators are directed to open safety relief valves (SRVs). Even though the RCS is being vented into the suppression pool, a Loss of the RCS barrier exists due to the diminished effectiveness of the RCS to retain fission products within its boundary.

EP-2 RPV Control - Emergency Depressurization allows terminating the depressurization if necessary to maintain RCIC as an injection source. This would require closing the SRVs. Even though the SRVs may be reclosed, this threshold is still met due to the requirement for an Emergency Depressurization having been met (ref. 2).

Reference(s):

1. 05-S-01-EP-2 RPV Control - Emergency Depressurization
2. EP FAQ 2015-003
3. NEI 99-01 RCS Leak Rate RCS Loss 3.B

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

RCB4

UNISOLABLE primary system leakage that results in exceeding **EITHER**:

- One or more EP-4 area radiation Operating Limits
- One or more EP-4 area temperature Operating Limits

Definition(s):

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

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

Basis:

Failure to isolate the leak, within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

The presence of elevated general area temperatures or radiation levels in the Secondary Containment may be indicative of UNISOLABLE primary system leakage outside the containment. The EP-4 entry condition values define this RCS threshold because they are the Operating Limits (maximum normal operating values) and signify the onset of abnormal system operation. When parameters reach this level, equipment failure or mis-operation may be occurring. Elevated parameters may also adversely affect the ability to gain access to or operate equipment within the affected area. The locations into which the primary system discharge is of concern correspond to the areas addressed in EP-4, Auxiliary Building Control (ref. 1).

In general, multiple indications should be used to determine if a primary system is discharging outside containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the Auxiliary Building since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room FLOODING, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the Secondary Containment.

Potential loss of RCS based on primary system leakage outside the containment is determined from EOP temperature or radiation EP-4 Operating Limits (Max Normal Operating values) in

areas such as main steam line tunnel, RCIC, etc., which indicate a direct path from the RCS to areas outside containment.

An EP-4 Operating Limit value is the highest value of the identified parameter expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

The indicators reaching the threshold barriers and confirmed to be caused by RCS leakage from a primary system warrant an Alert classification. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a reduction in the steam or water being discharged through an unisolated break in the system.

An UNISOLABLE leak which is indicated by EP-4 Operating Limit values escalates to a Site Area Emergency when combined with Containment Barrier Loss thresholds CNB 2 or CNB9 (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

Reference(s):

1. 05-S-01-EP-4 Auxiliary Building Control
2. NEI 99-01 RCS Leak Rate RCS Potential Loss 3.A

Barrier: Reactor Coolant System

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

RCB5

Drywell pressure > 1.23 psig due to RCS leakage

Definition(s):

None

Basis:

The drywell high pressure scram setpoint is an entry condition to EP-2, RPV Control, and EP-3, Containment Control (ref. 1, 2). Normal containment pressure control functions (e.g., operation of drywell and containment cooling, vent using containment vessel purge, etc.) are specified in EP-3 in advance of less desirable but more effective functions (e.g., operation of containment sprays, etc.).

In the design basis, containment pressures above the drywell high pressure scram setpoint are assumed to be the result of a high-energy release into the containment for which normal pressure control systems are inadequate or incapable of reversing the rising pressure trend. Pressures of this magnitude, however, can be caused by non-LOCA events such as a loss of drywell cooling or inability to control containment vent/purge (ref. 3).

The threshold phrase "...due to RCS leakage" focuses the barrier failure on the RCS instead of the non-LOCA malfunctions that may adversely affect containment pressure. Drywell pressure greater than 1.23 psig with corollary indications (e.g., drywell temperature, indications of loss of RCS inventory) should therefore be considered a Loss of the RCS barrier. Loss of drywell cooling that results in pressure greater than 1.23 psig should not be considered an RCS barrier Loss.

The 1.23 psig value is the drywell high pressure setpoint which indicates a LOCA by automatically initiating the ECCS or equivalent makeup system.

There is no RCS barrier Potential Loss threshold associated with CTMT Conditions.

Reference(s):

1. 05-S-01-EP-2 RPV Control
2. 05-S-01-EP-3 Containment Control
3. UFSAR Section 6.2.1, Containment Functional Design
4. NEI 99-01 Primary Containment Pressure RCS Loss 1.A

Barrier: Reactor Coolant System

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CTMT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

RCB6
Drywell radiation (RITS-K648A or D) > 100 R/hr

Definition(s):

None

Basis:

The drywell radiation monitor reading (150 R/hr rounded to 100 R/hr for readability) corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits (ref. 1). This value is lower than that specified for Fuel Clad Barrier Loss threshold FCB3 since it indicates a loss of the RCS Barrier only.

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

Reference(s):

1. XC-Q1D21-17001 Grand Gulf Nuclear Station (GGNS) Containment Radiation EAL Threshold Values
2. 04-1-01-D21-1 SOI Area Radiation Monitoring System
3. NEI 99-01 Primary Containment Radiation RCS Loss 4.A

Barrier: Reactor Coolant System

Category: D. CTMT Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

RCB7

Any condition in the opinion of the Emergency Director that indicates loss of the RCS Barrier

Definition(s):

None

Basis:

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

Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Loss 6.A

Barrier: Reactor Coolant System

Category: F. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

RCB8 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS Barrier

Definition(s):

None

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.

Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Potential Loss 6.A

Barrier: Containment

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

CNB1
SAP entry is required

Definition(s):

None

Basis:

EPs specify entry to the SAPs when core cooling is severely challenged. These EPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined (ref. 1, 2).

The EP conditions represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

This threshold is also a Loss of the Fuel Clad barrier (Loss FCB1). Since SAP entry occurs after core uncovery has occurred a Loss of the RCS barrier exists (Loss RCB1). SAP entry, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

The Potential Loss threshold is identical to the Fuel Clad Loss RPV Water Level threshold FCB1. The Potential Loss requirement for entry into the SAGs indicates adequate core cooling cannot be assured and that core damage is possible. BWR EPGs/SAGs specify the conditions when the EPGs are exited and SAGs are entered. Entry into SAGs is a logical escalation in response to the inability to assure adequate core cooling.

PRA studies indicate that the condition of this Potential Loss threshold could be a core melt sequence which, if not corrected, could lead to RPV failure and greater potential for containment failure. In conjunction with the RPV water level Loss thresholds in the Fuel Clad and RCS barrier columns, this threshold results in the declaration of a General Emergency.

There is no Containment barrier Loss threshold associated with RPV Water Level.

Reference(s):

1. 05-S-01-EP-2 RPV Control
2. 05-S-01-EP-5 RPV Flooding
3. EP FAQ 2015-004
4. NEI 99-01 RPV Water Level PC Potential Loss 2.A

Barrier: Containment

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

CNB2

UNISOLABLE primary system leakage that results in exceeding **EITHER**:

- One or more EP-4 MAX SAFE area radiation levels
- One or more EP-4 MAX SAFE area temperature levels

Definition(s):

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

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

Basis:

Failure to isolate the leak, within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

The presence of elevated general area temperatures or radiation levels in the secondary containment may be indicative of UNISOLABLE primary system leakage outside the containment. The MAX SAFE values define this Containment barrier threshold because they are indicative of problems in the Secondary Containment that are spreading and pose a threat to achieving a safe plant shutdown. This threshold addresses problematic discharges outside containment that may not originate from a high-energy line break. The locations into which the primary system discharge is of concern correspond to the areas addressed in EP-4, Auxiliary Building Control (ref. 1).

In general, multiple indications should be used to determine if a primary system is discharging outside containment. For example, a high area temperature condition does not necessarily indicate that a primary system is discharging into the secondary containment since this may be caused by a fire or loss of area cooling. Conversely, a high area temperature condition in conjunction with other indications (e.g. room FLOODING, high area radiation levels, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the secondary containment.

The Max Safe area temperature values and the Max Safe area radiation values are each the highest value of these parameters at which neither: (1) equipment necessary for the safe shutdown of the plant will fail, nor (2) personnel access necessary for the safe shutdown of the

plant will be precluded. EOPs utilize these temperatures and radiation levels to establish conditions under which RPV depressurization is required.

There is no Containment barrier Potential Loss threshold associated with RCS Leak Rate.

Reference(s):

1. 05-S-01-EP-4 Auxiliary Building Control
1. NEI 99-01 RCS Leak Rate PC Loss 3.C

Barrier: Containment

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

CNB3

UNPLANNED rapid drop in containment pressure following containment pressure rise

Definition(s):

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

Basis:

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

This threshold relies on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

Reference(s):

1. NEI 99-01 Primary Containment Conditions PC Loss 1.A

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

CNB4

Containment pressure response **not** consistent with LOCA conditions

Definition(s):

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

Basis:

Containment pressure should rise as a result of mass and energy release into the containment from a LOCA. Thus, containment pressure not rising under these conditions indicates a loss of containment integrity.

These thresholds rely on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

Reference(s):

1. USAR Table 6.2-5, Summary of Short Term Containment Responses to Recirculation Line and Main Steam Line Breaks
2. UFSAR Table 6.2-13, Maximum Calculated Accident for Containment Design
3. NEI 99-01 Primary Containment Conditions PC Loss 1.B

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

CNB5 Containment pressure > 15 psig

Definition(s):

None

Basis:

When the containment pressure exceeds the maximum allowable value (15 psig) (ref. 1), containment venting may be required even if offsite radioactivity release rate limits will be exceeded (ref. 2). This pressure is based on the containment design pressure as identified in the accident analysis. If this threshold is exceeded, a challenge to the containment structure has occurred because assumptions used in the accident analysis are no longer valid and an unanalyzed condition exists. This constitutes a Potential Loss of the Containment barrier even if a containment breach has not occurred.

The threshold pressure is the containment internal design pressure. Structural acceptance testing demonstrates the capability of the containment to resist pressures greater than the internal design pressure. A pressure of this magnitude is greater than those expected to result from any design basis accident and, thus, represent a Potential Loss of the Containment barrier.

Reference(s):

1. UFSAR Table 6.2-13, Maximum Calculated Accident for Containment Design
2. 05-S-01-EP-3 Containment Control
3. NEI 99-01, Primary Containment Conditions PC Potential Loss 1.A

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

CNB6

Drywell or containment hydrogen concentration > 4%

Definition(s):

None

Basis:

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 the containment. However, containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that hydrogen concentration has exceeded the minimum necessary to support a hydrogen burn (4%). The Igniter System is designed to prevent hydrogen accumulation by locally burning hydrogen in a controlled manner as soon as the hydrogen enters the containment atmosphere and reaches the igniters. For high rates of hydrogen production, ignition occurs at the lowest concentration that can support ignition. Following ignition, hydrogen is consumed through formation of diffusion flames where the gas enters the containment, thus controlling hydrogen concentration at approximately 4% (ref. 1).

If hydrogen concentration reaches or exceeds the lower flammability limit, as defined in plant EOPs, in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside the containment, loss of the Containment barrier could occur.

Reference(s):

1. 02-S-01-40 EP Technical Bases (EP-3 step H-3)
2. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.B

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

CNB7

Parameters **cannot** be restored and maintained within the safe zone of the HCTL curve (EP Figure 1)

Definition(s):

None

Basis:

The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool temperature from which Emergency RPV Depressurization will not raise:

- Suppression chamber temperature above the maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized,

  OR

- Suppression chamber pressure above Primary Containment Pressure Limit, while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

The HCTL is a function of RPV pressure, suppression pool temperature and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant and therefore, the inability to maintain plant parameters below the limit constitutes a potential loss of containment.

The term "**cannot** be restored and maintained within" means the parameter value(s) is not able to be brought within the specified limit. The determination requires an evaluation of system performance and availability in relation to the parameter value and trend. A threshold prescribing declaration when a threshold value cannot be restored and maintained within a specified limit does not require immediate action simply because the current value is outside the limit, but does not permit extended operation outside the limit; the threshold must be considered reached as soon as it is apparent that operation within the limit cannot be attained.

Reference(s):

1. 05-S-01-EP-3 Containment Control
2. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.C

Barrier: Containment

Category: D. CTMT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: D. CTMT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

CNB8
Containment radiation (RITS-K648B or C) > 7,000 R/hr

Definition(s):

None

Basis:

In order to reach this Containment barrier Potential Loss threshold, a loss of the RCS barrier (Loss RCB6) and a loss of the Fuel Clad barrier (Loss FCB3) have already occurred. This threshold, therefore, represents a General Emergency classification.

The containment radiation monitor reading (7,350 R/hr rounded to 7,000 R/hr for readability) corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that 20% of the fuel cladding has failed (ref. 1). 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 emergency classification level to a General Emergency.

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

Reference(s):

1. XC-Q1D21-17001 Grand Gulf Nuclear Station (GGNS) Containment Radiation EAL Threshold Values
2. 04-1-01-D21-1 SOI Area Radiation Monitoring System
3. NEI 99-01 NEI 99-01 Primary Containment Radiation Potential Loss 4.A

Barrier: Containment

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

CNB9
UNISOLABLE direct downstream pathway to the environment exists after Containment isolation signal

Definition(s):

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

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

Basis:

Failure to isolate the leak, within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway directly to the environment. The concern is the UNISOLABLE open pathway to the environment. A failure of the ability to isolate any one line indicates a breach of containment integrity.

This threshold also applies to a containment bypass due to a HPCS or LPCS line break outside containment with injection check valve failure allowing an UNISOLABLE direct pathway for RCS release to the environment.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems or minor release pathways, such as instrument lines, not protected by the Primary Containment Isolation System (PCIS). Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include UNISOLABLE main steam line or RCIC steam line breaks, UNISOLABLE RWCU system breaks, and UNISOLABLE containment atmosphere vent paths. If the main condenser is available with an UNISOLABLE main steam line, there may be releases through the steam jet air ejectors and gland seal exhausters. These pathways are monitored, however, and do not meet the intent of a nonisolable release path to the

environment. These minor releases are assessed using the Category A, Abnormal Rad Release / Rad Effluent, EALs.

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.

EP-3 Containment Control, may specify containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1). Under these conditions with a VALID containment isolation signal, the Containment barrier should be considered lost.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable containment leakage through various penetrations or system components. Minor releases may also occur if a containment isolation valve(s) fails to close but the containment atmosphere escapes to an enclosed system. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category A ICs.

There is no Containment barrier Potential Loss threshold associated with CTMT Integrity or Bypass.

Reference(s):

1. 05-S-01-EP-3 Containment Control
2. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.A

Barrier: Containment

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

CNB10
Intentional Containment venting per EPs

Definition(s):

None

Basis:

EP-3, Containment Control, may specify containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded. The threshold is met when the operator begins venting the containment in accordance with Attachment 13, not when actions are taken to bypass interlocks prior to opening the vent valves (ref. 1).

Intentional venting of containment for containment pressure or combustible gas control to the secondary containment and/or the environment is a Loss of the Containment. Venting for containment pressure control when not in an accident situation (e.g., to control pressure below the drywell high pressure scram setpoint) does not meet the threshold condition.

There is no Containment barrier Potential Loss threshold associated with CTMT Integrity or Bypass.

Reference(s):

1. 05-S-01-EP-3 Containment Control
2. NEI 99-01 CTMT Integrity or Bypass Containment Loss 3.B

Barrier: Containment

Category: E. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

CNB11

Any condition in the opinion of the Emergency Director that indicates loss of the Containment Barrier

Definition(s):

None

Basis:

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

Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

Barrier: Containment

Category: E. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

CNB12

Any condition in the opinion of the Emergency Director that indicates potential loss of the Containment Barrier

Definition(s):

None

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.

Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Potential Loss 6.A

Category H – Hazards and Other Conditions Affecting Plant Safety

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

Hazards are non-plant, system-related events that can directly or indirectly affect plant operation, reactor plant safety or personnel safety.

1. Security

Unauthorized entry attempts into the PROTECTED AREA, bomb threats, sabotage attempts, and actual security compromises threatening loss of physical control of the plant.

2. Seismic Event

Natural events such as earthquakes have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety.

3. Natural or 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 can pose significant hazards to personnel and reactor safety. Appropriate for classification are FIRES within the plant PROTECTED AREA or which may affect operability of equipment needed for safe shutdown

5. Hazardous Gas

Toxic, corrosive, asphyxiant or flammable gas leaks can affect normal plant operations or preclude access to plant areas required to safely shutdown the plant.

6. Control Room Evacuation

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

7. Emergency Director 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.

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1 Unusual Event

A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by GGNS Security Shift Supervision

OR

Notification of a credible security threat directed at the site

OR

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

Mode Applicability:

All

Definition(s):

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 GGNS 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 GGNS. 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 (OCA) - For the purposes of classification, the Security area between the OCA detection fence and the PROTECTED AREA boundary known as the Security Owner Controlled Area (SOCA) in the GGNS Emergency Plan.

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

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

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

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

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

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

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

Basis:

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 and HS1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program].*

The first threshold references the Security Shift Supervision because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.39 information.

The second threshold addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the Security Plan for GGNS.

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 11-S-82-1 Security Contingency Events (ref. 2).

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan for GGNS (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

Reference(s):

1. GGNS Security Plan
2. 11-S-82-1 Security Contingency Events
3. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HA1.1 Alert

A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by GGNS Security Shift Supervision

OR

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

Mode Applicability:

All

Definition(s):

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 GGNS 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 GGNS. 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.

OWNER CONTROLLED AREA - For the purposes of classification, the Security area between the OCA detection fence and the PROTECTED AREA boundary known as the Security Owner Controlled Area (SOCA) in the GGNS Emergency Plan.

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program].*

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of Offsite Response Organizations (OROs), allowing them to be better prepared should it be necessary to consider further actions.

This EAL 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.

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

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 11-S-82-1 Security Contingency 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 for GGNS (ref. 1).

Escalation of the emergency classification level would be via IC HS1.

Reference(s):

1. GGNS Security Plan
2. 11-S-82-1 Security Contingency Events
3. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA

EAL:

HS1.1 Site Area Emergency

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

Mode Applicability:

All

Definition(s):

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 GGNS 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 GGNS. 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.

OWNER CONTROLLED AREA - For the purposes of classification, the Security area between the OCA detection fence and the PROTECTED AREA boundary known as the Security Owner Controlled Area (SOCA) in the GGNS Emergency Plan.

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the PROTECTED AREA. This event will require rapid response and assistance due to the possibility for damage to plant equipment.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1, 2).

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Site Area Emergency declaration will mobilize Offsite Response Organization (ORO) resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

This EAL 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 Security Plan for GGNS (ref. 1).

Reference(s):

1. GGNS Security Plan
2. 11-S-82-1 Security Contingency Events
3. NEI 99-01 HS1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE levels

EAL:

HU2.1 Unusual Event

Seismic event > OBE as indicated by annunciation of **EITHER** of the following on SH13P856:

- Containment Operating Basis Earthquake (P856-1A-A3)
- Drywell Operating Basis Earthquake (P856-1A-A5)

Mode Applicability:

All

Definition(s):

None

Basis:

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE). An earthquake greater than an OBE but less than a Safe Shutdown Earthquake (SSE) should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., perform walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event. The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the U.S. Geological Survey (USGS), check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

To avoid inappropriate emergency classification resulting from spurious actuation of the seismic instrumentation or felt motion not attributable to seismic activity, an offsite agency (USGS, National Earthquake Information Center (NEIC)) can confirm that an earthquake has occurred in the area of the plant. Such confirmation should not, however, preclude a timely emergency declaration based on receipt of the OBE alarm. If requested, provide the analyst

with the following GGNS coordinates: **32° 0' 27" north latitude, 91° 2' 53" west longitude** (ref. 2). Alternatively, near real-time seismic activity can be accessed via the NEIC website.

Reference(s):

1. 05-S-02-VI-3 Earthquake
2. UFSAR 2.1.1 Site Location and Description
3. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.1	**Unusual Event**
A tornado strike within the PROTECTED AREA	

Mode Applicability:

All

Definition(s):

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

Basis:

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

This EAL addresses a tornado striking (touching down) within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories A, F, S or C.

If damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under EAL CA6.1 or SA8.1.

A tornado striking (touching down) within the PROTECTED AREA warrants declaration of an Unusual Event regardless of the measured wind speed at the meteorological tower. A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

Reference(s):

1. 05-1-02-VI-2 Hurricanes, Tornados and Severe Weather
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.2 Unusual Event

Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component required by Technical Specifications for the current operating mode

Mode Applicability:

All

Definition(s):

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

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

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

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

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

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

Basis:

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

This EAL addresses FLOODING of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

Escalation of the emergency classification level would be based on ICs in Recognition Categories A, F, S or C.

Refer to EAL CA6.1 or SA8.1 for internal FLOODING affecting more than one SAFETY SYSTEM train.

Reference(s):

1. 05-1-02-VI-1 Flooding
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.3	**Unusual Event**
Movement of personnel within the PROTECTED AREA is IMPEDED due to an event external to the PROTECTED AREA involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)	

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

Basis:

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

This EAL addresses a hazardous materials event originating at a location outside the PROTECTED AREA and of sufficient magnitude to IMPEDE the movement of personnel within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories A, F, S or C.

Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.4 Unusual Event

A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)

Note 7: This EAL does **not** apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Mode Applicability:

All

Definition(s):

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.

Basis:

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

This EAL addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site FLOODING caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.

This EAL is not intended to apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the FLOODING around the Cooper Station during the Midwest floods of 1993, or the FLOODING around Ft. Calhoun Station in 2011.

Escalation of the emergency classification level would be based on ICs in Recognition Categories A, F, S or C.

Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.1 Unusual Event

A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):

- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

The FIRE is located within **any** Table H-1 area

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table H-1 Fire Areas
• Unit 1 Containment • Unit 1 Auxiliary Building • Unit 1 Turbine Building • Control Building • Diesel Generator Rooms • SSW Pump & Valve Rooms

Mode Applicability:

All

Definition(s):

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

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

Basis:

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.

The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

The 15 minute requirement begins with a credible notification that a FIRE is occurring, or receipt of multiple VALID fire detection system alarms or field validation of a single fire alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field.

Table H-1 Fire Areas are those areas that contain equipment necessary for safe operation and shutdown of the plant (ref. 1, 2).

Reference(s):

1. 05-S-02-V-1 Response to Fires
2. 10-S-03-2 Response to Fires
3. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.2 Unusual Event

Receipt of a single fire alarm (i.e., **no** other indications of a FIRE) (Note 11)

AND

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

AND

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Note 11: During Modes 1 and 2, HU4.2 is not applicable to a single fire alarm in the containment or drywell.

Table H-1 Fire Areas
• Unit 1 Containment • Unit 1 Auxiliary Building • Unit 1 Turbine Building • Control Building • Diesel Generator Rooms • SSW Pump & Valve Rooms

Mode Applicability:

All – This MUST be all modes because only the containment and drywell are excluded in modes 1 and 2, but other areas are not **Definition(s):**

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

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

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

This EAL is not applicable for the containment or drywell in Modes 1 and 2. The air flow design and TS requirements for operation of Containment Fan Coolers and the drywell cooling system are such that multiple detectors would be expected to alarm for a fire in the containment or drywell. A fire in the containment or drywell in these modes would therefore be classified under EAL HU4.1.

If an actual FIRE is verified by a report from the field, then HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

<u>Basis-Related Requirements from Appendix R</u>

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."
>
> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.
>
> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in HU4.2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

The 30 minute requirement begins upon receipt of a single VALID fire detection system alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field. Actual field reports must be made within the 30 minute time limit or a classification must be made. If a fire is verified to be occurring by field report, classification shall be made based on EAL HU4.1, with the 15 minute requirement beginning with the verification of the fire by field report.

Table H-1 Fire Areas are those areas that contain equipment necessary for safe operation and shutdown of the plant (ref. 1, 2).

Reference(s):

1. 05-S-02-V-1 Response to Fires
2. 10-S-03-2 Response to Fires
3. UFSAR Appendix 9A Fire Hazard Analysis Report
4. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.3 Unusual Event

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

In addition to a FIRE addressed by EAL HU4.1 or HU4.2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.4 Unusual Event

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

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

If a FIRE within the plant PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gas

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

EAL:

HA5.1 Alert

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

AND

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

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

Table H-2 Safe Operation & Shutdown Rooms/Areas

Room/Area	Mode
Control Building 111' SWGR Rms (0C202, 0C215)	3
Auxiliary Building 93' RHR A Pump Room (1A103)	3
Auxiliary Building 93' RHR B Pump Room (1A105)	3
Auxiliary Building 93' Corridor (1A101)	3
Auxiliary Building 119' Corridor (1A201)	3
Auxiliary Building 119' RHR A Pump Room (1A203)	3
Auxiliary Building 119' RHR B Pump Room (1A205)	3
Auxiliary Building 119' RCIC Room (1A204)	3
Auxiliary Building 139' RHR A Room (1A303, 1A304)	3
Auxiliary Building 139' RHR B Room (1A306, 1A307)	3
Radwaste Building 118' Radwaste Control Room (0R241)	3

Mode Applicability:

3 – Hot Shutdown

Definition(s):

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

Basis:

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 3.
- 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 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 generate smoke and that automatically or manually activate a fire suppression system in an area.

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

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

EAL HA5.1 mode applicability has been limited to the mode limitations of Table H-2 (Mode 3 **only**).

Reference(s):

1. Attachment 2 Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases
2. NEI 99-01 HA5

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

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

EAL:

HA6.1 Alert

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel

Mode Applicability:

All

Definition(s):

None

Basis:

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

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Transfer of plant control begins when the last licensed operator leaves the Control Room.

Escalation of the emergency classification level would be via IC HS6.

Reference(s):

1. 05-1-02-II-1 Shutdown from the Remote Shutdown Panel
2. NEI 99-01 HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

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

EAL:

HS6.1 Site Area Emergency

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel

AND

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

- Reactivity (Modes 1 and 2 **only**)
- RPV water level
- RCS heat removal

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded; or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refueling

Definition(s):

None

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on Emergency 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 safe shutdown location(s).

Transfer of plant control and the time period to establish control begins when the last licensed operator leaves the Control Room.

Escalation of the emergency classification level would be via IC FG1 or CG1

Reference(s):

1. 05-1-02-II-1 Shutdown from the Remote Shutdown Panel
2. EP FAQ 2015-014
3. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

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

EAL:

HU7.1 Unusual Event

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.

Mode Applicability:

All

Definition(s):

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

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

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

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

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

Basis:

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 emergency classification level description for an UNUSUAL EVENT.

Reference(s):

1. NEI 99-01 HU7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

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

EAL:

HA7.1 Alert

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.

Mode Applicability:

All

Definition(s):

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 GGNS 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 GGNS. 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 - For the purposes of classification, the Security area between the OCA detection fence and the PROTECTED AREA boundary known as the Security Owner Controlled Area (SOCA) in the GGNS Emergency Plan.

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

Basis:

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 emergency classification level description for an ALERT.

Reference(s):

1. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist that in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY

EAL:

HS7.1 Site Area Emergency

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

Mode Applicability:

All

Definition(s):

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 GGNS 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 GGNS. 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 - For the purposes of classification, the Security area between the OCA detection fence and the PROTECTED AREA boundary known as the Security Owner Controlled Area (SOCA) in the GGNS Emergency Plan.

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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 emergency classification level description for a SITE AREA EMERGENCY.

Reference(s):

1. NEI 99-01 HS7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist that in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY

EAL:

HG7.1 General Emergency

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

Mode Applicability:

All

Definition(s):

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 GGNS 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 GGNS. 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.

OWNER CONTROLLED AREA - For the purposes of classification, the Security area between the OCA detection fence and the PROTECTED AREA boundary known as the Security Owner Controlled Area (SOCA) in the GGNS Emergency Plan.

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

Basis:

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 emergency classification level description for a GENERAL EMERGENCY.

Reference(s):

1. NEI 99-01 HG7

Category S – System Malfunction

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

Numerous system-related equipment failure events that warrant emergency classification have been identified in this category. They may pose actual or potential threats to plant safety.

The events of this category pertain to the following subcategories:

1. Loss of ESF AC 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 onsite and offsite power sources for 4.16 KV ESF buses.

2. Loss of Vital DC Power

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

3. Loss of Control Room Indications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant rise from these base-line levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

5. RCS Leakage

The reactor pressure vessel provides a volume for the coolant that covers the reactor core. The reactor pressure vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail. Excessive RCS leakage greater than Technical Specification limits indicates potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and containment integrity.

6. RPS Failure

This subcategory includes events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor scrams. In the plant licensing basis, postulated failures of the RPS to complete a reactor scram comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean any scram failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and containment integrity.

7. Loss of Communications

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

8. Hazardous Event Affecting Safety Systems

Various natural and technological events that result in degraded plant SAFETY SYSTEM performance or significant VISIBLE DAMAGE warrant emergency classification under this subcategory.

Category: S – System Malfunction

Subcategory: 1 – Loss of ESF AC Power

Initiating Condition: Loss of **all** offsite AC power capability to ESF buses for 15 minutes or longer

EAL:

SU1.1 Unusual Event

Loss of **all** offsite AC power capability, Table S-1, to DIV I and DIV II ESF 4.16 KV buses for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-1 AC Power Sources
Offsite • ESF Transformer 11 • ESF Transformer 12 • ESF Transformer 21 **Onsite** • DIV I DG (DG 11) • DIV II DG (DG 12)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

The HPCS bus (DIV III) is not credited because it only supplies power to the HPCS pump and associated loads, not any long term decay heat removal systems. In particular, suppression pool cooling mechanisms would be essential subsequent to a station blackout.

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC ESF buses. This condition represents a potential reduction in the level of safety of the plant.

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the ESF buses, whether or not the buses are powered from it.

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

Escalation of the emergency classification level would be via IC SA1.

Reference(s):

1. UFSAR Figure 8.1-001 Main One Line Diagram
2. UFSAR section 8.1 Electric Power Introduction
3. UFSAR section 8.3 Onsite Power
4. UFSAR section 8A Loss of all AC Power
5. 05-1-02-I-4 Loss of AC Power
6. NEI 99-01 SU1

Category: S – System Malfunction

Subcategory: 1 – Loss of ESF AC Power

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

EAL:

SA1.1 Alert

AC power capability, Table S-1, to DIV I and DIV II ESF 4.16 KV buses reduced to a single power source for ≥ 15 min. (Note 1)

AND

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-1 AC Power Sources
Offsite • ESF Transformer 11 • ESF Transformer 12 • ESF Transformer 21 **Onsite** • DIV I DG (DG 11) • DIV II DG (DG 12)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

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

Basis:

The HPCS bus (DIV III) is not credited because it only supplies power to the HPCS pump and associated loads, not any long term decay heat removal systems. In particular, suppression pool cooling mechanisms would be essential subsequent to a station blackout.

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

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

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- A loss of all offsite power and loss of all ESF emergency power sources (e.g., onsite diesel generators) with a single train of ESF buses being back-fed from the unit main generator.
- A loss of ESF emergency power sources (e.g., onsite diesel generators) with a single train of ESF emergency buses being fed from an offsite power source.

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

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

This EAL is the hot condition equivalent of the cold condition EAL CU2.1.

Reference(s):

1. UFSAR Figure 8.1-001 Main One Line Diagram
2. UFSAR section 8.1 Electric Power Introduction
3. UFSAR section 8.3 Onsite Power
4. UFSAR section 8A Loss of all AC Power
5. 05-1-02-I-4 Loss of AC Power
6. NEI 99-01 SA1

Category: S – System Malfunction

Subcategory: 1 – Loss of ESF AC Power

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

EAL:

SS1.1 Site Area Emergency

Loss of **all** offsite and **all** onsite AC power to DIV I and DIV II ESF 4.16 KV buses for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

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

Basis:

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink Mitigative strategies using other power sources (HPCS DIV III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout.. 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.

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

Escalation of the emergency classification level would be via IC AG1, FG1 or SG1.

This EAL is the hot condition equivalent of the cold condition EAL CA2.1.

Reference(s):

1. UFSAR Figure 8.1-001 Main One Line Diagram
2. UFSAR section 8.1 Electric Power Introduction
3. UFSAR section 8.3 Onsite Power
4. UFSAR section 8A Loss of all AC Power
5. 05-1-02-I-4 Loss of AC Power
6. NEI 99-01 SS1

Category: S –System Malfunction

Subcategory: 1 – Loss of ESF AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to ESF buses

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power to DIV I and DIV II ESF 4.16 KV buses

AND EITHER:

- Restoration of at least one ESF 4.16 KV bus in < 4 hours is **not** likely (Note 1)
- RPV water level **cannot** be restored and maintained > -191 in.

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

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

Basis:

Indication of continuing core cooling degradation is manifested by the inability to restore and maintain RPV water level above the Minimum Steam Cooling Reactor Water Level (-191 in.) (ref. 6). Core submergence is the most desirable means of core cooling, however when RPV level is below TAF, the uncovered portion of the core can be cooled by less reliable means (i.e., steam cooling or spray cooling).

This IC addresses a prolonged loss of all power sources to AC ESF emergency buses. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat

removal/pressure control, spent fuel heat removal and the ultimate heat sink. Mitigative strategies using other power sources (HPCS DIV III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout. A prolonged loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

The EAL should require declaration of a General Emergency prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC ESF emergency bus by the end of the analyzed station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is a greater likelihood of challenges to multiple fission product barriers. 4 hours is the site-specific SBO coping analysis time (ref. 4).

The estimate for restoring at least one ESF emergency bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, protective actions for the public.

The EAL will also require a General Emergency declaration if the loss of AC power results in parameters that indicate an inability to adequately remove decay heat from the core.

Reference(s):

1. UFSAR Figure 8.1-001 Main One Line Diagram
2. UFSAR section 8.1 Electric Power Introduction
3. UFSAR section 8.3 Onsite Power
4. UFSAR section 8A Loss of all AC Power
5. 05-1-02-I-4 Loss of AC Power
6. 02-S-01-40 EP Technical Bases
7. NEI 99-01 SG1

Category: S – System Malfunction

Subcategory: 1 – Loss of ESF AC Power

Initiating Condition: Loss of **all** ESF AC and vital DC power sources for 15 minutes or longer

EAL:

SG1.2 General Emergency

Loss of **all** offsite and **all** onsite AC power to DIV I and DIV II ESF 4.16 KV buses for ≥ 15 min. (Note 1)

AND

Indicated voltage is < 105 VDC on vital 125 VDC buses 11DA and 11DB for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

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

Basis:

Vital DC buses 11DA and 111DB feed the Division 1 and Division 2 loads respectively. The Division 1 and Division 2 batteries each have 61 cells with a design minimum of 1.72 volts/cell. These cell voltages yield minimum design bus voltages of 104.92 VDC (rounded to 105 VDC) (ref. 6, 7).

This IC addresses a concurrent and prolonged loss of both emergency ESF AC and Vital DC power. A loss of all emergency ESF AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling,

containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. Mitigative strategies using other power sources (HPCS DIV III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance
with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both emergency ESF AC and Vital DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

Reference(s):

1. UFSAR Figure 8.1-001 Main One Line Diagram
2. UFSAR section 8.1 Electric Power Introduction
3. UFSAR section 8.3 Onsite Power
4. UFSAR section 8A Loss of all AC Power
5. 05-1-02-I-4 Loss of AC Power
6. Calculation No: EC-Q1111-14001 Station Division I Battery 1A3 and Division II Battery 1B3 Discharge Capacity during Extended Loss of AC Power
7. UFSAR 8.3.2.1.1 Station DC Power
8. NEI 99-01 SG8

Category: S – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

Initiating Condition: Loss of **all** vital DC power for 15 minutes or longer

EAL:

SS2.1	**Site Area Emergency**
Indicated voltage is < 105 VDC on vital 125 VDC buses 11DA and 11DB for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

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

Basis:

Vital DC buses 11DA and 11DB feed the Division 1 and Division 2 loads respectively. The Division 1 and Division 2 batteries each have 61 cells with a design minimum of 1.72 volts/cell. These cell voltages yield minimum design bus voltages of 104.92 VDC (rounded to 105 VDC) (ref. 1, 2).

This IC addresses a loss of vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via IC AG1, FG1 or SG1.

This EAL is the hot condition equivalent of the cold condition EAL CU4.1.

Reference(s):

1. Calculation No: EC-Q1111-14001 Station Division I Battery 1A3 and Division II Battery 1B3 Discharge Capacity during Extended Loss of AC Power
2. UFSAR 8.3.2.1.1 Station DC Power
3. NEI 99-01 SS8

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer

EAL:

SU3.1 Unusual Event

An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-2 Safety System Parameters
• Reactor power • RPV water level • RPV pressure • Containment pressure • Suppression Pool water level • Suppression Pool temperature

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

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

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:

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital or recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, RPV water level and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RPV water level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via EAL SA3.1.

Reference(s):

1. UFSAR 7.5 Safety-Related Display Instrumentation
2. NEI 99-01 SU2

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress

EAL:

SA3.1 Alert

An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

Any significant transient is in progress, Table S-3

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-2 Safety System Parameters
• Reactor power • RPV water level • RPV pressure • Containment pressure • Suppression Pool water level • Suppression Pool temperature

Table S-3 Significant Transients
• Reactor scram • UNPLANNED drop in reactor thermal power > 25% • Electrical load rejection > 25% electrical load • ECCS injection • Thermal power oscillations > 10%

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

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

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:

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital or recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, RPV water level and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RPV water level cannot be determined from the indications and recorders on a main control board,

the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via IC FS1 or AS1.

Reference(s):

1. UFSAR 7.3 Engineered Safety Features Systems
2. NEI 99-01 SA2

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

SU4.1	**Unusual Event**
Offgas Pretreatment radiation monitor high-high alarm	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

The Offgas Pretreatment monitors radioactivity in the Offgas system downstream of the Offgas condenser. The monitor detects the radiation level that is attributable to the fission gases produced in the reactor and transported with steam through the turbine to the condenser. The Hi-Hi alarm, if alarming, indicates that the radioactivity present at the recombiner effluent discharge is at or above the Technical Specification 3.7.5 limit of 380 millicuries per second of Noble Gases. (ref. 1)

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

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

In the event that the Offgas Pretreatment Radiation Monitor High-High Alarm is out of service, the use of offgas flowrates and Offgas Pretreatment Radiation monitor readings is a viable contingency action to classify the EAL. See chart in 04-1-02-1H13-P601-19A-D7, Alarm Response Instruction for OG PRE-TREAT RAD HI-HI alarm.

Reference(s):

1. Alarm Response Instruction 04-1-02-1H13-P601-19A-D7
2. UFSAR 11.5 Process and Effluent Radiological Monitoring and Sampling Systems
3. Technical Specification 3.7.5 Main Condenser Offgas
4. 05-1-02-II-2 Offgas Activity High
5. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

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

EAL:

SU4.2 Unusual Event

Coolant activity > 0.2 µCi/gm dose equivalent I-131 for > 48 hours

OR

Coolant activity > 4.0 µCi/gm dose equivalent I-131 instantaneous

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

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

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

Reference(s):

1. Technical Specification B3.4.8, RCS Specific Activity bases
2. UFSAR Section 15.6.4 Steam System Piping Break Outside Containment
3. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1 Unusual Event

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

OR

RCS identified leakage > 25 gpm for ≥ 15 min. (Note 1)

OR

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

Basis:

Failure to isolate the leak within 15 minutes, or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

Identified leakage is leakage into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a collecting sump; or leakage into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage.

Unidentified leakage is all leakage into the drywell that is not identified leakage (ref. 2, 3).

Pressure boundary leakage is leakage through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall (ref. 2, 3).

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

The first and second EAL conditions are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications). The third condition addresses an RCS

mass loss caused by an UNISOLABLE leak through an interfacing system. These conditions thus apply to leakage into the containment, or a location outside of containment.

The leak rate values for each condition were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). The first condition uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. A stuck-open Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.

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 A or F.

Reference(s):

1. UFSAR Section 5.2.5, Detection of Leakage Through Reactor Coolant Pressure Boundary
2. Technical Specification Definitions Section 1.1
3. Technical Specification 3.4.5
2. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

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

EAL:

SU6.1 Unusual Event

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

AND

A subsequent automatic scram or manual scram action taken at the reactor control console (Mode Switch, Manual PBs, ARI/RPT) is successful in shutting down the reactor as indicated by reactor power ≤ 5% (APRM downscale) (Note 8)

Note 8: A manual scram action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation, 2 - Startup

Definition(s):

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

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram 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.

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function. A reactor scram is automatically initiated by the RPS when certain continuously monitored parameters exceed predetermined setpoints (ref. 1).

A successful scram has occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power to or below the APRM downscale setpoint of 5%.

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., Mode Switch, manual scram pushbuttons, or ARI/RPT initiation). Reactor shutdown achieved by use of alternate control rod insertion methods (i.e., EP-2A step Q-1) does not constitute a successful manual scram (ref. 2).

Following any automatic RPS scram signal, operating procedures (e.g., EP-2) prescribe insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved. Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Unusual Event (ref. 3).

Taking the Mode Switch to Shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

For the purposes of this EAL, a successful automatic initiation of ARI/RPT that reduces reactor power to ≤ 5% is not considered a successful automatic scram. If automatic initiation of ARI/RPT has occurred and caused reactor shutdown, the automatic RPS scram must have failed. ARI/RPT is a backup means of inserting control rods in the unlikely event that an automatic RPS scram signal exists but the reactor continues to generate significant power. However, a successful automatic or manual initiation of ARI/RPT is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

In the event that the operator identifies a reactor scram is IMMINENT and initiates a successful manual reactor scram before the automatic scram setpoint is reached, no declaration is required. The successful manual scram of the reactor before it reaches its automatic scram setpoint or reactor scram signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss.

If by procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

Following the failure of an automatic reactor scram, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram using a different switch). Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles". Taking the Reactor Mode Switch to Shutdown is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram 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 EAL SA6.1. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

Should a reactor scram signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal generated as a result of plant work causes a plant transient that results in a condition that should have included an automatic reactor scram and the RPS fails to automatically shutdown the reactor, then this IC and associated EALs are applicable, and should be evaluated.
- If the signal generated as a result of plant work does not cause a plant transient and the scram failure is determined through other means (e.g., assessment of test results), then this IC and associated EALs are not applicable and no classification is warranted.

Reference(s):

1. Technical Specification Table 3.3.1.1-1 Reactor Protection System Instrumentation
2. 05-S-01-EP-2A ATWS RPV Control
3. 05-S-01-EP-2 RPV Control
4. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

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

EAL:

SU6.2 Unusual Event

A manual scram did **not** shut down the reactor as indicated by reactor power > 5% after **any** manual scram action was initiated

AND

A subsequent automatic scram or manual scram action taken at the reactor control console (Mode Switch, Manual PBs, ARI/RPT) is successful in shutting down the reactor as indicated by reactor power ≤ 5% (APRM downscale) (Note 8)

Note 8: A manual scram action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation, 2 - Startup

Definition(s):

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

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

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

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

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

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram 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.

This EAL addresses a failure of a manually initiated scram in the absence of having exceeded an automatic RPS trip setpoint and a subsequent automatic or manual scram is successful in shutting down the reactor (reactor power ≤ 5%) (ref. 1).

A successful scram has occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power to or below the APRM downscale setpoint of 5%.

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., Mode Switch, manual scram pushbuttons, or ARI/RPT initiation). Reactor shutdown achieved by use of alternate control rod insertion methods (i.e., EP-2A step Q-1) does not constitute a successful manual scram (ref. 2).

Taking the Mode Switch to Shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

Successful automatic or manual initiation of ARI/RPT is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

If both subsequent automatic and subsequent manual reactor scram actions in the Control Room fail to reduce reactor power below the power associated with the SAFETY SYSTEM design (≤ 5%) following a failure of an initial manual scram, the event escalates to an Alert under EAL SA6.1.

Following the failure of an automatic reactor scram, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram) using a different switch. Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

Taking the Reactor Mode Switch to Shutdown is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles

are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

Should a reactor scram signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal generated as a result of plant work causes a plant transient that results in a condition that should have included an automatic reactor scram and the RPS fails to automatically shutdown the reactor, then this IC and associated EALs are applicable, and should be evaluated.
- If the signal generated as a result of plant work does not cause a plant transient and the scram failure is determined through other means (e.g., assessment of test results), then this IC and associated EALs are not applicable and no classification is warranted.

Reference(s):

1. Technical Specification Table 3.3.1.1-1 Reactor Protection System Instrumentation
2. 05-S-01-EP-SA ATWS RPV Control
3. 05-S-01-EP-2 RPV Control
4. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

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

EAL:

SA6.1 Alert

An automatic or manual scram fails to shut down the reactor as indicated by reactor power > 5%

AND

Manual scram actions taken at the reactor control console (Mode Switch, Manual PBs, ARI/RPT) are **not** successful in shutting down the reactor as indicated by reactor power > 5% (Note 8)

Note 8: A manual scram action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation, 2 - Startup

Definition(s):

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

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

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

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

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

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action

taken away from the reactor control consoles since this event entails a significant failure of the RPS.

This EAL addresses any automatic or manual reactor scram signal that fails to shut down the reactor followed by subsequent manual scram actions that fail 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%).

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., Mode Switch, manual scram pushbuttons, or ARI/RPT initiation). Reactor shutdown achieved by use of alternate control rod insertion methods (i.e., EP-2A step Q-1) does not constitute a successful manual scram (ref. 2).

For the purposes of this EAL, a successful automatic initiation of ARI/RPT that reduces reactor power to or below 5% is not considered a successful automatic scram. If automatic actuation of ARI/RPT has occurred and caused reactor shutdown, the automatic RPS scram must have failed. ARI/RPT is a backup means of inserting control rods in the unlikely event that an automatic RPS scram signal exists but the reactor continues to generate significant power. However, a successful automatic initiation of ARI/RPT is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control consoles (e.g., locally opening breakers). Actions taken at back panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

Taking the Reactor Mode Switch to Shutdown is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram) 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 RPV water level or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6 or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

Reference(s):

1. Technical Specification Table 3.3.1.1-1 Reactor Protection System Instrumentation
2. 05-S-01-EP-2A ATWS RPV Control
3. 05-S-01-EP-2 RPV Control
4. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to RPV water level or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual scram fails to shut down the reactor as indicated by reactor power > 5%

AND

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

AND EITHER:

RPV water level **cannot** be restored and maintained > -191 in.

OR

Heat Capacity Temperature Limit (HCTL) exceeded (EP Figure 1)

Mode Applicability:

1 - Power Operation, 2 - Startup

Definition(s):

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

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

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

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

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

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

This EAL addresses the following:

- Any automatic reactor scram signal followed by subsequent manual scram actions 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 (EAL SA6.1), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

Reactor shutdown achieved by use of control rod insertion methods in EP-2A step Q-1 are also credited as a successful shutdown provided reactor power can be reduced to or below the APRM downscale trip setpoint before indications of an extreme challenge to either core cooling or heat removal exist. (ref. 1)

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS barriers.

Indication that core cooling is extremely challenged is manifested by inability to restore and maintain RPV water level above the Minimum Steam Cooling RPV Water Level (MSCRWL) (ref. 1). The MSCRWL is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500°F. This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core spray cooling could be a precursor of a core melt sequence (ref 2).

The Heat Capacity Temperature Limit (HCTL, EP Figure 1) is the highest suppression pool water temperature from which Emergency RPV Depressurization will not raise suppression pool temperature above the maximum design suppression pool temperature.

The HCTL is a function of RPV pressure and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant. This threshold is met when the final step of section SPT in EP-3, Containment Control, is reached (ref. 3). This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature.

In some instances, the emergency classification resulting from this 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 EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

Escalation of the emergency classification level would be via IC AG1 or FG1.

Reference(s):

1. 05-S-01-EP-2A, ATWS RPV Control
2. 05-S-01-EP-5, RPV Flooding
3. 05-S-01-EP-3, Containment Control
4. NEI 99-01 SS5

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

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

EAL:

SU7.1 Unusual Event

Loss of **all** Table S-4 onsite communication methods

OR

Loss of **all** Table S-4 State and local agency communication methods

OR

Loss of **all** Table S-4 NRC communication methods

Table S-4 Communication Methods

System	Onsite	State/ Local	NRC
Station Radio System	X		
GGNS Plant Phone System	X		
Public Address System	X		
Emergency Notification System (ENS)			X
Commercial Telephone System		X	X
Satellite Phones		X	X
Operational Hotline		X	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

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 State and local agencies and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all State and local agencies of an emergency declaration. The State and local agencies referred to here are the Mississippi Emergency Management Agency, Claiborne County Civil Defense, Mississippi Highway Safety Patrol, Claiborne County Sheriff's Department, Louisiana Department of Environmental Quality, Tensas Parish Sheriff's Office, and the Louisiana Governor's Office of Homeland Security and Emergency Preparedness.

The third EAL condition addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

Reference(s):

1. GGNS Emergency Plan Section 7.5, Communications Systems
2. 04-S-01-R61-1 Plant Communications
3. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 8 – Hazardous Event Affecting Safety Systems

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

EAL:

SA8.1 Alert

The occurrence of **any** Table S-5 hazardous event

AND

Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode

AND EITHER:

- Event damage has caused indications of degraded performance to the second train of the SAFETY SYSTEM needed for the current operating mode
- Event damage has resulted in VISIBLE DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode

(Notes 9, 10)

Note 9: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then emergency classification is **not** warranted.

Note 10: If the hazardous event **only** resulted in VISIBLE DAMAGE, with **no** indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is **not** warranted.

Table S-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

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

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

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

> Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:
>
> (1) The integrity of the reactor coolant pressure boundary;
>
> (2) The capability to shut down the reactor and maintain it in a safe shutdown condition;
>
> (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

VISIBLE DAMAGE - Damage to a SAFETY SYSTEM train 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 SAFETY SYSTEM train.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues.

Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make a determination of VISIBLE DAMAGE 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. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via IC FS1 or AS1.

This EAL is the hot condition equivalent of the cold condition EAL CA6.1.

Reference(s):

1. EP FAQ 2016-002
2. NEI 99-01 SA9

Background

NEI 99-01 Revision 6 ICs AA3 and HA5 prescribe declaration of an Alert based on impeded access to rooms or areas (due to either area radiation levels or hazardous gas concentrations) where equipment necessary for normal plant operations, cooldown or shutdown is located. These areas are intended to be plant operating mode dependent. Specifically the Developers Notes for AA3 and HA5 states:

> *The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.*
>
> *The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).*

Further, as specified in IC HA5:

> *The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.*

GGNS Table A-3 and H-2 Bases

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:

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
IOI 03-1-01-2 Power Operations			
LOWER Power by reducing Recirculation flow until 62.2% core flow (70 mlbm/hr) is reached.	MCR	1	
INSERT Control Rods per Control Rod Movement Sequence.	MCR	1	
TECH SPEC TRIGGER (SR 3.3.2.1.2, SR 3.3.2.1.4) IF Reactor power has been reduced below the HPSP OR the LPSP, THEN PERFORM one of the following: Required Surveillances or enter LCO for TS 3.3.2.1	MCR	1	
CHECK OPEN the following valves on 1H13-P870-6C: a. N11-F029A, HP TURB EXTR To MSR A 1ST STG RHT b. N11-F029B, HP TURB EXTR To MSR B 1ST STG RHTIF N11-F029A OR N11-F029B are NOT open, THEN RETURN MSR 1ST Stage Reheaters to service per SOI 04-1-01-N11-1.	MCR	1	
CHECK OPEN the following valves on panel 1H13-P870-6C: a. N36-F010A, EXTR STM SPLY TO FW HTR 5A b. N36-F010B, EXTR STM SPLY TO FW HTR 5B c. N36-F011A, EXTR STM SPLY TO FW HTR 6A d. N36-F011B, EXTR STM SPLY TO FW HTR 6B TAKE handswitches for the following valves to OPEN position on panel 1H13-P870-6C: a. N36-F013A, FW HTR 5A EXTR STM BTV b. N36-F013B, FW HTR 5B EXTR STM BTV c. N36-F012A, FW HTR 6A EXTR STM BTV d. N36-F012B, FW HTR 6B EXTR STM BTV	MCR	1	
NOTIFY the following of the power reduction: • Load Dispatcher (Woodlands) • *Duty Manager (IF unexpected power reduction) • (SMEPA)(1-601-261-2318 OR 1-601-261-2313) • * (SMEPA) Site Representative • Radwaste • Radiation Protection • Chemistry • *NRC Resident Inspector	MCR	1	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
*These notifications Must be made by Shift Manager			
IF LP Turbine inlet temperature is >491°F, and N11-F028A and N11-F028B are open, THEN SIMULTANEOUSLY THROTTLE the following valves on 1H22-P177 to CONTROL LP Inlet Temperatures within a band of 470° F to 490° F while monitoring LP Turbine Inlet differential temperatures within 30° F (comparing A side to B side). • N11-F028A • N11-F028B •I IF LP Turbine inlet temperature is >491°F, and N11-F028A and N11-F028B are closed, THEN SLOWLY, SIMULTANEOUSLY LOWER MSR-A/B HTG STM FEED CONT manual setpoint to CONTROL LP Inlet Temperatures within a band of 470° F to 490° F while monitoring LP Turbine Inlet differential temperatures within 30° F (comparing A side to B side).	TURB BLDG ELEV 133 AREA 4 ROOM 1T325	1	Not Required
LOWER Reactor power by INSERTING control rods to specified Control Rod in-sequence position per 17-S-02-400.	MCR	1	
At approximately 48% Reactor power, PERFORM the following on panel 1H13-P601. VERIFY the following valves Open: • B21-F033 INBD MSL DR SOL TO MN CNDSR • B21-F069 OTBD MSL DR SOL TO MN CNDSR • OPEN B21-F016	MCR	1	
At approximately 50% Reactor Power, PERFORM the following: **SHUTDOWN 1 Reactor Feed Pump per SOI 04-1-01-N21-1.**			
VERIFY RFPT B is operating normally on master controller.	MCR	1	
RAISE FW MASTER LVL CONT setpoint to approximately 39"	MCR	1	
TRANSFER the RFPT A SP CONT to MAN.	MCR	1	
SLOWLY LOWER speed of RFPT A USING RFPT A SP CONT by DEPRESSING the OUT □ pushbutton. OBSERVE speed of RFPT B raises to maintain RPV water level, OR control it manually	MCR	1	
FURTHER REDUCE speed of RFPT A using RFPT A SP CONT in MAN until it reaches low speed stop.	MCR	1	
TRANSFER speed control of RFPT A to SPEED AUTO by DEPRESSING the OBSERVE the FW AUTO pushbutton extinguishes AND the SPEED AUTO, RAISE, AND LOWER pushbuttons backlight.	MCR	1	
FURTHER REDUCE RFPT A speed using RFPT A LOWER pushbutton.	MCR	1	
WHEN RFPT A speed reaches 1100 rpm, THEN TRIP RFPT A by DEPRESSING the RFPT A MAN TRIP pushbutton	MCR	1	
CHECK F014A, RFP A DISCH VLV starts to close.	MCR	1	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
REOPEN F014A, RFP A DISCH VLV			
WHEN RFPT A coasts down to zero speed, THEN RESET turning gear by pressing A TURN GEAR OPER RESET pushbutton. OBSERVE turning gear engages automatically, unless RFPT A is rolling on min flow.	MCR	1	
IF turning gear fails to engage, THEN MANUALLY ENGAGE the turning gear locally by PRESSING DOWN the manual engaging lever.	TURB BLDG ELEV 133 AREA 3 ROOM 1T307, 1T309	1	Not Required
CHECK OPEN/OPEN the following Drain valves on 1H22-P175: 1N11-F019A, RFPT A HP IN DR VLV 1N11-F023A, RFPT A HP IN DR VLV 1N11-F018A, RFPT A IP IN DR VLV 1N11-F021A, RFPT A IP IN DR VLV 1N11-F042A, RFPT A IP IN DR VLV 1N33-F021A, RFPT A ABOVE SEAT DR 1N33-F022A, RFPT A ABOVE SEAT DR 1N33-F023A, RFPT A BELOW SEAT DR 1N33-F024A, RFPT A BELOW SEAT DR	N/A	N/A	These steps are not required to be performed to Shut down and Cooldown the plant.
RETURN FW MASTER LVL CONT setpoint to approximately 36"	MCR	1	
IF desired, RESET RFPT A trip using the RFPT A TRIP RESET pushbutton	MCR	1	
SHUTDOWN 1 Circulating Wtr Pump per SOI 04-1-01-N71-1			
CHECK that CTCS balls are collected AND system shut down.			
DEPRESS the BALL CATCH FLAP CATCH pushbutton on P001A (B) MIMIC AND OBSERVE the flap rotates to the CATCH position.	Turb Bldg 113' Area 4 (1T203)	1	Not Required
OBSERVE ball collection starts by a rising number of balls in ball collector tank.	Turb Bldg 113' Area 4 (1T203)	1	Not Required
After 10 minutes STOP Ball Recirculation pump by DEPRESSING RECIRC PUMP OFF pushbutton on P001A(B) MIMIC	Turb Bldg 113' Area 4 (1T203)	1	Not Required
CLOSE Pump Discharge Valve F323A(B).	Turb Bldg 113' Area 4 (1T203)	1	Not Required
PLACE Screens #1 AND #2 in BACKWASH position by DEPRESSING	Turb Bldg	1	Not

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
SCREEN BACKWASH pushbutton on P001A (B) MIMIC AND OBSERVE screens rotate to BACKWASH position.	113' Area 4 (1T203)		Required
PRESS the CIRC WTR PMP A(B) STOP pushbutton on 1H13-P680.	MCR	1	
CHECK that F002A(B) Circulating Water Pump Discharge valve closes on 1H13-P680	MCR	1	
ENSURE that A(B) Circulating Water pump has shut down USING pump indication light on 1H13-P680 WHEN its discharge valve is CLOSED.	MCR	1	
OPEN OR CHECK OPEN F001 USING CIRC WTR LOOP A/B XTIE handswitch on 1H13-P870.	MCR	1	
CLOSE OR CHECK CLOSED F040A (B) Acid Feed Valve.	N/A	N/A	Not required to be performed to Shut down and Cooldown the plant.
CLOSE OR CHECK CLOSED LV-F513 A(B), Blowdown valve	MCR	1	
OPEN F039A(B), CIRC WTR PUMP A(B) COLUMN VENT	N/A	N/A	Not required to be performed to Shut down and Cooldown the plant.
ENSURE Condenser vacuum is maintained > 23.8" Hg	MCR	1	
SHUTDOWN one Heater Drain Pump per SOI 04-1-01-N23-1			
JOG CLOSED N23-F051A(B), HTR DR PMP A(B) DISCH VLV on 1H13-P680 for desired pump.	MCR	1	
STOP HTR DR PMP A(B) on 1H13-P680.	MCR	1	
WHEN Reactor power has been reduced < 40%, SHUTDOWN 2nd Heater Drain Pmp per SOI 04-1-01-N23-1			
Before securing second Heater Drain Pump, PLACE N23-LK-R053, HTR DR TK DR, in Manual AND Slowly REDUCE output to 0%.	MCR	1	
ENSURE Heater Drain Tank level is maintained by Dump Valves N23-LV-F518A-E	MCR	1	
JOG CLOSED N23-F051B(A) HTR DR PMP B(A) DISCH VLV on 1H13-P680 for second pump.	MCR	1	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
STOP Heater Drain Pump HTR DR PMP B(A) on 1H13-P680.	MCR	1	
WHEN BOTH Heater Drain Pumps are shutdown, THEN CLOSE N23-F054, HTR DR PMP COMMON DISCH VLV on 1H22 P175	TURB BLDG ELEV 133 AREA 6 ROOM 1T327	1	Not Required
IOI 03-1-01-2 Continued			
SHIFT the Reactor Recirculation Pump(s) to slow speed as follows: INSERT Control Rods until Load Line is between 50 AND 65% VERIFY Control Rods are in sequence of the Control Rod Pattern Controller.	MCR	1	
BEFORE entering Controlled Entry Region of Figure 3, PERFORM the following WHEN TS 3.3.1.1, Action J.1 is in effect: VERIFY Fraction of Core Boiling Boundary (FCBB) is ≤ 1.0 per 06 RE-1J11-V-0002. IMPLEMENT TS 3.3.1.1, Action J.2, within 12 hours of entry AND J3 within 90 days.	MCR	1	
IF any APRM gain is out of tolerance, THEN ADJUST gain per 06-RE-1C51-W-0001 prior to downshift of Recirculation Pumps.	MCR	1	
CLOSE Both Recirculation A AND B Flow Control Valves (FCV's) to Min Ed position using RECIRC A(B) FLO CONT on 1H13-P680	MCR	1	
TRANSFER Both Reactor Recirculation Pumps to slow speed per SOI 04-1-01-B33-1	MCR	1	
CONTINUE Reactor Power reduction to 25 - 30% by insertion of Control Rods	MCR	1	
SHUTDOWN Hydrogen Water Chemistry Injection per **SOI 04-1-01-P73-1.**			
At H13-P845, momentarily DEPRESS HWC SHUTDOWN pushbutton AND OBSERVE the following: HWC SHUTDOWN pushbutton 1P73-M602 Will be flashing as H2 AND O2 flows ramp down to 0. O2 isolation valves Will Close WHEN O2 levels remain at normal levels with no O2 injection for at least 5 minutes. HWC SHUTDOWN pushbutton Will be in solid WHEN all control valves AND isolation valves are fully Closed. HWC RUNNING pushbutton extinguishes.	MCR	1	
CLOSE P73-F107, H2 Inj Sply Line Man Line Shutoff valve. After O2 valves F515 AND F512 (as indicated by white dots on red cap being perpendicular to pipe) have Closed, CLOSE OR CHECK CLOSED Both F207 AND F208, O2 Rack Sply Isol to OG Preheater A(B).	N/A	N/A	Not required to be performed to Shut

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
CLOSE 1P73-F209, O2 injection to Condensate pumps			down and Cooldown the plant.
IF Drywell entry is scheduled, WHEN Reactor Power has been reduced to less than 30%, THEN PERFORM the following: PERFORM the following for 1D21-K607, DRWL PERS HATCH ARM: DIRECT I&C to CONNECT Canon plug to the plug labeled "ALARM" AND "J3" at the back of 1D21K607. PLACE Function Selector switch on front of 1D21K607 (DRWL PERS HATCH ARM) to OPERATE position. PERFORM EPI 04-1-03-D21-1 for 1D21K607.	N/A	N/A	Not required to be performed to Shut down and Cooldown the plant.
IOI 03-1-01-2 Continued **REMOVE Both Second Stage MSR Reheaters from service per SOI 04-1-01-N11-1.**			
OBSERVE PDS Computer Points N11N044A,B,C AND N11N045A,B, C to monitor LP Turbine Inlet Temperature □T during removal of Second Stage Reheaters from service.	MCR	1	
ENSURE Both MSR HTG STM FEED CONT are in MANUAL on 1H13-P680.	MCR	1	
CLOSE the following MSR 2ND STG HTG STM valves on 1H13-P680: • N11-F304C • N11-F304D	MCR	1	
SIMULTANEOUSLY CLOSE the following MSR 2ND STG HTG STM valves on 1H13-P680: • N11-F304A • N11-F304B	MCR	1	
LOWER the manual outputs on Both MSR HTG STM FEED CONT to minimum on 1H13-P680 to close the temperature control valves.	MCR	1	
CLOSE the following MSR SUPPLY VLVS valves on 1H22-P177. • N11- F028A • N11- F028B	TURB BLDG ELEV 133 AREA 4 ROOM 1T325	1	Not Required
VERIFY the following valve lineup on local panels: • N35-F015A Closed, HS-M003A • N35-F015B Closed, HS-M003B • N35-F018A Closed, HS-M007A • N35-F018B Closed, HS-M007B	N/A	N/A	Not required to be performed to Shut down and Cooldown the plant.

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
IF Feedwater Heater 6A/B are being supplied from extraction steam (i.e., IF 1N36-F010A/B AND 1N36-F011A/B on 1H13-P870 are open), THEN CLOSE the following valves on 1H22-P177: • N35-F008A, 2ND STG RHTR DR TK A TO HTR 6A • N35-F008B, 2ND STG RHTR DR TK B TO HTR 6B	TURB BLDG ELEV 133 AREA 4 ROOM 1T325	1	Not Required
REMOVE Both First Stage MSR Reheaters from service per SOI 04-1-01-N11-1.			
OPEN the following valves on 1H13-P870: • N11-F005A, MSR 1ST STG RHT RO BYP DR VLVS • N11-F005B, MSR 1ST STG RHT RO BYP DR VLVS	MCR	1	
SIMULTANEOUSLY CLOSE the following valves on 1H13-P870: • N11-F029A, HP TURB EXTR TO MSR A • N11-F029B, HP TURB EXTR TO MSR B	MCR	1	
CLOSE the following valves by taking its respective handswitch to TEST: • N11-F003A, MSR A 1ST STG RHT EXTR STM BTV (1H13-P870) • N11-F003B, MSR B 1ST STG RHT EXTR STM BTV (1H13-P870)	MCR	1	
REMOVE Condensate Precoat filters from service per SOI 04-1-01-N22-1, IF in service.			Not required to be performed to Shut down and Cooldown the plant.
OPEN the following BSCV UPSTRM DR VLV's: a. N33-F300A b. N33-F300B c. N33-F300C	MCR	1	
At approximately 23 – 26 % Reactor Power, RAISE the SPEED DEMAND setpoint to approximately 35%, as monitored on PDS Computer point N32K246, by DEPRESSING the SP DEMAND RAISE AND REL pushbuttons.	MCR	1	
SIMULTANEOUSLY DEPRESS LOAD REF OFF AND REL pushbuttons on 1H13-P680-9C to turn off load demand Control AND VERIFY OFF light is illuminated.	MCR	1	
LOWER load by DEPRESSING SPEED DEMAND LOWER AND REL pushbutton. (Expected value 150 – 175 MWe)	MCR	1	
OBTAIN Shift Manager permission for Manual Scram	MCR	1/2	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
NOTIFY the following that Main Generator is being disconnected from the grid: • Entergy Load Dispatcher (Woodlands) • (SMEPA) 1-601-261-2318 OR 1-601-261-2313) • Entergy Mississippi Dispatcher • Duty Manager	MCR	1/2	
VERIFY Switchyard lineup is acceptable for trip of J5228 AND J5232	MCR	1/2	
INSERT IRMs	MCR	1/2	
NOTIFY the following personnel/departments that a manual scram is being initiated: • Radwaste • Chemistry • Radiation Protection ANNOUNCE over plant pager that manual Scram is being initiated.	MCR	1/2	
TAKE initial temperature data per Attachment III, Data Sheet I of IOI 03-1-01-3 prior to scram	MCR	1/2	
Manually SCRAM the Reactor using the MANUAL SCRAM pushbuttons. a. VERIFY all Control Rods are fully inserted. b. VERIFY Reactor Power is decreasing. c. IF Pressure Control System is maintaining reactor pressure greater than 850 psig, THEN PLACE Reactor Mode switch to SHUTDOWN. d. VERIFY Reactor Recirculation pumps are running in slow speed.	MCR	1/2/3	
ENSURE Main Turbine and Generator trip. (Reverse power 15 seconds time delay, 5 seconds time delay IF turbine has already tripped.). a. VERIFY the Generator Output Breakers open. b. VERIFY the Turbine Stop and Control Valves close.	MCR	3	
WHEN reactor water level Can be restored AND maintained above 11.4 inches, THEN PERFORM the following to prevent Reactor water level from reaching Level 9 RFPT trip setpoint (58 in.): IF Reactor pressure is dropping rapidly, THEN SELECT SPEED AUTO OR MANUAL on the running Reactor Feed Pump AND LOWER Reactor Feed Pump discharge pressure to MAINTAIN Reactor level below 58 inches. TRANSFER Feedwater Control to Start-Up Level Control per SOI 04-1-01-N21-1. (Attachment VII of SOI 04-1-01-N21-1 May be used.)	MCR	3	
ENSURE Scram Discharge Volume Vent AND Drain valves closed	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
IOI 03-1-01-4 SCRAM Recovery			
INSERT all SRM's AND VERIFY response on SRM recorders.	MCR	3	
SWITCH IRM/APRM LVL recorders to IRM AND VERIFY neutron monitoring established on IRM's	MCR	3	
IF scram signal Can be cleared AND Reactor level AND pressure are stable, THEN RESET scram AND RETURN CRD System to normal as follows: BYPASS Scram Instrument Volume High Level signal by PLACING CRD DISCH VOL HI TRIP BYP switches RPS Div 1, 2, 3, 4 to BYPASS. RESET scram by PLACING SCRAM RESET handswitches RPS Div 1, 2, 3, 4 to RESET. VERIFY all CRDs settle into Position '00'. IF any Control Rod is NOT at the '00' position, THEN PERFORM one notch insert to attempt to force the rod to settle into the '00' position. WHEN "CRD DISCH VOL WTR LVL HI TRIP" annunciator is clear, THEN RETURN CRD DISCH VOL HI TRIP BYP switches to NORMAL. VERIFY that the HCU scram accumulators have been recharged by OBSERVING the ACCUM FAULT indicating lights on 1H13-P680 are out.	MCR	3	
THROTTLE G33-F102 to raise bottom head drain flow AND limit Bottom Head Drain Line Heatup/Cooldown to < 100°F/HR. Bottom head drain flow greater than 250 gpm May be required.	MCR	3	
IF Reactor water level is high, THEN REJECT water to Main Condenser per SOI 04-1-01-G33-1 to MAINTAIN level band.			
PLACE NSSSS OTBD MOV TEST handswitch on 1H13-P601-19B to the TEST position. VERIFY that "RX DIV 1 ISOL SYS OOSVC" annunciator (1H13-P601-19A-H3) Alarms.	MCR	3	
PLACE NSSSS INBD MOV TEST handswitch on 1H13-P601-18B to the TEST position. VERIFY that "RX DIV 2 ISOL SYS OOSVC" annunciator (1H13- P601-19A-G3) Alarms.	MCR	3	
ADJUST F033, RWCU SYS BLWDN F/D CONT VLV is ~ 10% Open.	MCR	3	
OPEN OR CHECK OPEN the following valves: F028 RWCU BLWDN CTMT INBD ISOL 1H13-P680 F034, RWCU BLWDN CTMT OTBD ISOL1H13-P680	MCR	3	
IF rejecting to main condenser, OPEN OR CHECK OPEN in the following order: F046 RWCU BLWDN TO MN CNDSR 1H13-P680	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
F041 RWCU BLWDN TO MN CNDSR BYP 1H13-P680 F235 RWCU BLWDN TO MN CNDSR 1H13-P870-3C F234 RWCU BLWDN TO MN CNDSR 1H13-P870-9C			
IF desired, while rejecting during depressurized OR low pressure conditions, F031, RWCU BLWDN ORF BYP VLV May be Open to allow maximum flow	MCR	3	
Begin rejecting by SLOWLY OPENING F033, RWCU SYS BLWDN FLO CONT valve, AND IF necessary THROTTLING CLOSED F042	MCR	3	
OBSERVE FI-R602, RWCU BLWDN FLO indicator on 1H13-P680	MCR	3	
MONITOR reactor water level, blowdown flow AND area/room temperature indication while reject is in progress.	MCR	3	
ENSURE Bypass valves are maintaining Reactor pressure	MCR	3	
IF proceeding to Cold Shutdown, THEN PERFORM Cooldown per Attachment II of IOI 03-1-01-3 concurrent with remaining steps of this attachment.	MCR	3	
DEPRESS the MHC START DVC "LOWER" pushbutton on 1H13-P680-9C to reduce the MHC START DVC to Zero.	MCR	3	
CONFIRM the following Bleeder Trip valves are Closed: a. N36-F013A, FW HTR 5A EXTR STM BTV b. N36-F013B, FW HTR 5B EXTR STM BTV c. N36-F012A, FW HTR 6A EXTR STM BTV d. N36-F012B, FW HTR 6B EXTR STM BTV e. N11-F003A, MSR A 1ST STG RHT EXTR STM BTV f. N11-F003B, MSR B 1ST STG RHT EXTR STM BTV	MCR	3	
ENSURE Seal Steam Pressure AND Reactor Feed Pump operation maintained by main steam	MCR	3	
CLOSE the following valves as soon as possible following Turbine trip at Gas Rack 1N44D001-N to isolate Hydrogen Pressure Regulators N44-PCV-F505 AND F506: a. N44-FA20 b. N44-FA21	N/A	N/A	Not required to be performed to Shut down and Cooldown the plant.
OBSERVE the following actions occur: Field amps AND generator output voltage indicate 0. Generator field breaker Will trip on a generator/transformer lockout condition (including reverse power) AND the TVR feeder switch Will open IF a lockout was NOT initiated WHEN the Turbine speed drops to ~1620 rpm. TURB AUX OIL PMPS A, B OR C starts at about 1335 rpm. AUX PW CIRC PUMP starts at about 815 rpm. (Locally)	MCR	3	Aux Primary Water Circ Pump can be verified running by computer

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
TURB SHAFT LIFT OIL PMP starts at about 510 rpm. TURB GEAR OIL VLVs N34-FE01/FE02 open at about 210 rpm.			point in the MCR.
THROTTLE P43-F053 to maintain Main Turbine Lube Oil temp between 90-119□F.	N/A	N/A	Not required to be performed to Shut down and Cooldown the plant.
WHEN fast speed trend recording is no longer necessary AND vessel level is greater than 11.4" AND vessel pressure is less than 1064.7 psig., THEN PERFORM the following: DEPRESS the POST ACC MON HI SP RESET pushbutton for POST ACC MON B21-R623A on 1H13-P601-20B. DEPRESS the POST ACC MON HI SP RESET pushbutton for POST ACC MON B21-R623B on 1H13-P601-17B.	MCR	3	
OPEN the Generator motor operated air break GEN DISC J5230. PLACE Red Tag on the Control Room handswitch for J5230 in open position. (This step May be performed after step 9.30.3)	MCR	3	
AFTER GEN DISC J5230 is opened, THEN PERFORM the following: IF tripped, THEN RESET the following Generator reverse power relays by PRESSING the relay reset rod upwards: a. 432/G12 (1N41-M752) b. 432/UT11 (1N41-M756) AFTER Generator reverse power relays are reset, THEN RESET the following Generator Lockout relays, IF tripped: a. 486-1/G12 (1N41-M769) b. 486-2/G12 (1N41-M770) c. 786-1/UT11 (1N41-M759) d. 786-2/UT11 (1N41-M760)	MCR	3	
AFTER all Generator Lockout relays are reset AND "GEN UNIT TRIP" annunciator clears on 1H13-P680-9A-A8, THEN OBTAIN Entergy Mississippi dispatcher's permission AND PERFORM the following to close breakers J5228 AND J5232 from 1H13-P680 panel: PLACE SYNC CONT BRKR J5228 switch to ON position. CLOSE 500 KV BRKR J5228. PLACE SYNC CONT BRKR J5228 switch to OFF position PLACE SYNC CONT BRKR J5232 switch to ON position. CLOSE 500 KV BRKR J5232. PLACE SYNC CONT BRKR J5232 switch to OFF position.	MCR	3	
IF all Generator Lockout relays Will NOT reset, THEN PERFORM the	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
following: CONTACT Electrical Maintenance to investigate reason any other Generator relays other than reverse power May have tripped. REQUEST Entergy Mississippi dispatcher to open disconnects to de-energize breaker(s) J5228 AND J5232.			
DEPRESS EHC SP DEMAND LOWER AND REL pushbutton on 1H13-P680-9C to reduce SP DEMAND indicator to 0 percent. WAIT for SP LTD meter to decrease to 0 percent.	MCR	3	
At each Main Transformer Control Cabinet (Phase A, Phase B, AND Phase C), VERIFY lead cooler group fans are OFF	Outside at MN XFMRs	3	Not Required
SECURE the following steam loads to limit plant cooldown: • **SJAE per SOI 04-1-01-N62-1**			
CLOSE Recombiner Drain Valves N64-F264 AND F265 (N64-F268 AND F269)	93' OG Preheater A/B Rooms 1T109 1T110	3	Not Required
CLOSE N64-F007A(B) Preheater Inlet Drain using handswitch on N64-P001.	113' Turb Area 1 1T202	3	Not Required
OPEN RECOMBINER AIR PURGE A(B) Manual Valve 1N64-F004A(B) Train A(B) Purge Air Sply Sol Byp for the corresponding recombiner train to ESTABLISH a purge flow of approximately 60 scfm through the recombiner train.	93' OG Preheater A/B Rooms 1T109 1T110	3	Not Required
CLOSE N62-F003A(B) CNDSR AIR TO 1 STG SJAE A(B) locally at 1H22 P176 OBSERVE that F003A(B) CNDSR AIR TO 1 STG SJAE A(B) indicates Closed before continuing to the next step.	133' Turb Area 1/4 1T305, 1T324	3	Not Required
DEPRESS N62-F003A(B) SJAE A(B) 1ST STG SUCT VLV CLOSE pushbutton on 1H13-P680 [10C].	MCR	3	
CHECK the indication on 1H13-P680 and the following valves Close: SJAE A(B) 1ST STG STM INL VLV, N62-F024A(B) SJAE ICNDSR DR VLV, N62-F011A(B) SJAE A(B) 2ND STG SUCT VLV, N62-F006A(B) SJAE A(B) MN STM SPLY VLV, N62-F001A(B) SJAE A(B) EXH VLV, N62-F012A(B) SJAE A(B) SEP DR VLV, N62-F002A(B)	MCR	3	
REDUCE setpoint of N62-PIC-R010A(B) to zero 0 psi	113' Turb Area 1 1T202	3	Not Required

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
ENSURE OPEN following handswitches on 1H22-P176: N62-F004A the COND AIR TO 1 STG SJAE A N62-F004B, the COND AIR TO 1 STG SJAE B	133' Turb Area 1/4 1T305, 1T324	3	Not Required
ENSURE OPEN N62-F034 A, B, C, DISCH PIPE DRN VLV for draining discharge piping.	113' Turb A MVP Area 1T218	3	Not Required
WHEN discharge piping has drained, THEN CLOSE N62-F034 A, B, C, DISCH PIPE DRN VLV.	113' Turb A MVP Area 1T218	3	Not Required
OPEN N62-F014 MECH VAC PUMPS COM SUCT VLV, at 1H22-P176.	133' Turb Area 1/4 1T305, 1T324	3	Not Required
ENSURE proper mechanical vacuum pump oil level (>50%), THEN Prelube with manual oiler as follows: ENGAGE manual oiler pump handle ROTATE for a minimum of 60 seconds. DEPRESS each plunger 5 times CHECK oil flow visible from each oil return line.	113' Turb A MVP Area 1T218	3	Not Required
CLOSE P44-F348 A(B,C) MECH VAC PMP COOLER DRAIN. OPEN P44-F109 A(B,C) MECH VAC PMP PSW INL ISOL. OPEN P44-F344 A(B,C) MECH VAC PMP PSW DISCH ISOL. BLOW DOWN strainer as follows: (1) OPEN P44-F316 A(B,C), MECH VAC PMPA(B)(C) STR DR. (2) WHEN blowdown has been completed, THEN CLOSE P44-F316 A(B,C) MECH VAC PMPA(B)(C) STR DR.	113' Turb A MVP Area 1T218	3	Not Required
START MECH VAC PMP A(B)(C) with START pushbutton on 1H13 P680.	MCR	3	
CHECK proper vacuum pump operation for each running pump by OBSERVING the following: Cooling Water Inlet Valve 1P44-SV-F514A, B, OR C has opened by MOMENTARILY OPENING drain valve 1P44-F348A, B, C MECH VAC PMP A(B)(C) CLR DR. OBSERVING pressurized water flow, THEN CLOSE drain valve1P44-F348A, B, C MECH VAC PMP A(B)(C) CLR DR. IF 1P44-SV-F514A, B, OR C did NOT open, THEN OPEN respective MECH VAC PMP A(B)(C) PSW SPLY BYP valve 1P44-F347A,B,C to provide cooling as needed for operation of Mechanical Vacuum Pump. Suction Drain Valve SV-F507A, B OR C has Closed by OBSERVING	113' Turb A MVP Area 1T218	3	Not Required

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
no air suction flow. Mechanical Vacuum Pump Inlet Valve F007A, B, OR C has Opened. Proper oiler operation by OBSERVING oil flow from each oil return line.			
Secure Seal Steam Generator per SOI 04-1-01-N33-1			
PLACE Controller PK-R617 in MANUAL on 1H13-P878, AND CLOSE F506 as necessary to control reactor cooldown. The turbine Can be sealed with seal steam header pressure as low as 15 psig, PI-R622.	MCR	3	
Secure Reactor Feed Pump per SOI 04-1-01-N21-1			All areas previously addressed in securing 1st RFPT
IOI 03-1-01-4 Continued			
Offgas Preheater by placing controllers 1N64-R009A and 1N64-R009B in manual and reducing output to 0 percent	Turbine Building 93' Area 1 (1T113)	3	Not Required
Main Steam Isolation valves AND/OR Main Steam Line	MCR	3	
SHUTDOWN a Condensate Booster Pump AND CLOSE respective discharge valve per SOI 04-1-01-N19-1, leaving one Condensate Booster Pump in service	MCR	3	
SHUTDOWN a Condensate Pump AND CLOSE respective discharge valve per SOI 04-1-01-N19-1, leaving one Condensate Pump in service	MCR	3	
CLOSE B21-F069 OPEN the following MSIV drain valves: a. B21-F067A b. B21-F067B c. B21-F067C d. B21-F067D	MCR	3	
OPEN the following valves: a. B21-F033 b. B21-F068	MCR	3	
ISOLATE extraction steam to the HP Feedwater heaters as follows: CLOSE the following valves: a. N36-F010A EXTR STM SPLY TO FW HTR 5A b. N36-F010B EXTR STM SPLY TO FW HTR 5B c. N36-F011A EXTR STM SPLY TO FW HTR 6A d. N36-F011B EXTR STM SPLY TO FW HTR 6B	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
OBSERVE the following drain valves open: N36-F008A FW HTR 6A EXTR STM RO BYP DR VLV N36-F008B FW HTR 6B EXTR STM RO BYP DR VLV	MCR	3	
OPEN the following drain valves: OPEN HP Stop AND Control Valve Drain Valves by DEPRESSING each of the following MSCV UPSTRM DR VLV "JOG OPEN" pushbuttons: a. N33-F078A b. N33-F078B c. N33-F078C d. N33-F078D OPEN Left Side Crossover piping drains by DEPRESSING each of the following XOVER PIPE LS DR VLV "JOG OPEN" pushbuttons: a. N11-F043A (FR IST) b. N11-F036A (FR 2ST) c. N11-F044A (RE IST) d. N11-F038A (RE 2ST) OPEN Right Side Crossover piping drains by DEPRESSING each of the following XOVER PIPE RS DR VLV "JOG OPEN" pushbuttons: a. N11-F044B (FR IST) b. N11-F038B (FR 2ST) c. N11-F043B (RE IST) d. N11-F036B (RE 2ST) OPEN N11-F015, MSCV A/B DNSTRM DR VLV. OPEN the following MSR 2ND STG STM DR VLVS: a. N11-F301 b. N11-F302	MCR	3	
OPEN the following drain valves unless required closed to minimize cooldown: OPEN Main Steam Line Drain Valves N11-F056, F055, F009, F011, F049, AND F050 by DEPRESSING MSL DR LINE ISOL VLVS "OPEN" pushbutton. OPEN MSL Bypass Drain valves (N11-F002A, F002B, F002C, F002D, F010, F007, F052A, F052B, F057) using MSL DR VLVS DR LINE BYP VLV "OPEN" pushbutton.	MCR	3	
DEPRESS Both NSSSS INBD ISOL RESET pushbutton (1H13-P601-18B) AND NSSSS OTBD ISOL RESET pushbutton (1H13-P601-19B) to reset logic AND re-energize RHR Logic lights on 1H13-P622 AND 1H13-P623 panels.	MCR	3	
TRANSFER to startup level control IF NOT already in service	MCR	3	
TRANSFER the RFPT A(B) SP CONT to MAN.	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
IF MSIV's are open with Main Condenser available, THEN INITIATE AND MAINTAIN cooldown at ≤ 90°F/hr with one of the following methods: CONTROL Reactor cooldown with Manual Bypass Jack on 1H13-P680-9C	MCR	3	
At approximately 200 psig Reactor pressure, SHUTDOWN one RWCU Pump per SOI 04-1-01-G33-1, IF Both are running.			
SLOWLY OPEN 1G33-F044, RWCU FLTR DMIN BYP VLV on 1H13-P680 while reducing F/D flow with flow controller 1G36-FC-R022A(B) on 1G36-P002.	MCR CTMT 185' RWCU Panel (1A509)	3	Not Required
MAINTAIN a nearly constant system flow rate, (450-500 gpm Is recommended), as indicated on 1G33-FI-R609, RWCU INL FLO, on1H13-P680.	MCR	3	
On 1G36-P002, OBSERVE that holding pump comes on WHEN F/D flow is < 80%.	CTMT 185' RWCU Panel (1A509)	3	Not Required
WHEN filter flow is < 20%, TURN Filter/Hold switch A(B) on 1G36-P002 to HOLD position. OBSERVE the following valves fully Close: • G36-F001A(B) F/D Inlet • G36-F002A(B) F/D Inlet • G36-F003A(B) F/D Outlet • G36-F004A(B) F/D Outlet	CTMT 185' RWCU Panel (1A509)	3	Not Required
OBSERVE HOLD light on AND FILTER light out on 1G36-P002	CTMT 185' RWCU Panel (1A509)	3	Not Required
PLACE the MANUAL/AUTO selector on controller 1G36-FC-R022A (B) in MANUAL position with controller output at 0% output.	CTMT 185' RWCU Panel (1A509)	3	Not Required
REPEAT Steps 4.6.2a AND 4.6.2b for second F/D.	CTMT 185' RWCU Panel (1A509)	3	Not Required
LOWER system flow rate to < 280 gpm by THROTTLING 1G33F044 as indicated on 1G33FI-R609, RWCU INL FLO, on 1H13-P680.	MCR	3	
TRIP one of the running RWCU pumps	MCR	3	
ESTABLISH 90 to 300 gpm flow as indicated on 1G33-FI-R609, RWCU INL FLO, on 1H13-P680 by THROTTLING the Bypass Valve 1G33F044	MCR	3	
WHEN Reactor pressure is reduced to < 135 psig, THEN at			

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
approximately 40 psig, PLACE one loop of RHR System in SHUTDOWN COOLING mode per SOI 04-1-01-E12-2.			
RACK OUT RHR A/B PMP Breaker, 152-1509/1606	Control Bldg. 111' SWGR Rms 0C202, 0C215	3	**Required**
SHUTDOWN RHR JOCKEY PUMP A/B on 1H13-P871.	MCR	3	
CLOSE F082A/B, RHR JCKY PMP SUCT ISOL VLV, on 1H13-P871.	MCR	3	
CLOSE F064A/B, RHR MIN FLO TO SUPP POOL.	MCR	3	
CLOSE F004A/B, RHR PMP SUCT FM SUPP	MCR	3	
ENSURE OPEN F003A/B, RHR HX OUTL VLV	MCR	3	
ENSURE OPEN F048A/B, RHR HX A BYP VLV.	MCR	3	
CLOSE F047A/B, RHR HX INL VLV.	MCR	3	
CLOSE F428A/B, PRESSURE LOCK ISOL for F024	RHR A/B Pump Rm Aux Bldg 93' (1A103/1A105)	3	**Required**
CLOSE F438A/B, PRESSURE LOCK ISOL for F064	RHR A/B Pump Rm Aux Bldg 93' (1A103/1A105)	3	**Required**
SLOWLY OPEN F020, Manual Flush Valve.	Aux Bldg 119' RCIC Rm (1A204)	3	**Required**
OPEN F006A, RHR PMP A SUCT FM SHUTDN CLG AND MONITOR RHR HR A STM press indicator for rise in pressure.	MCR	3	
VENT Shutdown Cooling suction header as follows: (a) OPEN F323. (b) OPEN F399. (c) WHEN a solid stream of water is observed out of vent line, THEN CLOSE F399. (d) CLOSE F323.	Aux Bldg 119' RCIC Rm (1A204)	3	**Required**
OPEN F073A, RHR HX A OTBD VENT VLV.	MCR	3	
OPEN F074A, RHR HX A INBD VENT VLV.	MCR	3	
VENT RHR A Heat Exchanger A as follows: (a) OPEN F400A, A RHR HX VENT.	Aux. 139' RHR A/B Rm	3	**Required**

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
(b) OPEN F401A, A RHR HX VENT. (c) WHEN water is observed from vent, THEN CLOSE F401A. (d) CLOSE F400A.	1A303, 1A304/1A306 , 1A307		
OPEN F064A/B. AFTER approximately one minute, THEN CLOSE F064A/B	MCR	3	
WHEN Conductivity as indicated on HX A/B OUT CNDCT, is as low as practical (Should be less than 2.0 µmhos/cm), THEN CLOSE F073A/B, RHR HX A/B OTBD VENT VLV.	MCR	3	
CLOSE F074A/B, RHR HX A/B INBD VENT VLV	MCR	3	
LOCK CLOSED F020, Manual Flush Valve.	Aux Bldg 119' RCIC Rm (1A204)	3	**Required**
CLOSE F048A/B	MCR	3	
OPEN F063A/B, Manual Flush Valve.	RHR A/B Pump Rm Aux Bldg 119' (1A203/1A205)	3	**Required**
OPEN F073A/B, RHR HX A/B OTBD VENT VLV	MCR	3	
OPEN F074A/B, RHR HX A/B INBD VENT VLV	MCR	3	
WHEN Conductivity as indicated on HX A/B OUT CNDCT, is as low as practical (Should be less than 2.0 µmhos/cm), THEN CLOSE F073A/B, RHR HX A/B OTBD VENT VLV.	MCR	3	
CLOSE F074A/B, RHR HX A/B INBD VENT VLV	MCR	3	
LOCK CLOSED F063A/B, Manual Flush Valve.	RHR A/B Pump Rm Aux Bldg 93' (1A203/1A205)	3	**Required**
OPEN F048A.	MCR	3	
OPEN F047A.	MCR	3	
ENSURE OPEN F003A.	MCR	3	
ENSURE Shutdown Cooling Isolation Logic is reset by PRESSING NSSSS INBD ISOL RESET pushbutton AND NSSSS OTBD ISOL RESET pushbutton on 1H13-P601.	MCR	3	
PLACE Standby Service Water A System in service to RHR A Heat Exchanger on 1H13-P870 as follows. START SSW Pump A per SOI 04-1-01-P41-1.	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
OPEN P41-F014A, SSW INL TO RHR HX A. ENSURE OPEN P41-F068A, SSW OUTL FM RHR HX A. START RHR RM A FAN COIL UNIT.			
ENSURE OPEN F010, SHUTDN CLG MAN SUCT VLV.	MCR	3	
ENSURE CLOSED F040, RHR TO RADWST OTBD SHUTOFF VLV.	MCR	3	
ENSURE CLOSED F049, RHR TO RADWST INBD SHUTOFF VLV.	MCR	3	
OPEN F020, Manual Flush Valve approximately 3 turns. Valve May be opened further IF required for level control.	Aux Bldg 119' RCIC Rm (1A204)	3	**Required**
OPEN F008, RHR SHUTDN CLG OTBD SUCT VLV	MCR	3	
OPEN F009, RHR SHUTDN CLG INBD SUCT VLV as follows; ENSURE breaker 52-163137 is CLOSE position OPEN F009, RHR SHUTDN CLG INBD SUCT VLV MONITOR Reactor water level WHILE 1E12F009 AND 1E12F020 are OPEN. PERFORM IMMEDIATELY the next step 4.1.2.b(14) IF a rise in Reactor water level is NOT desired.	MCR	3	
LOCK CLOSED F020, Manual Flush Valve.	Aux Bldg 119' RCIC Rm (1A204)	3	**Required**
NOTIFY Radwaste Operators to be prepared for Reactor water flush to Waste Surge tank.	MCR Radwaste Building 118' Radwaste Control Room (0R241)	3	**Required**
OPEN F203, RHR SYS FLUSH TO LIQ RADWST by the following handswitches to OPEN: F203 SVA-RHR SYS FLUSH TO LIQ RADWST (1H13-P870-3C) F203 SVB-RHR SYS FLUSH TO LIQ RADWST (1H13-P870-8C)	MCR	3	
ENSURE CLOSED F070A/B, Manual RHR Drain Valve	Aux Bldg 93' Corridor (1A101)	3	**Required**
OPEN F072A/B, RHR Drain Valve	RHR A/B Pump Rm Aux Bldg 93' (1A103/1A105)	3	**Required**

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
SLOWLY OPEN F070A, RHR Drain Valve approximately one turn to start flow to Radwaste. IF "RHR A DISCH PRESS ABNORMAL" annunciator alarms while warming RHR A, THEN CLOSE F047A AND F048A to prevent draining of downstream piping.	Aux Bldg 93' Corridor (1A101)	3	**Required**
THROTTLE F070A/B to warm RHR Pump A/B at less than 100°F/hr until RHR DISCH TO RADWST ON RHR TEMP recorder is 200°F OR within 100°F of RX water temp, whichever is less.	Aux Bldg 93' Corridor (1A101)	3	**Required**
LOCK CLOSED F070A, RHR Drain Valve.	Aux Bldg 93' Corridor (1A101)	3	**Required**
LOCK CLOSED F072A, RHR Drain Valve.	RHR A/B Pump Rm Aux Bldg 93' (1A103/1A105)	3	**Required**
CLOSE F203, RHR SYS FLUSH TO LIQ RADWST by TAKING the following handswitches to CLOSE: F203 SVA-RHR SYS FLUSH TO LIQ RADWST (1H13-P870-3C) F203 SVB-RHR SYS FLUSH TO LIQ RADWST (1H13-P870-8C)	MCR	3	
RACK IN RHR A/B PMP Breaker, 152-1509/1606	Control Bldg. 111' SWGR Rms 0C202, 0C215	3	**Required**
NOTIFY Chemistry AND Radiation Protection that possibility of a crud burst Could occur due to starting of RHR pump in SDC mode	MCR	3	
START OR ENSURE running RHR RM A FAN COIL UNIT on 1H13-P870.	MCR	3	
ENSURE CLOSED F064A, RHR A MIN FLO TO SUPP POOL.	MCR	3	
ENSURE RHR JOCKEY PUMP A is shutdown.	MCR	3	
ENSURE CLOSED F082A, RHR A JCKY PMP SUCT ISOL VLV.	MCR	3	
ENSURE CLOSED F004A, RHR A SUCT FM SUPP POOL.	MCR	3	
ENSURE OPEN the following valves: (a) F010 (Concurrent Verification Required) (b) F008 (c) F009 as follows; (1) ENSURE breaker 52-163137 is CLOSE position (2) ENSURE OPEN F009, RHR SHUTDN CLG SUCT VLV (d) F006A	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
(e) F047A (f) F048A			
CLOSE F003A, RHR HX A OUTL VLV .	MCR	3	
ENSURE CLOSED B21-F065A, FW INL SHUTOFF VLV.	MCR	3	
START RHR PMP A AND IMMEDIATELY FULLY OPEN one of the following valves: (a) E12-F053A, RHR A SHUTDN CLNG RTN TO FW (b) E12-F037A, RHR A TO CTMT POOL (c) E12-F042A, RHR A INJ SHUTOFF VLV	MCR	3	
MONITOR RHR HX A differential temperature on RHR TEMPERATURE RECORDER as follows: RHR HX A Point 1(inlet) - Point 5(outlet)	MCR	3	
ESTABLISH a cool down rate of less than 90°F/hr, as follows: Slowly JOG OPEN F003A to allow flow through heat exchanger, AND MONITOR cooldown rate. THROTTLE one of the following valves to maintain RHR pump flow ~8600 gpm AND RHR heat exchanger flow ~8200 gpm: IF flow is through F053A, THEN THROTTLE F053A AS LONG AS flow through valve is maintained < 8550 gpm.	MCR	3	
IF E12-F003A is closed while in SHUTDOWN COOLING, THEN MONITOR REACTOR COOLANT TEMPERATURE using the following indications: REACTOR RECIRC LOOP A/B suction temperature (IF recirc pump(s) running) RWCU REGENERATIVE HEAT EXCHANGER INLET temperature (IF RWCU pump(s) are running.) Point 5 of RHR TEMPERATURE RECORDER. Installed thermocouple suspended above Reactor core.	MCR	3	
WHEN F003A valve is full open AND additional cooling is required, THEN SLOWLY THROTTLE CLOSE F048A as needed to establish desired cooldown rate.	MCR	3	
WHEN F048A valve is full closed, THEN, IF desired, THROTTLE F003A to MAINTAIN desired coolant temperature OR SDC flow while MAINTAINING ≥ 3000 gpm flow. F048A may be fully opened to reduce cooldown rate but CANNOT be left in a throttled position UNTIL F003A is full open.	MCR	3	
SELECT "Shutdown Cooling-RHR A" OP GUIDE on PDS computer. The guide Should be left on-screen OR icon'd WHEN the respective shutdown cooling loop is in service until Reactor Coolant has been stabilized at desired temperature so that the guide Will warn operators IF Shutdown Cooling parameters are out of range	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
LOG Reactor coolant temperature on Data Sheet I of 03-1-01-3 OR other applicable IOI. TAKE temperatures as required by 03-1-01-3 during cooldown AND CONTINUE to take readings once per hour WHEN temperature is stable.	MCR	3	
LOG temperatures for SSW/RHR HX AND reactor coolant on log similar to Attachment I to ENSURE SSW temperature does NOT exceed design temperatures. (Ref. CR1997-0282) IF SSW A auto start signal from RHR A pump running is defeated by Temporary Alteration, THEN START/STOP SSW A AND B fans as necessary to MAINTAIN SSW A Supply temp. (E12-R601, pt. 12) between 50 AND 75 deg.	MCR	3	
IF RPV level control via RWCU blowdown is unavailable, THEN RPV level control May be established by USING E12-F073A AND E12-F074A RHR heat exchanger vent to establish RPV level control, AND THROTTLE OPEN E12-F073A AND E12-F074A as required to establish AND maintain the desired RPV level. MONITOR RPV level while reject is in progress.	MCR	3	
IF desired to add water to Reactor with SDC in operation WHEN in Modes 4 OR 5, THEN PERFORM the following: THROTTLE OPEN, F020. WHEN desired Reactor Vessel Level is reached, THEN LOCK CLOSED F020.	Aux Bldg 119' RCIC Rm (1A204)	4, 5	Not Required
IOI 03-1-01-3 Continued			
At approximately 120 psig, PERFORM the following: **TRANSFER RWCU to Pre-pump mode per SOI 04-1-01-G33-1.**			
PLACE NSSSS OTBD MOV TEST handswitch on 1H13-P601-19B to the TEST position. VERIFY that "RX DIV 1 ISOL SYS OOSVC" annunciator (1H13-P601-19A-H3) Alarms.	MCR	3	
PLACE NSSSS INBD MOV TEST handswitch on 1H13-P601-18B to the TEST position. VERIFY that "RX DIV 2 ISOL SYS OOSVC" annunciator (1H13- P601-19A-G3) Alarms.	MCR	3	
SECURE RWCU blowdown flow per Section 5.1 of this instruction.	MCR	3	
STOP running RWCU pump AND leave F044, RWCU FLTR DMIN BYP VLV THROTTLED SLIGHTLY OPEN.	MCR	3	
CLOSE the following valves AND proceed to Step 4.4.2g without delay: F250 RWCU SPLY TO RWCU HXS 1H13-P870-3C F251 RWCU SPLY TO RWCU HXS 1H13-P870-9C	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
F252 RWCU HX RTN TO RWCU PMPS 1H13-P870-9C F253 RWCU HX RTN TO RWCU PMPS 1H13-P870-3C F255 RWCU FLTR/DMIN INL FM RWCU PMP 1H13-P870-5C			
OPEN OR CHECK OPEN the following valves: F004 PMP SUCT CTMT OTBD ISOL 1H13-P680 F001 PMP SUCT DRWL INBD ISOL 1H13-P680 F254 RWCU FLTR/DMIN INL FM RWCU HX 1H13-P870-5C F256 RWCU HX INL FM RWCU PMP 1H13-P870-5C	MCR	3	
CLOSE OR CHECK CLOSED F044, RWCU FLTR DMIN BYP VLV; And THEN RESTART one RWCU pump AND JOG OPEN F044 to establish flow greater than 90 gpm but less than 300 gpm.	MCR	3	
START one RWCU the pump AND THROTTLE F044 to achieve a system flow greater than 90 gpm, But less than 300 gpm as indicated on FI-R609, RWCU INL FLO. IF performing system warm-up. THEN MAINTAIN minimum flow, AVOIDING low flow trip.	MCR	3	
IF desired, START a second pump as follows: START the pump AND THROTTLE F044 to maintain 300 - 500 gpm system flow as indicated on FI-R609, RWCU INL FLO, with Both Pumps running.	MCR	3	
IF desired, ESTABLISH RWCU blowdown flow in accordance with Section of this instruction			All areas previously addressed for this evolution
IF desired, PLACE F/Ds in service in accordance with Section 4.5 of this instruction.			All areas previously addressed for this evolution
PLACE NSSSS OTBD MOV TEST handswitch on 1H13-P601-19B to the NORM position. VERIFY that "RX DIV 1 ISOL SYS OOSVC" annunciator (1H13-P601-19A-H3) Clears.	MCR	3	
PLACE NSSSS INBD MOV TEST handswitch on 1H13-P601-18B to the NORM position. VERIFY that "RX DIV 2 ISOL SYS OOSVC" annunciator (1H13- P601-19A-G3) Clears.	MCR	3	
IOI 03-1-01-3 Continued			
SHUTDOWN the running Condensate Booster Pump AND CLOSE respective discharge valve per SOI 04-1-01-N19-1.	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
IF scheduled, THEN PERFORM 06-OP-1B21-R-0010 (Att. I AND/OR II) WHEN reactor pressure is between 50 AND 100 psig			Not required to be performed to Shut down and Cooldown the plant.
At approximately 60 psig Reactor pressure, PERFORM the following: VERIFY that RCIC system isolates automatically. IMMEDIATELY NOTIFY CAS, SAS, OR Security Island that RCIC is not available (non-functional). COMPLETE shutdown of RCIC system per SOI 04-1-01-E51-1.	MCR	3	
WHEN cooldown using Bypass Valves is no longer desired AND Shutdown Cooling is in service, THEN CLOSE the Bypass Valves as follows: SET the TURB STM PRESS DEMAND setpoint approximately 100 psig above Reactor pressure using the PRESS REF "RAISE" OR "LOWER" pushbuttons on 1H13-P680-9C. DEENERGIZE the Manual Bypass Valve Controller by depressing the MAN BYP CONT "OFF" pushbutton on 1H13-P680-9C.	MCR	3	
IF MSIV's are open AND stroke time testing was NOT scheduled, THEN PERFORM the following: CLOSE the following Inboard MSIVs: • B21-F022A • B21-F022B • B21-F022C • B21-F022D WHEN Main Steam Line pressure downstream of MSIVs is near zero psig, THEN CLOSE the following Outboard MSIVs: • B21-F028A • B21-F028B • B21-F028C • B21-F028D CLOSE B21-F016 CLOSE B21-F019	MCR	3	
NOTIFY Radiation Protection that the Reactor is to be vented to Drywell sump AND REQUEST Drywell survey after Head Vent realignment.	MCR	3	
WHEN Reactor coolant temperature is less than 210°F, THEN REALIGN Reactor Head Vents on 1H13-P601 as follows: OPEN 1B21-F001, RPV OTBD VENT VLV.	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
PEN 1B21-F002, RPV INBD VENT VLV. CLOSE 1B21-F005, RPV VENT TO MSL A.			

Control Room ventilation systems have adequate engineered safety/design features in place to preclude a Control Room evacuation due to the release of a hazardous gas. Therefore, the Control Room is not included in this assessment or in Table H-2.

Table A-3 & H-2 Results

Table A-3 & H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode**
Control Building 111' SWGR Rms (0C202, 0C215)	3
Auxiliary Building 93' RHR A Pump Room (1A103)	3
Auxiliary Building 93' RHR B Pump Room (1A105)	3
Auxiliary Building 93' Corridor (1A101)	3
Auxiliary Building 119' Corridor (1A201)	3
Auxiliary Building 119' RHR A Pump Room (1A203)	3
Auxiliary Building 119' RHR B Pump Room (1A205)	3
Auxiliary Building 119' RCIC Room (1A204)	3
Auxiliary Building 139' RHR A Room (1A303, 1A304)	3
Auxiliary Building 139' RHR B Room (1A306, 1A307)	3
Radwaste Building 118' Radwaste Control Room (0R241)	3

GNRO-2018/00048, ENCLOSURE

ATTACHMENT 3

GGNS EAL BASIS DOCUMENT MARKUP

Grand Gulf Nuclear Station
EAL Technical Basis

Table of Contents

1.0 INTRODUCTION

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Grand Gulf Nuclear Station (GGNS). It should be used to facilitate review of the GGNS EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of 10-S-01-1, Activation of the Emergency Plan, 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 or less in all cases of conditions present. Use of this document for assistance is not intended to delay the emergency classification.

Because the information in a basis document can affect emergency classification decision-making (e.g., the Emergency 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).

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 GGNS Emergency Plan.

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

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

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

Subsequently, Revision 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels

for Non-Passive Reactors," November 2012 (ref. 4.1.1), GGNS conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

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

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

The primary fission product barriers are:

A. Fuel Clad Barrier (FCB): The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.

B. Reactor Coolant System Barrier (RCB): The RCS Barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping out to and including the isolation valves.

C. Containment Barrier (CNB): The Containment Barrier includes the drywell, the containment, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. 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 GGNS EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
  - EALs applicable under <u>any</u> plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
  - EALs applicable only under <u>hot</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Startup, or Power Operation mode.
  - EALs applicable only under <u>cold</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refueling 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 and subcategories:

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

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

EAL Groups, Categories and Subcategories

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

2.5 Technical Bases Information

EAL technical bases are provided in Attachment 1 for each EAL according to EAL group (Any, Hot, Cold), EAL category (A, C, E, F, H and S) 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:

Category Letter & Title

Subcategory Number & Title

Initiating Condition (IC)

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

EAL Identifier (enclosed in rectangle)

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

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

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

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

Classification (enclosed in rectangle):

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

EAL (enclosed in rectangle)

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

Mode Applicability

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

Definitions:

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

Basis:

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

Reference(s):

Source documentation from which the EAL is derived

2.6 Operating Mode Applicability

1 Power Operation
Reactor is critical and the mode switch is in RUN

2 Startup
The mode switch is in REFUEL (with all reactor vessel head closure bolts fully tensioned) or STARTUP/HOT STANDBY

3 Hot Shutdown
The mode switch is in SHUTDOWN and average reactor coolant temperature is >200°F

4 Cold Shutdown
The mode switch is in SHUTDOWN and average reactor coolant temperature is ≤ 200°F

5 Refueling
The mode switch is in REFUEL or SHUTDOWN with one or more reactor vessel head closure bolts are less than fully tensioned

DEF Defueled
RPV contains no irradiated fuel

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

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Emergency 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.

EAL matrices should be read from left to right, from General Emergency to Unusual Event, and top to bottom. Declaration decisions should be independently verified before declaration is made except when gaining this verification would exceed the 15 minute declaration requirement. Place keeping should be used on all EAL matrices.

3.1.1 Classification Timeliness

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, "Interim Staff Guidance, Emergency Planning for Nuclear Power Plants" (ref. 4.1.8).

3.1.2 Valid Indications

All emergency classification assessments shall be based upon valid indications, reports or conditions. A valid indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy. For example, verification could be accomplished through an instrument channel check, response on related or redundant indicators, or direct observation by plant personnel.

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 10 § CFR 50.72 (ref. 4.1.4).

3.1.5 Classification Based on Analysis

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

3.1.6 Emergency 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. The NEI 99-01 EAL scheme provides the Emergency Director with the ability to classify events and conditions based upon judgment using EALs that are consistent with the Emergency Classification Level (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 in the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

3.2 Classification Methodology

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

When assessing an EAL that specifies a time duration for the off-normal condition, the "clock" for the EAL time duration runs concurrently with the emergency classification process "clock." For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01 (ref. 4.1.8).

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 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 units, an Alert should be declared.

3.2.2 Consideration of Mode Changes During Classification

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

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

3.2.3 Classification of Imminent Conditions

Although EALs provide specific thresholds, the Emergency 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 emergency classification levels, this approach is particularly important at the higher emergency classification levels since it provides additional time for implementation of protective measures.

3.2.4 Emergency Classification Level Upgrading and Termination

An ECL may be terminated when the event or condition that meets the classified IC and EAL no longer exists, and other site-specific termination requirements are met.

3.2.5 Classification of Short-Lived Events

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

3.2.6 Classification of Transient Conditions

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

<u>EAL momentarily met during expected plant response</u> - In instances in which 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 the high pressure ECCS systems fail to automatically start. The plant enters an inadequate core cooling condition (a potential loss of both the Fuel Clad and RCS Barriers). If an operator manually starts a high pressure ECCS system in accordance with an EOP step and clears the inadequate core cooling condition prior to an emergency declaration, then the classification should be based on the ATWS only.

It is important to stress that the 15-minute emergency classification assessment period (process clock) is not a "grace period" during which a classification may be delayed to allow the performance of a corrective action that would obviate the need to classify the event. Emergency classification assessments must be deliberate and timely, with no undue delays. The provision discussed above addresses only those rapidly evolving situations when an operator is able to take a successful corrective action prior to the Emergency 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 NEI 99-01 Revision 6, Methodology for the Development of Emergency Action Levels for Non-Passive Reactors, ADAMS Accession Number ML12326A805

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

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

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

4.1.5 10 § CFR 50.73 License Event Report System

4.1.6 GGNS Technical Specifications Table 1.1-1, Modes

4.1.7 GGNS Offsite Dose Calculation Manual (ODCM)

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

4.1.9 GGNS Emergency Plan

4.1.10 GGNS UFSAR 9.1.4.2.10.4 Storage of Fuel at the Independent Spent Fuel Storage Installation (ISFSI)

4.1.11 GGNS UFSAR 9.1.4.2.10 Description of Fuel Transfer

4.1.12 SOPP-018-1 Shutdown Operations Protection Plan

4.1.13 10-S-01-12 Radiological Assessment and Protective Action Recommendations

4.2 Implementing

4.2.1 10-S-01-1 Activation of the Emergency Plan

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

4.2.3 GGNS EAL Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

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

Alert

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

Confinement Boundary

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the GGNS ISFSI, Confinement Boundary is defined as the Holtec System Multi-Purpose Canister (MPC) (ref. 4.1.10).

Containment Closure

The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Containment Closure is established when either Primary or Secondary Containment integrity is established (ref. 4.1.12).

Emergency Action Level (EAL)

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

Emergency Classification Level (ECL)

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

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

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.

Fission Product Barrier Threshold

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

Flooding

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

General Emergency

Events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG 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 GGNS 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 GGNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the SECURITY 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).

Independent Spent Fuel Storage Installation (ISFSI)

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

Initiating Condition (IC)

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

~~**Owner Controlled Area (OCA)**~~

~~For the purposes of classification, the Security area between the OCA detection fence and the PROTECTED AREA boundary known as the Security Owner Controlled Area (SOCA) in the GGNS Emergency Plan (ref. 4.1.9).~~

Projectile

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

Protected Area

An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled. (ref. 4.1.9).

RCS Intact

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

Refueling Pathway

Reactor cavity (well), upper containment pool, fuel transfer canal, and auxiliary building fuel pools, but **not** including the reactor vessel, comprise the refueling pathway (ref. 4.1.11).

Restore

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

Safety System

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

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

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

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

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

Security Condition

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

Security Owner Controlled Area (SOCA)

The SOCA is the area demarcated as a Vehicle Barrier System (VBS) consisting of passive elements including a series of large concrete blocks on the inside of a delay fence with early warning capabilities. The SOCA is the area between the SOCA Fence and the PROTECTED AREA Boundary.

Site Area Emergency

Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE 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 PAG exposure levels beyond the SITE BOUNDARY.

Site Boundary

That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor (ref. 4.1.13)

Unisolable

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

Unplanned

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

Unusual Event

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

Valid

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

Visible Damage

Damage to a SAFETY SYSTEM train 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 SAFETY SYSTEM train.

5.2 Abbreviations/Acronyms

°F ..... Degrees Fahrenheit
° ..... Degrees
AB ..... Auxiliary Building
AC ..... Alternating Current
AOP ..... Abnormal Operating Procedure
APRM ..... Average Power Range Meter
ARI ..... Alternate Rod Insertion
ATWS ..... Anticipated Transient Without Scram
BWR ..... Boiling Water Reactor
BWROG ..... Boiling Water Reactor Owners Group
CDE ..... Committed Dose Equivalent
CFR ..... Code of Federal Regulations
CNB ..... Containment Barrier
CS ..... Core Spray
CTMT ..... Containment
DEF ..... Defueled
DBA ..... Design Basis Accident
DC ..... Direct Current
D/G ..... Diesel Generator
EAL ..... Emergency Action Level
ECCS ..... Emergency Core Cooling System
ECL ..... Emergency Classification Level
EOF ..... Emergency Operations Facility
EOP ..... Emergency Operating Procedure
EPA ..... Environmental Protection Agency
EPG ..... Emergency Procedure Guideline
EPP ..... Emergency Plan Procedure
ERO ..... Emergency Response Organization
ESF ..... Engineered Safety Feature
FAA ..... Federal Aviation Administration
FBI ..... Federal Bureau of Investigation
FCB ..... Fuel Clad Barrier
FEMA ..... Federal Emergency Management Agency
FSAR ..... Final Safety Analysis Report
GE ..... General Emergency

HCTL ..... Heat Capacity Temperature Limit
HPCS ..... High Pressure Core Spray
IC ..... Initiating Condition
IPEEE ..... Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI ..... Independent Spent Fuel Storage Installation
K_{eff} ..... Effective Neutron Multiplication Factor
LCO ..... Limiting Condition of Operation
LER ..... Licensee Event Report
LOCA ..... Loss of Coolant Accident
LPCS ..... Low Pressure Core Spray
LRW ..... Liquid Radwaste
LWR ..... Light Water Reactor
MPC ..... Maximum Permissible Concentration/Multi-Purpose Canister
MPH ..... Miles Per Hour
mR, mRem, mrem, mREM ..... milli-Roentgen Equivalent Man
MSCRWL ..... Minimum Steam Cooling RPV Water Level
MSIV ..... Main Steam Isolation Valve
MSL ..... Main Steam Line
MW ..... Megawatt
NEI ..... Nuclear Energy Institute
NEIC ..... National Earthquake Information Center
NESP ..... National Environmental Studies Project
NORAD ..... North American Aerospace Defense Command
(NO)UE ..... Notification of Unusual Event
NPP ..... Nuclear Power Plant
NRC ..... Nuclear Regulatory Commission
NSSS ..... Nuclear Steam Supply System
OBE ..... Operating Basis Earthquake
~~OCA ..... Owner Controlled Area~~
ODCM ..... Offsite Dose Calculation Manual
ONEP ..... Off-Normal Event Procedure
ORO ..... Offsite Response Organization
PA ..... Protected Area
PAG ..... Protective Action Guideline
PB ..... Pushbutton
PCIS ..... Primary Containment Isolation System

PRA/PSA ..... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PSIG ..... Pounds per Square Inch Gauge
R ..... Roentgen
RCB ..... RCS Barrier
RCIC ..... Reactor Core Isolation Cooling
RCS ..... Reactor Coolant System
Rem, rem, REM ..... Roentgen Equivalent Man
RETS ..... Radiological Effluent Technical Specifications
RHR ..... Residual Heat Removal
RPS ..... Reactor Protection System
RPT ..... Recirculation Pump Trip
RPV ..... Reactor Pressure Vessel
RWCU ..... Reactor Water Cleanup
SAP ..... Severe Accident Procedure
SAR ..... Safety Analysis Report
SBO ..... Station Blackout
SCBA ..... Self-Contained Breathing Apparatus
SOCA ..... Security Owner Controlled Area
SPDS ..... Safety Parameter Display System
SRO ..... Senior Reactor Operator
SRV ..... Safety Relief Valve
SSE ..... Safe Shutdown Earthquake
TEDE ..... Total Effective Dose Equivalent
TAF ..... Top of Active Fuel
TSC ..... Technical Support Center
UFSAR ..... Updated Final Safety Analysis Report
USGS ..... United States Geological Survey

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

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

GGNS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
AU1.1	AU1	1, 2
AU1.2	AU1	3
AU2.1	AU2	1
AA1.1	AA1	1
AA1.2	AA1	2
AA1.3	AA1	3
AA1.4	AA1	4
AA2.1	AA2	1
AA2.2	AA2	2
AA2.3	AA2	3
AA3.1	AA3	1
AA3.2	AA3	2
AS1.1	AS1	1
AS1.2	AS1	2
AS1.3	AS1	3
AS2.1	AS2	1
AG1.1	AG1	1
AG1.2	AG1	2
AG1.3	AG1	3

GGNS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
AG2.1	AG2	1
CU1.1	CU1	1
CU1.2	CU1	2
CU2.1	CU2	1
CU3.1	CU3	1
CU3.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1, 2, 3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA3.1	CA3	1, 2
CA6.1	CA6	1
CS1.1	CS1	1
CS1.2	CS1	2
CS1.3	CS1	3
CG1.1	CG1	2
EU1.1	EU1	1
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1, 2, 3
HU2.1	HU2	1

GGNS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
HU3.1	HU3	1
HU3.2	HU3	2
HU3.3	HU3	3
HU3.4	HU3	4
HU4.1	HU4	1
HU4.2	HU4	2
HU4.3	HU4	3
HU4.4	HU4	4
HU7.1	HU7	1
HA1.1	HA1	1, 2
HA5.1	HA5	1
HA6.1	HA6	1
HA7.1	HA7	1
HS1.1	HS1	1
HS6.1	HS6	1
HS7.1	HS7	1
HG7.1	HG7	1
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	1
SU4.2	SU3	2
SU5.1	SU4	1, 2, 3
SU6.1	SU5	1

GGNS	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
SU6.2	SU5	2
SU7.1	SU6	1, 2, 3
N/A	SU7	1, 2
SA1.1	SA1	1
SA3.1	SA2	1
SA6.1	SA5	1
SA8.1	SA9	1
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1
SG1.1	SG1	1
SG1.2	SG8	1

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases

Category A – Abnormal Rad Levels / Rad Effluent

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

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

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

Events of this category pertain to the following subcategories:

1. Radiological Effluent

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

2. Irradiated Fuel Event

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

3. Area Radiation Levels

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

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AU1.1 Unusual Event

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is 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 due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

Table A-1 Effluent Monitor Classification Thresholds

Release Point		GE	SAE	Alert	UE
Gaseous	SBGT A/B	8.1E+2 Ci/sec	8.1E+1 Ci/sec	8.1E+0 Ci/sec	6.7E-2 Ci/sec
	CTMT Vent	6.4E+2 Ci/sec	6.4E+1 Ci/sec	6.4E+0 Ci/sec	6.7E-2 Ci/sec
	Radwaste Building Vent	5.1E+1 Ci/sec	5.1E+0 Ci/sec	5.1E-1 Ci/sec	6.7E-2 Ci/sec
	Turbine Building Vent	1.3E+1 Ci/sec	1.3E+0 Ci/sec	1.3E-1 Ci/sec	6.7E-2 Ci/sec
	Fuel Handling (Aux BLDG) Vent	8.6E+3 Ci/sec	8.6E+2 Ci/sec	8.6E+1 Ci/sec	6.7E-2 Ci/sec
Liquid	Radwaste	----	----	----	7.33E+5 cpm

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

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

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

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

This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways as well as radioactivity releases that cause effluent radiation monitor readings to exceed 2 times the limit established by a radioactivity discharge permit. Such releases are typically associated with planned batch releases from non-continuous release pathways (e.g., radwaste, waste gas).

Escalation of the emergency classification level would be via IC AA1.

Reference(s):

1. IAS-04-1-01-D17-1 Process Radiation Monitoring
2. Offsite Dose Calculation Manual
3. XC-Q1D17-17001 Grand Gulf Nuclear Station (GGNS) Radiological Effluent EAL Threshold Values
4. NEI 99-01 AU1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AU1.2 Unusual Event

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

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

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

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

Escalation of the emergency classification level would be via IC AA1.

Reference(s):

1. Offsite Dose Calculation Manual
2. NEI 99-01 AU1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AA1.1 Alert

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is 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 due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

Note 4 The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table A-1 Effluent Monitor Classification Thresholds

Release Point		GE	SAE	Alert	UE
Gaseous	SBGT A/B	8.1E+2 Ci/sec	8.1E+1 Ci/sec	8.1E+0 Ci/sec	6.7E-2 Ci/sec
	CTMT Vent	6.4E+2 Ci/sec	6.4E+1 Ci/sec	6.4E+0 Ci/sec	6.7E-2 Ci/sec
	Radwaste Building Vent	5.1E+1 Ci/sec	5.1E+0 Ci/sec	5.1E-1 Ci/sec	6.7E-2 Ci/sec
	Turbine Building Vent	1.3E+1 Ci/sec	1.3E+0 Ci/sec	1.3E-1 Ci/sec	6.7E-2 Ci/sec
	Fuel Handling (Aux BLDG) Vent	8.6E+3 Ci/sec	8.6E+2 Ci/sec	8.6E+1 Ci/sec	6.7E-2 Ci/sec
Liquid	Radwaste	----	----	----	7.33E+5 cpm

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

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

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

Escalation of the emergency classification level would be via IC AS1.

Reference(s):

1. IAS-04-1-01-D17-1 Process Radiation Monitoring
2. XC-Q1D17-17001 Grand Gulf Nuclear Station (GGNS) Radiological Effluent EAL Threshold Values
3. NEI 99-01 AA1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AA1.2 Alert

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

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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

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

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

Escalation of the emergency classification level would be via IC AS1.

Reference(s):

1. 10-S-01-12 Radiological Assessment and Protective Action Recommendations
2. NEI 99-01 AA1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AA1.3 Alert

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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

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

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

This EAL is assessed per the ODCM (ref. 2)

Escalation of the emergency classification level would be via IC AS1.

Reference(s):

1. 10-S-01-12 Radiological Assessment and Protective Action Recommendations
2. Offsite Dose Calculation Manual
3. NEI 99-01 AA1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AA1.4 Alert

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

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

(Notes 1, 2)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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

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

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

Escalation of the emergency classification level would be via IC AS1.

Reference(s):

1. 10-S-01-12 Radiological Assessment and Protective Action Recommendations
2. NEI 99-01 AA1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AS1.1 Site Area Emergency

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is 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 due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table A-1 Effluent Monitor Classification Thresholds

Release Point		GE	SAE	Alert	UE
Gaseous	SBGT A/B	8.1E+2 Ci/sec	8.1E+1 Ci/sec	8.1E+0 Ci/sec	6.7E-2 Ci/sec
	CTMT Vent	6.4E+2 Ci/sec	6.4E+1 Ci/sec	6.4E+0 Ci/sec	6.7E-2 Ci/sec
	Radwaste Building Vent	5.1E+1 Ci/sec	5.1E+0 Ci/sec	5.1E-1 Ci/sec	6.7E-2 Ci/sec
	Turbine Building Vent	1.3E+1 Ci/sec	1.3E+0 Ci/sec	1.3E-1 Ci/sec	6.7E-2 Ci/sec
	Fuel Handling (Aux BLDG) Vent	8.6E+3 Ci/sec	8.6E+2 Ci/sec	8.6E+1 Ci/sec	6.7E-2 Ci/sec
Liquid	Radwaste	----	----	----	7.33E+5 cpm

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

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

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

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

Reference(s):

1. IAS-04-1-01-D17-1 Process Radiation Monitoring
2. XC-Q1D17-17001 Grand Gulf Nuclear Station (GGNS) Radiological Effluent EAL Threshold Values
3. NEI 99-01 AS1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AS1.2 Site Area Emergency

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

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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

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

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

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

Reference(s):

1. 10-S-01-12 Radiological Assessment and Protective Action Recommendations
2. NEI 99-01 AS1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AS1.3 Site Area Emergency

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

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

(Notes 1, 2)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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

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

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

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

Reference(s):

1. 10-S-01-12 Radiological Assessment and Protective Action Recommendations
2. NEI 99-01 AS1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AG1.1 General Emergency

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is 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 due to actions to isolate the release path, then the effluent monitor reading is **no** longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table A-1 Effluent Monitor Classification Thresholds

Release Point		GE	SAE	Alert	UE
Gaseous	SBGT A/B	8.1E+2 Ci/sec	8.1E+1 Ci/sec	8.1E+0 Ci/sec	6.7E-2 Ci/sec
	CTMT Vent	6.4E+2 Ci/sec	6.4E+1 Ci/sec	6.4E+0 Ci/sec	6.7E-2 Ci/sec
	Radwaste Building Vent	5.1E+1 Ci/sec	5.1E+0 Ci/sec	5.1E-1 Ci/sec	6.7E-2 Ci/sec
	Turbine Building Vent	1.3E+1 Ci/sec	1.3E+0 Ci/sec	1.3E-1 Ci/sec	6.7E-2 Ci/sec
	Fuel Handling (Aux BLDG) Vent	8.6E+3 Ci/sec	8.6E+2 Ci/sec	8.6E+1 Ci/sec	6.7E-2 Ci/sec
Liquid	Radwaste	----	----	----	7.33E+5 cpm

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

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

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

Reference(s):

1. IAS-04-1-01-D17-1 Process Radiation Monitoring
2. XC-Q1D17-17001 Grand Gulf Nuclear Station (GGNS) Radiological Effluent EAL Threshold Values
3. NEI 99-01 AG1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AG1.2 General Emergency

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

Note 4: The pre-calculated effluent monitor values presented in EALs AA1.1, AS1.1 and AG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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

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

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

Reference(s):

1. 10-S-01-12 Radiological Assessment and Protective Action Recommendations
2. NEI 99-01 AG1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

AG1.3 General Emergency

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

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

(Notes 1, 2)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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

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

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

Reference(s):

1. 10-S-01-12 Radiological Assessment and Protective Action Recommendations
2. NEI 99-01 AG1

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: UNPLANNED loss of water level above irradiated fuel

EAL:

AU2.1 Unusual Event

UNPLANNED water level drop in the REFUELING PATHWAY as indicated by Fuel Pool Drain Tank low water level alarm, visual observation, water level drop in Upper Ctmt Pools, Aux Bldg Fuel Pools or the Fuel Transfer Canal

AND

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

- Ctmt 209 Airlock (1D21K630)
- Ctmt Fuel Hdlg Area (1D21K626)
- Aux Bldg Fuel Hdlg Area(1D21K622)

Mode Applicability:

All

Definition(s):

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

REFUELING PATHWAY- Reactor cavity (well), upper containment pool, fuel transfer canal, and auxiliary building fuel pools, but **not** including the reactor vessel, comprise the refueling pathway.

Basis:

This IC addresses a drop 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 drop 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 a rise 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 rise due to planned evolutions such as lifting of the reactor vessel head or movement of a fuel assembly. Note that this EAL is applicable only in cases where the elevated reading is due to an UNPLANNED loss of water level.

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

Escalation of the emergency classification level would be via IC AA2.

Reference(s):

1. 05-1-02-II-8 High Radiation During Fuel Handling
2. 04-1-01-D21-1 Area Radiation Monitoring System
3. UFSAR 12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation
4. NEI 99-01 AU2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AA2.1 Alert

IMMINENT uncovery of irradiated fuel in the REFUELING PATHWAY

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY- The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the GGNS ISFSI, Confinement Boundary is defined as the Holtec System Multi-Purpose Canister (MPC).

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.

REFUELING PATHWAY- Reactor cavity (well), upper containment pool, fuel transfer canal, and auxiliary building fuel pools, but **not** including the reactor vessel, comprise the refueling pathway.

Basis:

This IC addresses events that have caused IMMINENT or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the REFUELING PATHWAY. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant. This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC EU1.

This EAL escalates from AU2.1 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 a rise 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 with Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC AS1.

Reference(s):

1. NEI 99-01 AA2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity

AND

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

- Ctmt Vent (P601-19A-G9)
- FH Area Vent (P601-19A-C11)
- Ctmt 209 Airlock (P844-1A-A1)
- Ctmt Fuel Hdlg Area (P844-1A-A3)
- Aux Bldg Fuel Hdlg Area (P844-1A-A4)

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY- The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the GGNS ISFSI, Confinement Boundary is defined as the Holtec System Multi-Purpose Canister (MPC).

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

Basis:

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

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

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

Escalation of the emergency classification level would be via IC AS1.

Reference(s):

1. 05-1-02-II-8 High Radiation During Fuel Handling
2. 04-1-01-D21-1 Area Radiation Monitoring System
3. UFSAR 12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation
4. Offsite Dose Calculation Manual
5. NEI 99-01 AA2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AA2.3 Alert

Lowering of spent fuel pool level to 193 ft. (Level 2) on G41R040A(B)

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL addresses events that have caused 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.

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

Escalation of the emergency classification level would be via IC AS1 or AS2.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2 – ~~192~~ 193 ft. 2 $^{1}/_{8}$ in. rounded to 193 ft. for readability) and SFP level at the top of the fuel racks (Level 3 – 183~~2~~ ft. 2 $^{1}/_{8}$ in. rounded to 183 ft. for readability) (ref. 1).

G41R040A(B) Spent Fuel Pool Level Instrument is not located in the Control Room. The display cabinets are located in the 148' Control Building in the Lower Cable Spreading Room.

Reference(s):

1. 05-S-01-FSG-011 Alternate Spent Fuel Pool Makeup and Cooling
2. NEI 99-01 AA2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AS2.1 Site Area Emergency

Lowering of spent fuel pool level to 183 ft. (Level 3) on G41R040A(B)

Mode Applicability:

All

Definition(s):

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

Basis:

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

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

Escalation of the emergency classification level would be via IC AG1 or AG2.

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2 – ~~192~~ 193 ft. 2 $^1/_8$ in. rounded to 193 ft. for readability) and SFP level at the top of the fuel racks (Level 3 – 183~~2~~ ft. 2 $^1/_8$ in. rounded to 183 ft. for readability) (ref. 1).

G41R040A(B) Spent Fuel Pool Level Instrument is not located in the Control Room. The display cabinets are located in the 148' Control Building in the Lower Cable Spreading Room.

Reference(s):

1. 05-S-01-FSG-011 Alternate Spent Fuel Pool Makeup and Cooling
2. NEI 99-01 AS2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

AG2.1 General Emergency

Spent fuel pool level **cannot** be restored to at least 183 ft. (Level 3) on G41R040A(B) for ≥ 60 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

All

Definition(s):

None

Basis:

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

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

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2 – ~~192~~ 193 ft. 2 $^{1}/_{8}$ in. rounded to 193 ft. for readability) and SFP level at the top of the fuel racks (Level 3 – 183~~2~~ ft. 2 $^{1}/_{8}$ in. rounded to 183 ft. for readability) (ref. 1).

G41R040A(B) Spent Fuel Pool Level Instrument is not located in the Control Room. The display cabinets are located in the 148' Control Building in the Lower Cable Spreading Room.

Reference(s):

1. 05-S-01-FSG-011 Alternate Spent Fuel Pool Makeup and Cooling
2. NEI 99-01 AG2

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

AA3.1 Alert

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

- Control Room (SD21-K600)
- Central Alarm Station (by survey)

Mode Applicability:

All

Definition(s):

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

Basis:

Areas that meet this threshold include the Control Room (CR) and the Central Alarm Station (CAS). The Control Room is monitored for excessive radiation by one detector, SD21-K600 (ref. 1). The CAS is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations. While the CAS is in the Control Room Envelope, there are no permanently installed area radiation monitors in CAS that may be used to assess this EAL threshold. Therefore, this threshold is evaluated using local radiation survey for this area.

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 rise in radiation levels and determine if another IC may be applicable.

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

Reference(s):

1. 06-IC-1D21-R-1001 Area Radiation Monitoring Calibration
2. NEI 99-01 AA3

Category: A – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

AA3.2 Alert

An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table A-3 room or area (Note 5)

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

Table A-3 Safe Operation & Shutdown Rooms/Areas

Room/Area	Mode
Control Building 111' SWGR Rms (0C202, 0C215)	3
Auxiliary Building 93' RHR A Pump Room (1A103)	3
Auxiliary Building 93' RHR B Pump Room (1A105)	3
Auxiliary Building 93' Corridor (1A101)	3
Auxiliary Building 119' Corridor (1A201)	3
Auxiliary Building 119' RHR A Pump Room (1A203)	3
Auxiliary Building 119' RHR B Pump Room (1A205)	3
Auxiliary Building 119' RCIC Room (1A204)	3
Auxiliary Building 139' RHR A Room (1A303, 1A304)	3
Auxiliary Building 139' RHR B Room (1A306, 1A307)	3
Radwaste Building 118' Radwaste Control Room (0R241)	3

Mode Applicability:

3 – Hot Shutdown

Definition(s):

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

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

Basis:

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 rise in radiation levels and determine if another IC may be applicable.

For AA3.2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the higher 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 rise occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 3.
- The higher 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 access control measures are of a conservative or precautionary nature, and would not actually prevent or IMPEDE a required action.

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

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

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

EAL AA3.2 mode applicability has been limited to the mode limitations of Table A-3 (Mode 3 **only**).

Reference(s):

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

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 (4 - Cold Shutdown, 5 - Refueling, DEF – Defueled).

The events of this category pertain to the following subcategories:

1. RPV Level

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

2. Loss of Emergency AC Power

Loss of vital plant electrical power can compromise plant SAFETY SYSTEM operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite power sources for 4.16 KV ESF buses.

3. RCS Temperature

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

4. Loss of Vital DC Power

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

5. Loss of Communications

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

6. Hazardous Event Affecting Safety Systems

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

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: UNPLANNED loss of RPV inventory

EAL:

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

Basis:

Grand Gulf is equipped with multiple RPV water level instruments including: Wide Range, Fuel Zone, Shutdown Range, Upset Range, and Narrow Range (ref. 1). Multiple instruments on different reference and variable legs should be monitored. The Upset Range and Shutdown Range instruments share a common reference leg; therefore, Narrow Range instruments should be routinely monitored when relying on Shutdown or Upset Range instrument as the primary indication.

With the plant in Cold Shutdown, RPV water level is normally maintained above the RPV low level scram setpoint of 11.4 in. (ref. 2). However, if RPV level is being controlled below the RPV low level scram setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

With the plant in Refueling mode, RPV water level is normally maintained at or above the reactor vessel flange. Technical Specifications require at least 22 ft 8 in. of water above the top of the reactor vessel flange in the refueling cavity during refueling operations (ref. 3). The RPV flange is at approximately 212 in. on the Shutdown Range. (ref. 4).

This EAL addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor RPV level concurrent

with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

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

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

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

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

Reference(s):

1. 02-S-01-40 EP Technical Bases
2. 05-S-01-EP-2 RPV Control
3. Technical Specifications 3.9.6
4. 07-S-14-413 RPV Disassembly
5. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: UNPLANNED loss of RPV inventory

EAL:

CU1.2 Unusual Event

RPV water level **cannot** be monitored

AND EITHER

- UNPLANNED rise in **any** Table C-1 sump or pool level due to a loss of RPV inventory
- Visual observation of UNISOLABLE RCS leakage

Table C-1 Sumps/Pool
• Drywell equipment drain sump • Drywell floor drain sump • CTMT equipment drain sump • CTMT floor drain sump • Suppression Pool • RHR A, B, C, HPCS, LPCS, RCIC room sumps • Auxiliary Building floor drain sump

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

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

UNPLANNED-. A parameter changes 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:

In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. RPV level in the Refueling mode is normally monitored using the Shutdown Range instrument which is re-spanned to indicate water level in the refuel cavity and the Core Plate d/p instrument which is re-spanned and re-scaled to indicate water level (ref. 1).

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Rise in drywell equipment drain sump level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 2, 3). An Auxiliary Building sump level rise may also be indicative of RCS inventory losses external to the Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

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

Refueling evolutions that lower RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level lowering 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 RPV level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels (Table C-1). Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RPV.

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

Reference(s):

1. 03-1-01-5 Refueling
2. 04-1-02-1H13-P601 Alarm Response Instruction Panel 1H13-P601
3. 04-1-02-1H13-P680 Alarm Response Instruction Panel 1H13-P680
4. 05-S-01-EP-4 Auxiliary Building Control
5. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Significant Loss of RPV inventory

EAL:

CA1.1	**Alert**
Loss of RPV inventory as indicated by RPV water level < -42 in. (Level 2)	

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

None

Basis:

The threshold RPV water level of -42 in. is the Level 2 actuation setpoint for HPCS and RCIC. Although RCIC cannot restore RPV inventory in the cold condition, the Level 2 actuation setpoint is operationally significant and is indicative of a loss of RPV inventory significantly below the low RPV water level scram setpoint specified in CU1.1 (ref. 1, 2).

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

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

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

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

Reference(s):

1. Technical Specifications Table 3.3.5.1-1, Emergency Core Cooling System Instrumentation
2. 04-1-02-1H13-P601-16A-A4 Alarm Response Instruction Panel 1H13-P601 panel 16A-A4 RX LVL 2 (-42") LO
3. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Significant Loss of RPV inventory

EAL:

CA1.2 Alert

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

AND EITHER

- UNPLANNED rise in **any** Table C-1 sump or pool level due to a loss of RPV inventory
- Visual observation of UNISOLABLE RCS leakage

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table C-1 Sumps/Pool
• Drywell equipment drain sump • Drywell floor drain sump • CTMT equipment drain sump • CTMT floor drain sump • Suppression Pool • RHR A, B, C, HPCS, LPCS, RCIC room sumps • Auxiliary Building floor drain sump

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

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

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

Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. RPV level in the Refueling mode is normally monitored using the Shutdown Range instrument which is re-spanned to indicate water level in the refuel cavity and the Core Plate d/p instrument which is re-spanned and re-scaled to indicate water level. (ref. 1).

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Rise in drywell equipment drain sump level and drywell floor sump level is the normal method of monitoring and calculating leakage from the RPV (ref. 2, 3). An Auxiliary Building sump level rise may also be indicative of RCS inventory losses external to the Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in suppression pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

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

For this EAL, the inability to monitor RPV 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 RPV.

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

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

Reference(s):

1. 03-1-01-5 Refueling
2. 04-1-02-1H13-P601 Alarm Response Instruction Panel 1H13-P601
3. 04-1-02-1H13-P680 Alarm Response Instruction Panel 1H13-P680
4. 05-S-01-EP-4 Auxiliary Building Control
5. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.1 Site Area Emergency

CONTAINMENT CLOSURE **not** established

AND

RPV water level < -150 in. (Level 1)

Mode Applicability:

4 – Cold Shutdown, 5 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Containment Closure is established when either Primary or Secondary Containment integrity is established.

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

Basis:

The threshold RPV water level of -150 in. is the low-low-low ECCS actuation setpoint (Level 1). The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further lowering of RPV water level and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier (ref. 1, 2).

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

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

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control

functions. The difference in the specified RPV levels of EALs CS1.1 and CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

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

Escalation of the emergency classification level would be via IC CG1 or AG1.

Reference(s):

1. Technical Specifications Table 3.3.5.1-1, Emergency Core Cooling System Instrumentation
2. 04-1-02-1H13-P601-17A-E2 Alarm Response Instruction Panel 1H13-P601 panel 17A-E2 RX LVL 1 (-150") LO
3. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.2 Site Area Emergency CONTAINMENT CLOSURE established **AND** RPV water level < -167 in. (TAF)

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Containment Closure is established when either Primary or Secondary Containment integrity is established.

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

Basis:

When RPV level drops to the top of active fuel (TAF) (an indicated RPV level of -167 in.), core uncovery starts to occur (ref. 1).

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

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

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RPV levels of EALs CS1.1 and CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

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

Escalation of the emergency classification level would be via IC CG1 or AG1.

Reference(s):

1. 02-S-01-40 EP Technical Bases
2. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.3 Site Area Emergency

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

AND

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

- UNPLANNED rise in Suppression Pool level of sufficient magnitude to indicate core uncovery
- Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery
- Containment~~/Drywell~~ High Range Area Radiation Monitor (1D21-~~K648A~~K648B-~~D~~C) ≥ 100 R/hr~~high-high alarm~~

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 – Cold Shutdown, 5 – Refueling

Definition(s):

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.

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

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

Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. RPV level in the Refueling mode is normally monitored using the Shutdown Range instrument which is re-spanned to indicate water level in the refuel cavity and the Core Plate d/p instrument which is re-spanned and re-scaled to indicate water level (ref. 1).

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications. Level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are

indicative of RPV leakage. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in Suppression Pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

In the Refueling Mode, as water level in the RPV lowers, the dose rate above the core will rise, with corresponding indications on area radiation monitors. 100 R/hr is used for this indication ~~A high-high alarm (> 1,000 R/hr)~~ on Containment High Range Radiation Monitors (1D21-K648B and C)~~, or associated computer or recorder points (Ref 6.)) is selected as the control room alarm indication to provide this function~~. These detectors are located on the containment wall in a position to monitor the containment radiation environment above the refueling cavity elevation.

~~In Cold Shutdown Mode, when the core is uncovered, the dose rate in the containment or drywell around or above the core will rise, with corresponding indications on area radiation monitors. A high-high alarm (> 1,000 R/hr) on Drywell/Containment High Range Radiation Monitors (1D21-K648A, B, C, D, or associated computer or recorder points (Ref 6.)) is selected as the control room alarm indication to provide this function. The detectors are located in the drywell and the containment and are in a position to monitor the drywell and containment direct radiation environment.~~

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

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

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

The inability to monitor RPV 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 RPV.

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

Escalation of the emergency classification level would be via IC CG1 or AG1

Reference(s):

1. 03-1-01-5 Refueling
2. 04-1-02-1H13-P601 Alarm Response Instruction Panel 1H13-P601
3. 04-1-02-1H13-P680 Alarm Response Instruction Panel 1H13-P680
4. 05-S-01-EP-4 Auxiliary Building Control
5. 06-IC-1D21-R-1002 Containment/Drywell High Range Area Radiation Monitor Calibration
6. NEI 99-01 CS1
7. Calculation J-D21-1, Set Points Determination For High Range DW & Containment Radiation Monitors (D21 System)

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Loss of RPV inventory affecting fuel clad integrity with Containment challenged

EAL:

CG1.1 General Emergency

RPV level < -167 in. (TAF) for ≥ 30 min. (Note 1)

AND

Any Containment Challenge indication, Table C-2

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • Drywell or containment hydrogen concentration > 4% • UNPLANNED rise in containment pressure • Exceeding one or more Auxiliary Building Control MAX SAFE area radiation levels (EP-4)

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Containment Closure is established when either Primary or Secondary Containment integrity is established.

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.

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:

When RPV level drops below -167 in., core uncovery starts to occur (ref. 1).

Four conditions are associated with a challenge to Containment integrity:

- 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 the containment. However, containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that hydrogen concentration has exceeded the minimum necessary to support a hydrogen burn (4%). The Igniter System is designed to prevent hydrogen accumulation by locally burning hydrogen in a controlled manner as soon as the hydrogen enters the containment atmosphere and reaches the igniters. For high rates of hydrogen production, ignition occurs at the lowest concentration that can support ignition. Following ignition, hydrogen is consumed through formation of diffusion flames where the gas enters the containment, thus controlling hydrogen concentration at approximately 4% (ref. 2).
- Any UNPLANNED rise in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of CONTAINMENT CLOSURE capability. UNPLANNED containment pressure rise indicates CONTAINMENT CLOSURE cannot be assured and the containment cannot be relied upon as a barrier to fission product release.
- Secondary Containment radiation monitors should provide indication of a larger release that may be indicative of a challenge to CONTAINMENT CLOSURE. The MAX SAFE radiation levels are indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in EP-4, Auxiliary Building Control, (ref. 3).

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

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV 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 core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.

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

Reference(s):

1. 02-S-01-40 EP Technical Bases
2. BWROG Emergency Procedure and Severe Accident Guidelines, Revision 3, p. B-16-64
3. 05-S-01-EP-4, Auxiliary Building Control
4. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CG1.2 General Emergency

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

AND

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

- UNPLANNED rise in Suppression Pool level of sufficient magnitude to indicate core uncovery
- Visual observation of UNISOLABLE RCS leakage of sufficient magnitude to indicate core uncovery
- Containment/Drywell High Range Area Radiation Monitor (1D21-~~K648A~~K648B-~~D~~C) ≥ 100R/hr ~~high-high alarm~~

AND

Any Containment Challenge indication, Table C-2

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

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

Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • Drywell or containment hydrogen concentration > 4% • UNPLANNED rise in containment pressure • Exceeding one or more Auxiliary Building Control MAX SAFE area radiation levels (EP-4)

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Containment Closure is established when either Primary or Secondary Containment integrity is established.

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.

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

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

Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. RPV level in the Refueling mode is normally monitored using the Shutdown Range instrument which is re-spanned to indicate water level in the refuel cavity and the Core Plate d/p instrument which is re-spanned and re-scaled to indicate water level (ref. 1).

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications. Level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in Suppression Pool water level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS that cannot be isolated could be indicative of a loss of RPV inventory.

In the Refueling Mode, as water level in the RPV lowers, the dose rate above the core will rise, with corresponding indications on area radiation monitors. 100 R/hr is used for this indication ~~A high-high alarm (> 1,000 R/hr)~~ on Containment High Range Radiation Monitors (1D21-K648B and C)~~, or associated computer or recorder points (Ref 6.)) is selected as the control room alarm indication to provide this function~~. These detectors are located on the containment wall in a position to monitor the containment radiation environment above the refueling cavity elevation.

Four conditions are associated with a challenge to Containment integrity:

- 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 the containment. However, containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that hydrogen concentration has exceeded the minimum necessary to support a hydrogen burn (4%). The Igniter System is designed to prevent hydrogen accumulation by locally burning hydrogen in a controlled manner as soon as the hydrogen enters the containment atmosphere and reaches the igniters. For high rates of hydrogen production, ignition occurs at the lowest concentration that can support ignition. Following ignition, hydrogen is consumed through formation of diffusion flames where the gas enters the containment, thus controlling hydrogen concentration at approximately 4% (ref. 4).
- Any UNPLANNED rise in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of CONTAINMENT CLOSURE capability. UNPLANNED containment pressure rise indicates CONTAINMENT CLOSURE cannot be assured and the containment cannot be relied upon as a barrier to fission product release.
- Secondary Containment radiation monitors should provide indication of a larger release that may be indicative of a challenge to CONTAINMENT CLOSURE. The MAX SAFE radiation levels are indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in EP-4, Auxiliary Building Control, (ref. 5).

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

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV 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 core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.

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

The inability to monitor RPV 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 RPV.

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

Reference(s):

1. 03-1-01-5 Refueling
2. 04-1-02-1H13-P601 Alarm Response Instruction Panel 1H13-P601
3. 04-1-02-1H13-P680 Alarm Response Instruction Panel 1H13-P680
4. BWROG Emergency Procedure and Severe Accident Guidelines, Revision 3, p. B-16-64
5. 05-S-01-EP-4, Auxiliary Building Control
6. 06-IC-1D21-R-1002 Containment/Drywell High Range Area Radiation Monitor Calibration
7. NEI 99-01 CG1
8. Calculation J-D21-1, Set Points Determination For High Range DW & Containment Radiation Monitors (D21 System)

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of ESF AC Power

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

EAL:

CU2.1 Unusual Event

AC power capability, Table C-3, to DIV I and DIV II ESF 4.16 KV buses reduced to a single power source for ≥ 15 min. (Note 1)

AND

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table C-3 AC Power Sources
Offsite • ESF Transformer 11 • ESF Transformer 12 • ESF Transformer 21 **Onsite** • DIV I DG (DG 11) • DIV II DG (DG 12)

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, DEF - Defueled

Definition(s):

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

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

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

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

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

Basis:

The HPCS bus (DIV III) is not credited because it only supplies power to the HPCS pump and associated loads, not any long term decay heat removal systems. In particular, suppression pool cooling mechanisms would be essential subsequent to a station blackout.

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

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as an Alert because of the greater time available to restore another power source to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

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

- A loss of all offsite power with a concurrent failure of all but one emergency ESF power source (e.g., an onsite diesel generator).
- A loss of all offsite power and loss of all emergency ESF power sources (e.g., onsite diesel generators) with a single train of emergency ESF buses being back-fed from the unit main generator.
- A loss of emergency ESF power sources (e.g., onsite diesel generators) with a single train of emergency ESF buses being fed from an offsite power source.

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

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

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

Reference(s):

1. UFSAR Figure 8.1-001 Main One Line Diagram
2. UFSAR section 8.1 Electric Power Introduction
3. UFSAR section 8.3 Onsite Power
4. UFSAR section 8A Loss of all AC Power
5. 05-1-02-I-4 Loss of AC Power
6. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of ESF AC Power

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

EAL:

CA2.1	**Alert**
Loss of **all** offsite and **all** onsite AC power to DIV I and DIV II ESF 4.16 KV buses for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, DEF - Defueled

Definition(s):

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

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

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

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

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

Basis:

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. Mitigative strategies using other power sources (HPCS DIV III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as a Site Area Emergency because of the greater time available to restore an ESF bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition represents an actual or potential substantial degradation of the level of safety of the plant.

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

Escalation of the emergency classification level would be via IC CS1 or AS1.

This EAL is the cold condition equivalent of the hot condition EAL SS1.1.

Reference(s):

1. UFSAR Figure 8.1-001 Main One Line Diagram
2. UFSAR section 8.1 Electric Power Introduction
3. UFSAR section 8.3 Onsite Power
4. UFSAR section 8A Loss of all AC Power
5. 05-1-02-I-4 Loss of AC Power
6. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED rise in RCS temperature

EAL:

CU3.1 Unusual Event

UNPLANNED rise in RCS temperature to > 200°F ~~due to loss of decay heat removal capability~~

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

*Containment Closure is established when either Primary or Secondary Containment integrity is establish*ed.

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

Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F) (ref. 1, 2). In the absence of reliable RCS temperature indication, classification is based on the concurrent loss of reactor vessel level indications per EAL CU3.2.

This IC addresses an UNPLANNED rise 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 also refer to EAL CA3.1.

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

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

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

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

Reference(s):

1. Technical Specifications Table 1.1-1
2. 03-1-01-3 Plant Shutdown
3. 04-1-01-E12-2 Shutdown Cooling and Alternate Decay Heat Removal
4. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED rise in RCS temperature

EAL:

CU3.2	**Unusual Event**
Loss of **all** RCS temperature and RPV water level indication for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5- Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Containment Closure is established when either Primary or Secondary Containment integrity is established.

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:

This EAL addresses the inability to determine RCS temperature and level, and represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Emergency Director should also refer to EAL CA3.1.

This EAL reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

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

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

Reference(s):

1. 02-S-01-40 EP Technical Bases
2. Technical Specifications Table 1.1-1
3. 03-1-01-3 Plant Shutdown
4. 04-1-01-E12-2 Shutdown Cooling and Alternate Decay Heat Removal
5. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.1 Alert

UNPLANNED rise in RCS temperature to > 200°F for > Table C-4 duration (Note 1)

OR

UNPLANNED RPV pressure rise > 10 psig

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table C-4 RCS Heat-up Duration Thresholds

RCS Status	CONTAINMENT CLOSURE Status	Heat-up Duration
Intact	N/A	60 min.*
Not intact	established	20 min.*
	not established	0 min.

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

CONTAINMENT CLOSURE - The actions taken to secure containment (Primary or Secondary) and their associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

Containment Closure is established when either Primary or Secondary Containment integrity is established.

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:

In the absence of reliable RCS temperature indication, classification should be based on the RCS pressure rise criteria when the RCS is intact in Mode 4 or based on time to boil data when in Mode 5 or the RCS is not intact in Mode 4.

This 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.

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

The RCS Heat-up Duration Thresholds table addresses a rise in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not intact. The 20-minute criterion was included to allow time for operator action to address the temperature rise.

The RCS Heat-up Duration Thresholds table also addresses a rise 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 rise without a substantial degradation in plant safety.

Finally, in the case where there is a rise in RCS temperature, the RCS is not intact and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because 1) the evaporated reactor coolant may be released directly into the containment atmosphere and subsequently to the environment, and 2) there is reduced reactor coolant inventory above the top of irradiated fuel.

The RCS pressure rise threshold provides a pressure-based indication of RCS heat-up in the absence of RCS temperature monitoring capability.

Escalation of the emergency classification level would be via IC CS1 or AS1.

Reference(s):

1 Technical Specifications Table 1.1-1
2. 03-1-01-3 Plant Shutdown
3. 04-1-01-E12-2 Shutdown Cooling and Alternate Decay Heat Removal
4. NEI 99-01 CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – Loss of Vital DC Power

Initiating Condition: Loss of Vital DC power for 15 minutes or longer

EAL:

CU4.1	**Unusual Event**
Indicated voltage is < 105 VDC on required vital 125 VDC buses 11DA and 11DB for ≥ 15 min. (Note 1)	

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

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

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

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

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

Basis

Vital DC buses 11DA and 11DB feed the Division 1 and Division 2 loads respectively. The Division 1 and Division 2 batteries each have 61 cells with a design minimum of 1.72 volts/cell. These cell voltages yield minimum design bus voltages of 104.92 VDC (rounded to 105 VDC) (ref. 1, 2).

This IC addresses a loss of vital DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or refueling mode. In these modes, the core decay heat load has been significantly reduced, and coolant system temperatures and pressures are lower; these conditions raise the time available to restore a vital DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, "required" means the vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if

Train A is out-of-service (inoperable) for scheduled outage maintenance work and Train B is in-service (operable), then a loss of Vital DC power affecting Train B would require the declaration of an Unusual Event. A loss of Vital DC power to Train A would not warrant an emergency classification.

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

Depending upon the event, escalation of the emergency classification level would be via IC CA1 or CA3, or an IC in Recognition Category A.

This EAL is the cold condition equivalent of the hot condition EAL SS2.1.

Reference(s):

1. Calculation No: EC-Q1111-14001 Station Division I Battery 1A3 and Division II Battery 1B3 Discharge Capacity during Extended Loss of AC Power
2. UFSAR 8.3.2.1.1 Station DC Power
3. NEI 99-01 CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.1 Unusual Event

Loss of **all** Table C-5 onsite communication methods

OR

Loss of **all** Table C-5 State and local agency communication methods

OR

Loss of **all** Table C-5 NRC communication methods

Table C-5 Communication Methods

System	Onsite	State/ Local	NRC
Station Radio System	X		
GGNS Plant Phone System	X		
Public Address System	X		
Emergency Notification System (ENS)			X
Commercial Telephone System		X	X
Satellite Phones		X	X
~~INFORM Notification System~~		~~X~~	
Operational Hotline		X	

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling, DEF – Defueled

Definition(s):

None

Basis:

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 State and local agencies and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all State and local agencies of an emergency declaration. The State and local agencies referred to here are the Mississippi Emergency Management Agency, Claiborne County Civil Defense, Mississippi Highway Safety Patrol, Claiborne County Sheriff's Department, Louisiana Department of Environmental Quality, Tensas Parish Sheriff's Office, and the Louisiana Governor's Office of Homeland Security and Emergency Preparedness.

The third EAL condition addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

This EAL is the cold condition equivalent of the hot condition EAL SU7.1.

Reference(s):

1. GGNS Emergency Plan Section 7.5, Communications Systems
2. 04-S-01-R61-1 Plant Communications
3. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting Safety Systems

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

EAL:

CA6.1 Alert

The occurrence of **any** Table C-6 hazardous event

AND

Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode

AND EITHER:

- Event damage has caused indications of degraded performance to the second train of the SAFETY SYSTEM needed for the current operating mode
- Event damage has resulted in VISIBLE DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode

(Notes 9, 10)

Note 9: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then emergency classification is **not** warranted.

Note 10: If the hazardous event **only** resulted in VISIBLE DAMAGE, with **no** indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is **not** warranted.

Table C-6 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

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

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

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

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

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

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

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

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

VISIBLE DAMAGE - Damage to a SAFETY SYSTEM train 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 SAFETY SYSTEM train.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance;

commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. 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. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via IC CS1 or AS1.

This EAL is the cold condition equivalent of the hot condition EAL SA8.1.

Reference(s):

1. EP FAQ 2016-002
2. NEI 99-01 CA6

Category E – Independent Spent Fuel Storage Installation (ISFSI)

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

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

An Unusual Event is declared on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated.

The GGNS ISFSI is located wholly within the plant PROTECTED AREA. Therefore any security event related to the ISFSI is classified under Category H1 security event related EALs.

Category: E - ISFSI

Subcategory: Confinement Boundary

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Unusual Event

Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask (HI-STORM overpack) > **EITHER** of the following:

- 60 mrem/hr (gamma + neutron) on the top of the overpack
- 600 mrem/hr (gamma + neutron) on the side of the overpack (excluding inlet and outlet ducts)

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY- The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the GGNS ISFSI, Confinement Boundary is defined as the Holtec System Multi-Purpose Canister (MPC).

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

Basis:

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

The existence of "damage" is determined by radiological survey. The specified EAL threshold values correspond to 2 times the cask technical specification values. The technical specification (licensing bases document) multiple of "2 times", which is also used in Recognition Category A IC AU1, is used here to distinguish between non-emergency and emergency conditions (ref. 2). The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose

rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

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

Reference(s):

1. UFSAR 9.1.4.2.10.4 Storage of Fuel at the Independent Spent Fuel Storage Installation
2. GGNS HI-STORM 100 10 CFR 72.212 Evaluation Report Licensing Basis Document, Revision 10, Section 4.2.4 (Section 5.7) Radiation Protection Program
3. NEI 99-01 E-HU1

Category F – Fission Product Barrier Degradation

EAL Group: Hot Conditions (RCS temperature > 200°F); 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 (FCB): The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.

B. Reactor Coolant System Barrier (RCB): The RCS Barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping out to and including the isolation valves.

C. Containment Barrier (CNB): The Containment Barrier includes the drywell, the containment, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. Containment Barrier thresholds are used as criteria for escalation of the Emergency Classification Level (ECL) from an 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. "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

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

Site Area Emergency:
Loss or potential loss of any two barriers

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

The logic used for emergency classification based on fission product barrier monitoring should reflect the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier.

- Unusual Event ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.
- For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC AG1 has been exceeded.
- The fission product barrier thresholds specified within a scheme reflect plant-specific GGNS 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, an interfacing system, or outside of the containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered to be RCS leakage.
- At the Site Area Emergency level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

Category: F - Fission Product Barrier Degradation

Subcategory: N/A

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

EAL:

FA1.1	**Alert**
Any loss or **any** potential loss of either Fuel Clad or RCS barrier (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 lists the fission product barrier thresholds, bases and references.

At the Alert classification level, Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1

Reference(s):

1. NEI 99-01 FA1

Category: F - Fission Product Barrier Degradation

Subcategory: N/A

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

EAL:

FS1.1	**Site Area Emergency**
Loss or potential loss of **any** two barriers (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

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

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 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.

Reference(s):

1. NEI 99-01 FS1

Category: F - Fission Product Barrier Degradation

Subcategory: N/A

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

EAL:

FG1.1 General Emergency
Loss of **any** two barriers
AND
Loss or potential loss of the third barrier (Table F-1)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 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

Reference(s):

1. NEI 99-01 FG1

Table F-1 Fission Product Barrier Threshold Matrix & Bases

Table F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Containment). The table is structured so that each of the three barriers occupies adjacent columns. Each fission product barrier column is further divided into two columns; one for Loss thresholds and one for Potential Loss thresholds.

The first column of the table (to the left of the Fuel Clad Loss column) lists the categories (types) of fission product barrier thresholds. The fission product barrier categories are:

A. RPV Water Level

B. RCS Leak Rate

C. Containment Conditions

D. Containment Radiation / RCS Activity

E. Containment Integrity or Bypass

F. Emergency Director Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word "None" is entered in the cell.

Thresholds are assigned sequential numbers within each barrier column beginning with number one (ex., FCB1, FCB2...FCB6).

If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of Table F-1, locates the likely category and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine if a threshold has been exceeded. If a threshold has not been exceeded, the EAL-user proceeds to the next likely category and continues review of the thresholds in the new category

If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded, only that one barrier is lost or potentially lost. The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if containment radiation is sufficiently high, a Loss of the Fuel

Clad and RCS Barriers and a Potential Loss of the Containment Barrier can occur. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, and FA1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad Barrier threshold bases appear first, followed by the RCS Barrier and finally the Containment Barrier threshold bases. In each barrier, the bases are given according to category Loss followed by category Potential Loss beginning with Category A, then B,..., F.

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad Barrier (FCB)		Reactor Coolant System Barrier (RCB)		Containment Barrier (CNB)	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A **RPV Water Level**	FCB1 SAP entry is required	FCB2 RPV water level **cannot** be restored and maintained > -167 in. (TAF) or **cannot** be determined	RCB1 RPV water level **cannot** be restored and maintained > -167 in. (TAF) or **cannot** be determined	None	None	CNB1 SAP entry is required
B **RCS Leak Rate**	None	None	RCB2 UNISOLABLE break in **any** of the following: • Main steam line • RCIC steam Line • RWCU • Feedwater • HPCS RCB3 Emergency Depressurization is required	RCB4 UNISOLABLE primary system leakage that results in exceeding **EITHER**: • One or more EP-4 radiation Operating Limits • One or more EP-4 area temperature Operating Limits	CNB2 UNISOLABLE primary system leakage that results in exceeding **EITHER**: • One or more EP-4 MAX SAFE area radiation levels • One or more EP-4 MAX SAFE area temperature levels	None
C **CTMT Conditions**	None	None	RCB5 Drywell pressure > 1.23 psig due to RCS leakage	None	CNB3 UNPLANNED rapid drop in containment pressure following containment pressure rise CNB4 Containment pressure response **not** consistent with LOCA conditions	CNB5 Containment pressure > 15 psig CNB6 Drywell or containment hydrogen concentration > 4% CNB7 Parameters **cannot** be restored and maintained within the safe zone of the HCTL curve (EP Figure 1)
D **CTMT Rad / RCS Activity**	FCB3 Containment radiation (RITS-K648B or C) > 400 R/hr FCB4 Primary coolant activity > 300 µCi/gm dose equivalent I-131	None	RCB6 Drywell radiation (RITS-K648A or D) > 100 R/hr	None	None	CNB8 Containment radiation (RITS-K648B or C) > 7000 R/hr
E **CTMT Integrity or Bypass**	None	None	None	None	CNB9 UNISOLABLE direct downstream pathway to the environment exists after Containment isolation signal CNB10 Intentional Containment venting per EPs	None
F **Emergency Director Judgment**	FCB5 **Any** condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier	FCB6 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier	RCB7 **Any** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	RCB8 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	CNB11 **Any** condition in the opinion of the Emergency Director that indicates loss of the Containment barrier	CNB12 **Any** condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Loss

Threshold:

FCB1
SAP entry is required

Definition(s):

None

Basis:

Emergency Procedures (EPs) specify entry to the Severe Accident Procedures (SAPs) when core cooling is severely challenged. These EPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined (ref. 1, 2).

The EP conditions represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

This threshold is also a Potential Loss of the Containment barrier (CNB1). Since SAP entry occurs after core uncovery has occurred a Loss of the RCS barrier exists (RCB1). SAP entry, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

The Loss threshold represents the EOP requirement for entry into the SAPs. This is identified in the BWROG EPGs/SAGs when adequate core cooling cannot be assured.

Reference(s):

1. 05-S-01-EP-2 RPV Control
2. 05-S-01-EP-5 RPV Flooding
3. EP FAQ 2015-004
4. NEI 99-01, RPV Water Level Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

FCB2

RPV water level **cannot** be restored and maintained > -167 in. (TAF) or **cannot** be determined

Definition(s):

None

Basis:

An RPV water level instrument reading of -167 in. indicates RPV level is at the top of active fuel (TAF) (ref. 1). When RPV level is at or above the TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EPs specify alternate, more extreme, RPV water level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops to TAF, the level is indicative of a challenge to core cooling and the Fuel Clad barrier.

When RPV water level cannot be determined, EPs require entry to EP-5, RPV Flooding. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained (ref. 2). When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EP-5 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in scram-failure events). If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the fuel clad barrier exists.

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, EALs SA6.1 or SS6.1 will dictate the need for emergency classification.

This water level corresponds to the top of the active fuel and is used in the EOPs to indicate a challenge to core cooling.

The RPV water level threshold is the same as RCS barrier Loss threshold RCB1. Thus, this threshold indicates a Potential Loss of the Fuel Clad barrier and a Loss of the RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

This threshold is considered to be exceeded when, as specified in the site-specific EOPs, RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this Fuel Clad barrier Potential Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

The term "cannot be restored and maintained above" means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value cannot be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active fuel, but does not permit extended operation below the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

Since the loss of ability to determine if adequate core cooling is being provided presents a significant challenge to the fuel clad barrier, a potential loss of the fuel clad barrier is specified.

Reference(s):

1. 05-S-01-EP-2 RPV Control
2. 05-S-01-EP-5 RPV Flooding
3. 05-S-01-EP-2A ATWS RPV Control
4 NEI 99-01 RPV Water Level Potential Loss 2.A

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

FCB3
Containment radiation (RITS-K648B or C) > 400 R/hr

Definition(s):

None

Basis:

The containment radiation monitor reading (425 R/hr rounded to 400 R/hr for readability) corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300 µCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to 1.6% fuel clad damage (ref. 1). 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 RCB6 since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the ECL to a Site Area Emergency.

There is no Fuel Clad barrier Potential Loss threshold associated with CTMT Radiation / RCS Activity.

Reference(s):

1. XC-Q1D21-17001 Grand Gulf Nuclear Station (GGNS) Containment Radiation EAL Threshold Values
2. 04-1-01-D21-1 SOI Area Radiation Monitoring System
3. NEI 99-01 Primary Containment Radiation Fuel Clad Loss 4.A

Barrier: Fuel Clad

Category: D. CTMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

FCB4
Coolant activity > 300 µCi/gm dose equivalent I-131

Definition(s):

None

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.

It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications.

There is no Fuel Clad barrier Potential Loss threshold associated with CTMT Radiation / RCS Activity.

Reference(s):

1. NEI 99-01 RCS Activity Fuel Clad Loss 1.A

Barrier: Fuel Clad

Category: D. CTMT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

FCB5

Any condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad Barrier

Definition(s):

None

Basis:

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

Reference(s):

1. NEI 99-01 Emergency Director Judgment Fuel Clad Loss 6.A

Barrier: Fuel Clad

Category: F. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

FCB6

Any condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad Barrier

Definition(s):

None

Basis:

This threshold addresses any other factors that are to 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.

Reference(s):

1. NEI 99-01 Emergency Director Judgment Potential Fuel Clad Loss 6.A

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

RCB1

RPV water level **cannot** be restored and maintained > -167 in. (TAF) or **cannot** be determined

Definition(s):

None.

Basis:

An RPV water level instrument reading of -167 in. indicates level is at the top of active fuel (TAF) (ref. 1). TAF is significantly lower than the normal operating RPV level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Containment barriers, and initiation of all ECCS. If RPV water level cannot be maintained above TAF, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the lowering level trend. The cause of the loss of RPV inventory is therefore assumed to be a LOCA. By definition, a LOCA event is a Loss of the RCS barrier.

When RPV water level cannot be determined, EOPs require entry to EP-5, RPV Flooding. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained (ref. 2). The instructions in EP-5 specify emergency depressurization of the RPV, which is defined to be a Loss of the RCS barrier (RCS Loss RCB3).

Note that EP-2A, ATWS RPV Control, may require intentionally lowering RPV water level to -167 in. and control level between the Minimum Steam Cooling RPV Water Level (MSCRWL) and the top of active fuel (ref. 3). Under these conditions, a high-power ATWS event exists and requires at least a Site Area Emergency classification in accordance with the System Malfunction - RPS Failure EALs.

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, EALs SA6.1 or SS6.1 will dictate the need for emergency classification.

This water level corresponds to the top of active fuel and is used in the EOPs to indicate a challenge to core cooling.

The RPV water level threshold is the same as Fuel Clad barrier Potential Loss threshold FCB2. Thus, this threshold indicates a Loss of the RCS barrier and Potential Loss of the Fuel Clad barrier and that appropriately escalates the emergency classification level to a Site Area Emergency.

This threshold is considered to be exceeded when, as specified in the site-specific EOPs, RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this RCS barrier Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

The term, "cannot be restored and maintained above," means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value cannot be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active fuel, but does not permit extended operation beyond the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

There is no RCS barrier Potential Loss threshold associated with RPV Water Level.

Reference(s):

1. 05-S-01-EP-2 RPV Control
2. 05-S-01-EP-5 RPV Flooding
3. 05-S-01-EP-2A ATWS RPV Control
4. NEI 99-01 RPV Water Level RCS Loss 2.A

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

RCB2

UNISOLABLE break in **any** of the following:

- Main steam line
- RCIC steam line
- RWCU
- Feedwater
- HPCS

Definition(s):

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

Basis:

Failure to isolate the leak, within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

The conditions of this threshold include required containment isolation failures allowing a flow path to the environment. A release pathway outside containment exists when flow is not prevented by downstream isolations. In the case of a failure of both isolation valves to close but in which no downstream flowpath exists, emergency declaration under this threshold would not be required. Similarly, if the emergency response requires the normal process flow of a system outside containment (e.g., EP requirement to bypass MSIV low RPV water level interlocks and maintain the main condenser as a heat sink using main turbine bypass valves), the threshold is not met. The combination of these threshold conditions represent the loss of both the RCS and Containment (see Loss CNB9) barriers and justifies declaration of a Site Area Emergency (i.e., Loss or Potential Loss of any two barriers).

Even though RWCU and Feedwater systems do not contain steam, they are included in the list because an UNISOLABLE break could result in the high-pressure discharge of fluid that is flashed to steam from relatively large volume systems directly connected to the RCS.

Even though the High Pressure Core Spray (HPCS) injects into the RCS, it is included in this EAL due to the potential for an inter-system loss of coolant back flowing from the discharge lines (via failed isolation valves and check valves) and out through a break in the piping. A HPCS failure that does not result in back flow of RCS and out through a break should not be considered as meeting the EAL threshold.

Large high-energy lines that rupture outside containment can discharge significant amounts of inventory and jeopardize the pressure-retaining capability of the RCS until they are isolated. If it is determined that the ruptured line cannot be promptly isolated, remotely or locally,, the RCS barrier Loss threshold is met.

Reference(s):

1. NEI 99-01 RCS Leak Rate RCS Loss 3.A

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

RCB3 Emergency Depressurization is required

Definition(s):

None

Basis:

Emergency Depressurization in accordance with the EOPs (ref. 1, 2) is indicative of a loss of the RCS barrier. If Emergency RPV Depressurization is performed, the plant operators are directed to open safety relief valves (SRVs). Even though the RCS is being vented into the suppression pool, a Loss of the RCS barrier exists due to the diminished effectiveness of the RCS to retain fission products within its boundary.

EP-2 RPV Control - Emergency Depressurization allows terminating the depressurization if necessary to maintain RCIC as an injection source. This would require closing the SRVs. Even though the SRVs may be reclosed, this threshold is still met due to the requirement for an Emergency Depressurization having been met (ref. 2).

Reference(s):

1. 05-S-01-EP-2 RPV Control - Emergency Depressurization
2. EP FAQ 2015-003
3. NEI 99-01 RCS Leak Rate RCS Loss 3.B

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

RCB4

UNISOLABLE primary system leakage that results in exceeding **EITHER**:

- One or more EP-4 area radiation Operating Limits
- One or more EP-4 area temperature Operating Limits

Definition(s):

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

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

Basis:

Failure to isolate the leak, within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

The presence of elevated general area temperatures or radiation levels in the Secondary Containment may be indicative of UNISOLABLE primary system leakage outside the containment. The EP-4 entry condition values define this RCS threshold because they are the Operating Limits (maximum normal operating values) and signify the onset of abnormal system operation. When parameters reach this level, equipment failure or mis-operation may be occurring. Elevated parameters may also adversely affect the ability to gain access to or operate equipment within the affected area. The locations into which the primary system discharge is of concern correspond to the areas addressed in EP-4, Auxiliary Building Control (ref. 1).

In general, multiple indications should be used to determine if a primary system is discharging outside containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the Auxiliary Building since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room FLOODING, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the Secondary Containment.

Potential loss of RCS based on primary system leakage outside the containment is determined from EOP temperature or radiation EP-4 Operating Limits (Max Normal Operating values) in

areas such as main steam line tunnel, RCIC, etc., which indicate a direct path from the RCS to areas outside containment.

An EP-4 Operating Limit value is the highest value of the identified parameter expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

The indicators reaching the threshold barriers and confirmed to be caused by RCS leakage from a primary system warrant an Alert classification. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a reduction in the steam or water being discharged through an unisolated break in the system.

An UNISOLABLE leak which is indicated by EP-4 Operating Limit values escalates to a Site Area Emergency when combined with Containment Barrier Loss thresholds CNB 2 or CNB9 (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

Reference(s):

1. 05-S-01-EP-4 Auxiliary Building Control
2. NEI 99-01 RCS Leak Rate RCS Potential Loss 3.A

Barrier: Reactor Coolant System

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

RCB5

Drywell pressure > 1.23 psig due to RCS leakage

Definition(s):

None

Basis:

The drywell high pressure scram setpoint is an entry condition to EP-2, RPV Control, and EP-3, Containment Control (ref. 1, 2). Normal containment pressure control functions (e.g., operation of drywell and containment cooling, vent using containment vessel purge, etc.) are specified in EP-3 in advance of less desirable but more effective functions (e.g., operation of containment sprays, etc.).

In the design basis, containment pressures above the drywell high pressure scram setpoint are assumed to be the result of a high-energy release into the containment for which normal pressure control systems are inadequate or incapable of reversing the rising pressure trend. Pressures of this magnitude, however, can be caused by non-LOCA events such as a loss of drywell cooling or inability to control containment vent/purge (ref. 3).

The threshold phrase "...due to RCS leakage" focuses the barrier failure on the RCS instead of the non-LOCA malfunctions that may adversely affect containment pressure. Drywell pressure greater than 1.23 psig with corollary indications (e.g., drywell temperature, indications of loss of RCS inventory) should therefore be considered a Loss of the RCS barrier. Loss of drywell cooling that results in pressure greater than 1.23 psig should not be considered an RCS barrier Loss.

The 1.23 psig value is the drywell high pressure setpoint which indicates a LOCA by automatically initiating the ECCS or equivalent makeup system.

There is no RCS barrier Potential Loss threshold associated with CTMT Conditions.

Reference(s):

1. 05-S-01-EP-2 RPV Control
2. 05-S-01-EP-3 Containment Control
3. UFSAR Section 6.2.1, Containment Functional Design
4. NEI 99-01 Primary Containment Pressure RCS Loss 1.A

Barrier: Reactor Coolant System

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CTMT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

RCB6
Drywell radiation (RITS-K648A or D) > 100 R/hr

Definition(s):

None

Basis:

The drywell radiation monitor reading (150 R/hr rounded to 100 R/hr for readability) corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits (ref. 1). This value is lower than that specified for Fuel Clad Barrier Loss threshold FCB3 since it indicates a loss of the RCS Barrier only.

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

Reference(s):

1. XC-Q1D21-17001 Grand Gulf Nuclear Station (GGNS) Containment Radiation EAL Threshold Values
2. 04-1-01-D21-1 SOI Area Radiation Monitoring System
3. NEI 99-01 Primary Containment Radiation RCS Loss 4.A

Barrier: Reactor Coolant System

Category: D. CTMT Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

RCB7

Any condition in the opinion of the Emergency Director that indicates loss of the RCS Barrier

Definition(s):

None

Basis:

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

Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Loss 6.A

Barrier: Reactor Coolant System

Category: F. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

RCB8

Any condition in the opinion of the Emergency Director that indicates potential loss of the RCS Barrier

Definition(s):

None

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.

Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Potential Loss 6.A

Barrier: Containment

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

CNB1
SAP entry is required

Definition(s):

None

Basis:

EPs specify entry to the SAPs when core cooling is severely challenged. These EPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined (ref. 1, 2).

The EP conditions represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

This threshold is also a Loss of the Fuel Clad barrier (Loss FCB1). Since SAP entry occurs after core uncovery has occurred a Loss of the RCS barrier exists (Loss RCB1). SAP entry, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

The Potential Loss threshold is identical to the Fuel Clad Loss RPV Water Level threshold FCB1. The Potential Loss requirement for entry into the SAGs indicates adequate core cooling cannot be assured and that core damage is possible. BWR EPGs/SAGs specify the conditions when the EPGs are exited and SAGs are entered. Entry into SAGs is a logical escalation in response to the inability to assure adequate core cooling.

PRA studies indicate that the condition of this Potential Loss threshold could be a core melt sequence which, if not corrected, could lead to RPV failure and greater potential for containment failure. In conjunction with the RPV water level Loss thresholds in the Fuel Clad and RCS barrier columns, this threshold results in the declaration of a General Emergency.

There is no Containment barrier Loss threshold associated with RPV Water Level.

Reference(s):

1. 05-S-01-EP-2 RPV Control
2. 05-S-01-EP-5 RPV Flooding
3. EP FAQ 2015-004
4. NEI 99-01 RPV Water Level PC Potential Loss 2.A

Barrier: Containment

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

CNB2

UNISOLABLE primary system leakage that results in exceeding **EITHER**:

- One or more EP-4 MAX SAFE area radiation levels
- One or more EP-4 MAX SAFE area temperature levels

Definition(s):

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

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

Basis:

Failure to isolate the leak, within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

The presence of elevated general area temperatures or radiation levels in the secondary containment may be indicative of UNISOLABLE primary system leakage outside the containment. The MAX SAFE values define this Containment barrier threshold because they are indicative of problems in the Secondary Containment that are spreading and pose a threat to achieving a safe plant shutdown. This threshold addresses problematic discharges outside containment that may not originate from a high-energy line break. The locations into which the primary system discharge is of concern correspond to the areas addressed in EP-4, Auxiliary Building Control (ref. 1).

In general, multiple indications should be used to determine if a primary system is discharging outside containment. For example, a high area temperature condition does not necessarily indicate that a primary system is discharging into the secondary containment since this may be caused by a fire or loss of area cooling. Conversely, a high area temperature condition in conjunction with other indications (e.g. room FLOODING, high area radiation levels, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the secondary containment.

The Max Safe area temperature values and the Max Safe area radiation values are each the highest value of these parameters at which neither: (1) equipment necessary for the safe shutdown of the plant will fail, nor (2) personnel access necessary for the safe shutdown of the

plant will be precluded. EOPs utilize these temperatures and radiation levels to establish conditions under which RPV depressurization is required.

There is no Containment barrier Potential Loss threshold associated with RCS Leak Rate.

Reference(s):

1. 05-S-01-EP-4 Auxiliary Building Control
1. NEI 99-01 RCS Leak Rate PC Loss 3.C

Barrier: Containment

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

CNB3

UNPLANNED rapid drop in containment pressure following containment pressure rise

Definition(s):

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

Basis:

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

This threshold relies on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

Reference(s):

1. NEI 99-01 Primary Containment Conditions PC Loss 1.A

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Loss

Threshold:

CNB4

Containment pressure response **not** consistent with LOCA conditions

Definition(s):

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

Basis:

Containment pressure should rise as a result of mass and energy release into the containment from a LOCA. Thus, containment pressure not rising under these conditions indicates a loss of containment integrity.

These thresholds rely on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

Reference(s):

1. USAR Table 6.2-5, Summary of Short Term Containment Responses to Recirculation Line and Main Steam Line Breaks
2. UFSAR Table 6.2-13, Maximum Calculated Accident for Containment Design
3. NEI 99-01 Primary Containment Conditions PC Loss 1.B

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

CNB5

Containment pressure > 15 psig

Definition(s):

None

Basis:

When the containment pressure exceeds the maximum allowable value (15 psig) (ref. 1), containment venting may be required even if offsite radioactivity release rate limits will be exceeded (ref. 2). This pressure is based on the containment design pressure as identified in the accident analysis. If this threshold is exceeded, a challenge to the containment structure has occurred because assumptions used in the accident analysis are no longer valid and an unanalyzed condition exists. This constitutes a Potential Loss of the Containment barrier even if a containment breach has not occurred.

The threshold pressure is the containment internal design pressure. Structural acceptance testing demonstrates the capability of the containment to resist pressures greater than the internal design pressure. A pressure of this magnitude is greater than those expected to result from any design basis accident and, thus, represent a Potential Loss of the Containment barrier.

Reference(s):

1. UFSAR Table 6.2-13, Maximum Calculated Accident for Containment Design
2. 05-S-01-EP-3 Containment Control
3. NEI 99-01, Primary Containment Conditions PC Potential Loss 1.A

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

CNB6

Drywell or containment hydrogen concentration > 4%

Definition(s):

None

Basis:

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 the containment. However, containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that hydrogen concentration has exceeded the minimum necessary to support a hydrogen burn (4%). The Igniter System is designed to prevent hydrogen accumulation by locally burning hydrogen in a controlled manner as soon as the hydrogen enters the containment atmosphere and reaches the igniters. For high rates of hydrogen production, ignition occurs at the lowest concentration that can support ignition. Following ignition, hydrogen is consumed through formation of diffusion flames where the gas enters the containment, thus controlling hydrogen concentration at approximately 4% (ref. 1).

If hydrogen concentration reaches or exceeds the lower flammability limit, as defined in plant EOPs, in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside the containment, loss of the Containment barrier could occur.

Reference(s):

1. 02-S-01-40 EP Technical Bases (EP-3 step H-3)
2. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.B

Barrier: Containment

Category: C. CTMT Conditions

Degradation Threat: Potential Loss

Threshold:

CNB7

Parameters **cannot** be restored and maintained within the safe zone of the HCTL curve (EP Figure 1)

Definition(s):

None

Basis:

The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool temperature from which Emergency RPV Depressurization will not raise:

- Suppression chamber temperature above the maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized,

  OR

- Suppression chamber pressure above Primary Containment Pressure Limit, while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

The HCTL is a function of RPV pressure, suppression pool temperature and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant and therefore, the inability to maintain plant parameters below the limit constitutes a potential loss of containment.

The term "**cannot** be restored and maintained within~~above~~ " means the parameter value(s) is not able to be brought within the specified limit. The determination requires an evaluation of system performance and availability in relation to the parameter value and trend. A threshold prescribing declaration when a threshold value cannot be restored and maintained within a specified limit does not require immediate action simply because the current value is outside the limit, but does not permit extended operation outside the limit; the threshold must be considered reached as soon as it is apparent that operation within the limit cannot be attained.

Reference(s):

1. 05-S-01-EP-3 Containment Control
2. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.C

Barrier: Containment

Category: D. CTMT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: D. CTMT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

CNB8
Containment radiation (RITS-K648B or C) > 7,000 R/hr

Definition(s):

None

Basis:

In order to reach this Containment barrier Potential Loss threshold, a loss of the RCS barrier (Loss RCB6) and a loss of the Fuel Clad barrier (Loss FCB3) have already occurred. This threshold, therefore, represents a General Emergency classification.

The containment radiation monitor reading (7,350 R/hr rounded to 7,000 R/hr for readability) corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that 20% of the fuel cladding has failed (ref. 1). 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 emergency classification level to a General Emergency.

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

Reference(s):

1. XC-Q1D21-17001 Grand Gulf Nuclear Station (GGNS) Containment Radiation EAL Threshold Values
2. 04-1-01-D21-1 SOI Area Radiation Monitoring System
3. NEI 99-01 NEI 99-01 Primary Containment Radiation Potential Loss 4.A

Barrier: Containment

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

CNB9

UNISOLABLE direct downstream pathway to the environment exists after Containment isolation signal

Definition(s):

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

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

Basis:

Failure to isolate the leak, within 15 minutes or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway directly to the environment. The concern is the UNISOLABLE open pathway to the environment. A failure of the ability to isolate any one line indicates a breach of containment integrity.

This threshold also applies to a containment bypass due to a HPCS or LPCS line break outside containment with injection check valve failure allowing an UNISOLABLE direct pathway for RCS release to the environment.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems or minor release pathways, such as instrument lines, not protected by the Primary Containment Isolation System (PCIS). Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include UNISOLABLE main steam line or RCIC steam line breaks, UNISOLABLE RWCU system breaks, and UNISOLABLE containment atmosphere vent paths. If the main condenser is available with an UNISOLABLE main steam line, there may be releases through the steam jet air ejectors and gland seal exhausters. These pathways are monitored, however, and do not meet the intent of a nonisolable release path to the

environment. These minor releases are assessed using the Category A, Abnormal Rad Release / Rad Effluent, EALs.

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.

EP-3 Containment Control, may specify containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1). Under these conditions with a VALID containment isolation signal, the Containment barrier should be considered lost.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable containment leakage through various penetrations or system components. Minor releases may also occur if a containment isolation valve(s) fails to close but the containment atmosphere escapes to an enclosed system. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category A ICs.

There is no Containment barrier Potential Loss threshold associated with CTMT Integrity or Bypass.

Reference(s):

1. 05-S-01-EP-3 Containment Control
2. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.A

Barrier: Containment

Category: E. CTMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

CNB10

Intentional Containment venting per EPs

Definition(s):

None

Basis:

EP-3, Containment Control, may specify containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded. The threshold is met when the operator begins venting the containment in accordance with Attachment 13, not when actions are taken to bypass interlocks prior to opening the vent valves (ref. 1).

Intentional venting of containment for containment pressure or combustible gas control to the secondary containment and/or the environment is a Loss of the Containment. Venting for containment pressure control when not in an accident situation (e.g., to control pressure below the drywell high pressure scram setpoint) does not meet the threshold condition.

There is no Containment barrier Potential Loss threshold associated with CTMT Integrity or Bypass.

Reference(s):

1. 05-S-01-EP-3 Containment Control
2. NEI 99-01 CTMT Integrity or Bypass Containment Loss 3.B

Barrier: Containment

Category: E. CTMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

CNB11

Any condition in the opinion of the Emergency Director that indicates loss of the Containment Barrier

Definition(s):

None

Basis:

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

Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

Barrier: Containment

Category: E. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

CNB12

Any condition in the opinion of the Emergency Director that indicates potential loss of the Containment Barrier

Definition(s):

None

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.

Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Potential Loss 6.A

Category H – Hazards and Other Conditions Affecting Plant Safety

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

Hazards are non-plant, system-related events that can directly or indirectly affect plant operation, reactor plant safety or personnel safety.

1. Security

Unauthorized entry attempts into the PROTECTED AREA, bomb threats, sabotage attempts, and actual security compromises threatening loss of physical control of the plant.

2. Seismic Event

Natural events such as earthquakes have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety.

3. Natural or 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 can pose significant hazards to personnel and reactor safety. Appropriate for classification are FIRES within the plant PROTECTED AREA or which may affect operability of equipment needed for safe shutdown

5. Hazardous Gas

Toxic, corrosive, asphyxiant or flammable gas leaks can affect normal plant operations or preclude access to plant areas required to safely shutdown the plant.

6. Control Room Evacuation

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

7. Emergency Director 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.

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1 Unusual Event

A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by GGNS Security Shift Supervision

OR

Notification of a credible security threat directed at the site

OR

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

Mode Applicability:

All

Definition(s):

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 GGNS 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 GGNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the SECURITY OWNER CONTROLLED AREA).

~~*OWNER CONTROLLED AREA (OCA)* - For the purposes of classification, the Security area between the OCA detection fence and the PROTECTED AREA boundary known as the Security Owner Controlled Area (SOCA) in the GGNS Emergency Plan.~~

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

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

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

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

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

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

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

SECURITY OWNER CONTROLLED AREA - The SOCA is the area demarcated as a Vehicle Barrier System (VBS) consisting of passive elements including a series of large concrete blocks on the inside of a delay fence with early warning capabilities. The SOCA is the area between the SOCA Fence and the PROTECTED AREA Boundary.

Basis:

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 and HS1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program].*

The first threshold references the Security Shift Supervision because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.39 information.

The second threshold addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the Security Plan for GGNS.

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 11-S-82-1 Security Contingency Events (ref. 2).

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the Security Plan for GGNS (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

Reference(s):

1. GGNS Security Plan
2. 11-S-82-1 Security Contingency Events
3. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HA1.1 Alert

A HOSTILE ACTION is occurring or has occurred within the SECURITY OWNER CONTROLLED AREA as reported by GGNS Security Shift Supervision

OR

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

Mode Applicability:

All

Definition(s):

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 GGNS 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 GGNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the SECURITY 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.

~~*OWNER CONTROLLED AREA* - For the purposes of classification, the Security area between the OCA detection fence and the PROTECTED AREA boundary known as the Security Owner Controlled Area (SOCA) in the GGNS Emergency Plan.~~

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

SECURITY OWNER CONTROLLED AREA - The SOCA is the area demarcated as a Vehicle Barrier System (VBS) consisting of passive elements including a series of large concrete blocks on the inside of a delay fence with early warning capabilities. The SOCA is the area between the SOCA Fence and the PROTECTED AREA Boundary.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the SECURITY OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program]*.

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of Offsite Response Organizations (OROs), allowing them to be better prepared should it be necessary to consider further actions.

This EAL 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.

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

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 11-S-82-1 Security Contingency 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 SECURITY 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 for GGNS (ref. 1).

Escalation of the emergency classification level would be via IC HS1.

Reference(s):

1. GGNS Security Plan
2. 11-S-82-1 Security Contingency Events
3. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA

EAL:

HS1.1 Site Area Emergency

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

Mode Applicability:

All

Definition(s):

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 GGNS 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 GGNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the SECURITY 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.

~~*OWNER CONTROLLED AREA* - For the purposes of classification, the Security area between the OCA detection fence and the PROTECTED AREA boundary known as the Security Owner Controlled Area (SOCA) in the GGNS Emergency Plan.~~

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

SECURITY OWNER CONTROLLED AREA - The SOCA is the area demarcated as a Vehicle Barrier System (VBS) consisting of passive elements including a series of large concrete blocks on the inside of a delay fence with early warning capabilities. The SOCA is the area between the SOCA Fence and the PROTECTED AREA Boundary.

Basis:

This IC addresses the occurrence of a HOSTILE ACTION within the PROTECTED AREA. This event will require rapid response and assistance due to the possibility for damage to plant equipment.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1, 2).

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program].*

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Site Area Emergency declaration will mobilize Offsite Response Organization (ORO) resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

This EAL 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 Security Plan for GGNS (ref. 1).

Reference(s):

1. GGNS Security Plan
2. 11-S-82-1 Security Contingency Events
3. NEI 99-01 HS1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE levels

EAL:

HU2.1 Unusual Event

Seismic event > OBE as indicated by annunciation of **EITHER** of the following on SH13P856:

- Containment Operating Basis Earthquake (P856-1A-A3)
- Drywell Operating Basis Earthquake (P856-1A-A5)

Mode Applicability:

All

Definition(s):

None

Basis:

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE). An earthquake greater than an OBE but less than a Safe Shutdown Earthquake (SSE) should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., perform walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event. The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the U.S. Geological Survey (USGS), check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

To avoid inappropriate emergency classification resulting from spurious actuation of the seismic instrumentation or felt motion not attributable to seismic activity, an offsite agency (USGS, National Earthquake Information Center (NEIC)) can confirm that an earthquake has occurred in the area of the plant. Such confirmation should not, however, preclude a timely emergency declaration based on receipt of the OBE alarm. If requested, provide the analyst

with the following GGNS coordinates: **32° 0' 27" north latitude, 91° 2' 53" west longitude** (ref. 2). Alternatively, near real-time seismic activity can be accessed via the NEIC website.

Reference(s):

1. 05-S-02-VI-3 Earthquake
2. UFSAR 2.1.1 Site Location and Description
3. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.1	**Unusual Event**
A tornado strike within the PROTECTED AREA	

Mode Applicability:

All

Definition(s):

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

Basis:

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

This EAL addresses a tornado striking (touching down) within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories A, F, S or C.

If damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under EAL CA6.1 or SA8.1.

A tornado striking (touching down) within the PROTECTED AREA warrants declaration of an Unusual Event regardless of the measured wind speed at the meteorological tower. A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

Reference(s):

1. 05-1-02-VI-2 Hurricanes, Tornados and Severe Weather
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.2 Unusual Event

Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component required by Technical Specifications for the current operating mode

Mode Applicability:

All

Definition(s):

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

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

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

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

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

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

Basis:

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

This EAL addresses FLOODING of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

Escalation of the emergency classification level would be based on ICs in Recognition Categories A, F, S or C.

Refer to EAL CA6.1 or SA8.1 for internal FLOODING affecting more than one SAFETY SYSTEM train.

Reference(s):

1. 05-1-02-VI-1 Flooding
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.3 Unusual Event

Movement of personnel within the PROTECTED AREA is IMPEDED due to an event external to the PROTECTED AREA involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

Basis:

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

This EAL addresses a hazardous materials event originating at a location outside the PROTECTED AREA and of sufficient magnitude to IMPEDE the movement of personnel within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories A, F, S or C.

Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.4 Unusual Event

A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)

Note 7: This EAL does **not** apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Mode Applicability:

All

Definition(s):

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.

Basis:

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

This EAL addresses a hazardous event that causes an on-site impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site FLOODING caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an on-site train derailment blocking the access road.

This EAL is not intended to apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the FLOODING around the Cooper Station during the Midwest floods of 1993, or the FLOODING around Ft. Calhoun Station in 2011.

Escalation of the emergency classification level would be based on ICs in Recognition Categories A, F, S or C.

Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.1 Unusual Event

A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):

- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

The FIRE is located within **any** Table H-1 area

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table H-1 Fire Areas
• Unit 1 Containment • Unit 1 Auxiliary Building • Unit 1 Turbine Building • Control Building • Diesel Generator Rooms • SSW Pump & Valve Rooms

Mode Applicability:

All

Definition(s):

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

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

Basis:

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.

The intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

The 15 minute requirement begins with a credible notification that a FIRE is occurring, or receipt of multiple VALID fire detection system alarms or field validation of a single fire alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field.

Table H-1 Fire Areas are those areas that contain equipment necessary for safe operation and shutdown of the plant (ref. 1, 2).

Reference(s):

1. 05-S-02-V-1 Response to Fires
2. 10-S-03-2 Response to Fires
3. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.2 Unusual Event

Receipt of a single fire alarm (i.e., **no** other indications of a FIRE) (Note 11)

AND

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

AND

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Note 11: During Modes 1 and 2, HU4.2 is not applicable to a single fire alarm in the containment or drywell.

Table H-1 Fire Areas
• Unit 1 Containment • Unit 1 Auxiliary Building • Unit 1 Turbine Building • Control Building • Diesel Generator Rooms • SSW Pump & Valve Rooms

Mode Applicability:

All

Definition(s):

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

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

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

This EAL is not applicable for the containment or drywell in Modes 1 and 2. The air flow design and TS requirements for operation of Containment Fan Coolers and the drywell cooling system are such that multiple detectors would be expected to alarm for a fire in the containment or drywell. A fire in the containment or drywell in these modes would therefore be classified under EAL HU4.1.

If an actual FIRE is verified by a report from the field, then HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

Basis-Related Requirements from Appendix R

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."
>
> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.
>
> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in HU4.2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

The 30 minute requirement begins upon receipt of a single VALID fire detection system alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field. Actual field reports must be made within the 30 minute time limit or a classification must be made. If a fire is verified to be occurring by field report, classification shall be made based on EAL HU4.1, with the 15 minute requirement beginning with the verification of the fire by field report.

Table H-1 Fire Areas are those areas that contain equipment necessary for safe operation and shutdown of the plant (ref. 1, 2).

Reference(s):

1. 05-S-02-V-1 Response to Fires
2. 10-S-03-2 Response to Fires
3. UFSAR Appendix 9A Fire Hazard Analysis Report
4. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.3 Unusual Event

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

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

In addition to a FIRE addressed by EAL HU4.1 or HU4.2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.4 Unusual Event

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

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

If a FIRE within the plant PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via EAL CA6.1 or SA8.1.

Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gas

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

EAL:

HA5.1 Alert

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

AND

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

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

Table H-2 Safe Operation & Shutdown Rooms/Areas

Room/Area	Mode
Control Building 111' SWGR Rms (0C202, 0C215)	3
Auxiliary Building 93' RHR A Pump Room (1A103)	3
Auxiliary Building 93' RHR B Pump Room (1A105)	3
Auxiliary Building 93' Corridor (1A101)	3
Auxiliary Building 119' Corridor (1A201)	3
Auxiliary Building 119' RHR A Pump Room (1A203)	3
Auxiliary Building 119' RHR B Pump Room (1A205)	3
Auxiliary Building 119' RCIC Room (1A204)	3
Auxiliary Building 139' RHR A Room (1A303, 1A304)	3
Auxiliary Building 139' RHR B Room (1A306, 1A307)	3
Radwaste Building 118' Radwaste Control Room (0R241)	3

Mode Applicability:

3 – Hot Shutdown

Definition(s):

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

Basis:

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 3.
- 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 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 generate smoke and that automatically or manually activate a fire suppression system in an area.

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

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

EAL HA5.1 mode applicability has been limited to the mode limitations of Table H-2 (Mode 3 **only**).

Reference(s):

1. Attachment 2 Safe Operation & Shutdown Areas Tables A-3 & H-2 Bases
2. NEI 99-01 HA5

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

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

EAL:

HA6.1 Alert

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel

Mode Applicability:

All

Definition(s):

None

Basis:

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

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Transfer of plant control begins when the last licensed operator leaves the Control Room.

Escalation of the emergency classification level would be via IC HS6.

Reference(s):

1. 05-1-02-II-1 Shutdown from the Remote Shutdown Panel
2. NEI 99-01 HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

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

EAL:

HS6.1 Site Area Emergency

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel

AND

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

- Reactivity (Modes 1 and 2 **only**)
- RPV water level
- RCS heat removal

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refueling

Definition(s):

None

Basis:

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on Emergency 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 safe shutdown location(s).

Transfer of plant control and the time period to establish control begins when the last licensed operator leaves the Control Room.

Escalation of the emergency classification level would be via IC FG1 or CG1

Reference(s):

1. 05-1-02-II-1 Shutdown from the Remote Shutdown Panel
2. EP FAQ 2015-014
3. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

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

EAL:

HU7.1 Unusual Event

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.

Mode Applicability:

All

Definition(s):

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

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

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

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

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

Basis:

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 emergency classification level description for an UNUSUAL EVENT.

Reference(s):

1. NEI 99-01 HU7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

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

EAL:

HA7.1 Alert

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.

Mode Applicability:

All

Definition(s):

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 GGNS 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 GGNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the SECURITY OWNER CONTROLLED AREA).

~~*OWNER CONTROLLED AREA* - For the purposes of classification, the Security area between the OCA detection fence and the PROTECTED AREA boundary known as the Security Owner Controlled Area (SOCA) in the GGNS Emergency Plan.~~

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

SECURITY OWNER CONTROLLED AREA - The SOCA is the area demarcated as a Vehicle Barrier System (VBS) consisting of passive elements including a series of large concrete blocks on the inside of a delay fence with early warning capabilities. The SOCA is the area between the SOCA Fence and the PROTECTED AREA Boundary.

Basis:

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 emergency classification level description for an ALERT.

Reference(s):

1. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist that in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY

EAL:

HS7.1 Site Area Emergency

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

Mode Applicability:

All

Definition(s):

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 GGNS 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 GGNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the SECURITY OWNER CONTROLLED AREA).

~~*OWNER CONTROLLED AREA* - For the purposes of classification, the Security area between the OCA detection fence and the PROTECTED AREA boundary known as the Security Owner Controlled Area (SOCA) in the GGNS Emergency Plan.~~

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

SECURITY OWNER CONTROLLED AREA - The SOCA is the area demarcated as a Vehicle Barrier System (VBS) consisting of passive elements including a series of large concrete blocks on the inside of a delay fence with early warning capabilities. The SOCA is the area between the SOCA Fence and the PROTECTED AREA Boundary.

SITE BOUNDARY - That boundary defined by a 696 meter (.43 miles) radius from the center of the reactor.

Basis:

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 emergency classification level description for a SITE AREA EMERGENCY.

Reference(s):

1. NEI 99-01 HS7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist that in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY

EAL:

HG7.1	**General Emergency**
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	

Mode Applicability:

All

Definition(s):

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 GGNS 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 GGNS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the SECURITY 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.

~~*OWNER CONTROLLED AREA* - For the purposes of classification, the Security area between the OCA detection fence and the PROTECTED AREA boundary known as the Security Owner Controlled Area (SOCA) in the GGNS Emergency Plan.~~

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

PROTECTED AREA - An area encompassed by physical barriers (i.e., the security fence) and to which access is controlled.

SECURITY OWNER CONTROLLED AREA - The SOCA is the area demarcated as a Vehicle Barrier System (VBS) consisting of passive elements including a series of large concrete blocks on the inside of a delay fence with early warning capabilities. The SOCA is the area between the SOCA Fence and the PROTECTED AREA Boundary.

Basis:

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 emergency classification level description for a GENERAL EMERGENCY.

Reference(s):

1. NEI 99-01 HG7

Category S – System Malfunction

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

Numerous system-related equipment failure events that warrant emergency classification have been identified in this category. They may pose actual or potential threats to plant safety.

The events of this category pertain to the following subcategories:

1. Loss of ESF AC 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 onsite and offsite power sources for 4.16 KV ESF buses.

2. Loss of Vital DC Power

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

3. Loss of Control Room Indications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant rise from these base-line levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

5. RCS Leakage

The reactor pressure vessel provides a volume for the coolant that covers the reactor core. The reactor pressure vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail. Excessive RCS leakage greater than Technical Specification limits indicates potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and containment integrity.

6. RPS Failure

This subcategory includes events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor scrams. In the plant licensing basis, postulated failures of the RPS to complete a reactor scram comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean any scram failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and containment integrity.

7. Loss of Communications

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

8. Hazardous Event Affecting Safety Systems

Various natural and technological events that result in degraded plant SAFETY SYSTEM performance or significant VISIBLE DAMAGE warrant emergency classification under this subcategory.

Category: S – System Malfunction

Subcategory: 1 – Loss of ESF AC Power

Initiating Condition: Loss of **all** offsite AC power capability to ESF buses for 15 minutes or longer

EAL:

SU1.1 Unusual Event

Loss of **all** offsite AC power capability, Table S-1, to DIV I and DIV II ESF 4.16 KV buses for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-1 AC Power Sources
Offsite • ESF Transformer 11 • ESF Transformer 12 • ESF Transformer 21 **Onsite** • DIV I DG (DG 11) • DIV II DG (DG 12)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

The HPCS bus (DIV III) is not credited because it only supplies power to the HPCS pump and associated loads, not any long term decay heat removal systems. In particular, suppression pool cooling mechanisms would be essential subsequent to a station blackout.

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC ESF buses. This condition represents a potential reduction in the level of safety of the plant.

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the ESF buses, whether or not the buses are powered from it.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of offsite power.

Escalation of the emergency classification level would be via IC SA1.

Reference(s):

1. UFSAR Figure 8.1-001 Main One Line Diagram
2. UFSAR section 8.1 Electric Power Introduction
3. UFSAR section 8.3 Onsite Power
4. UFSAR section 8A Loss of all AC Power
5. 05-1-02-I-4 Loss of AC Power
6. NEI 99-01 SU1

Category: S – System Malfunction

Subcategory: 1 – Loss of ESF AC Power

Initiating Condition: Loss of **all but one** AC power source to ESF buses for 15 minutes or longer

EAL:

SA1.1 Alert

AC power capability, Table S-1, to DIV I and DIV II ESF 4.16 KV buses reduced to a single power source for ≥ 15 min. (Note 1)

AND

Any additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-1 AC Power Sources
Offsite • ESF Transformer 11 • ESF Transformer 12 • ESF Transformer 21 **Onsite** • DIV I DG (DG 11) • DIV II DG (DG 12)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

The HPCS bus (DIV III) is not credited because it only supplies power to the HPCS pump and associated loads, not any long term decay heat removal systems. In particular, suppression pool cooling mechanisms would be essential subsequent to a station blackout.

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

An "AC power source" is a source recognized in AOPs and EOPs, and capable of supplying required power to an ESF bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- A loss of all offsite power and loss of all ESF emergency power sources (e.g., onsite diesel generators) with a single train of ESF buses being back-fed from the unit main generator.
- A loss of ESF emergency power sources (e.g., onsite diesel generators) with a single train of ESF emergency buses being fed from an offsite power source.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

Escalation of the emergency classification level would be via IC SS1.

This EAL is the hot condition equivalent of the cold condition EAL CU2.1.

Reference(s):

1. UFSAR Figure 8.1-001 Main One Line Diagram
2. UFSAR section 8.1 Electric Power Introduction
3. UFSAR section 8.3 Onsite Power
4. UFSAR section 8A Loss of all AC Power
5. 05-1-02-I-4 Loss of AC Power
6. NEI 99-01 SA1

Category: S – System Malfunction

Subcategory: 1 – Loss of ESF AC Power

Initiating Condition: Loss of **all** offsite power and **all** onsite AC power to ESF buses for 15 minutes or longer

EAL:

SS1.1 Site Area Emergency

Loss of **all** offsite and **all** onsite AC power to DIV I and DIV II ESF 4.16 KV buses for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink Mitigative strategies using other power sources (HPCS DIV III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout.. 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.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via IC AG1, FG1 or SG1.

This EAL is the hot condition equivalent of the cold condition EAL CA2.1.

Reference(s):

1. UFSAR Figure 8.1-001 Main One Line Diagram
2. UFSAR section 8.1 Electric Power Introduction
3. UFSAR section 8.3 Onsite Power
4. UFSAR section 8A Loss of all AC Power
5. 05-1-02-I-4 Loss of AC Power
6. NEI 99-01 SS1

Category: S –System Malfunction

Subcategory: 1 – Loss of ESF AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to ESF buses

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power to DIV I and DIV II ESF 4.16 KV buses

AND EITHER:

- Restoration of at least one ESF 4.16 KV bus in < 4 hours is **not** likely (Note 1)
- RPV water level **cannot** be restored and maintained > -191 in.

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

Indication of continuing core cooling degradation is manifested by the inability to restore and maintain RPV water level above the Minimum Steam Cooling Reactor Water Level (-191 in.) (ref. 6). Core submergence is the most desirable means of core cooling, however when RPV level is below TAF, the uncovered portion of the core can be cooled by less reliable means (i.e., steam cooling or spray cooling).

This IC addresses a prolonged loss of all power sources to AC ESF emergency buses. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat

removal/pressure control, spent fuel heat removal and the ultimate heat sink. Mitigative strategies using other power sources (HPCS DIV III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout. A prolonged loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

The EAL should require declaration of a General Emergency prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC ESF emergency bus by the end of the analyzed station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is a greater likelihood of challenges to multiple fission product barriers. 4 hours is the site-specific SBO coping analysis time (ref. 4).

The estimate for restoring at least one ESF emergency bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, protective actions for the public.

The EAL will also require a General Emergency declaration if the loss of AC power results in parameters that indicate an inability to adequately remove decay heat from the core.

Reference(s):

1. UFSAR Figure 8.1-001 Main One Line Diagram
2. UFSAR section 8.1 Electric Power Introduction
3. UFSAR section 8.3 Onsite Power
4. UFSAR section 8A Loss of all AC Power
5. 05-1-02-I-4 Loss of AC Power
6. 02-S-01-40 EP Technical Bases
7. NEI 99-01 SG1

Category: S – System Malfunction

Subcategory: 1 – Loss of ESF AC Power

Initiating Condition: Loss of **all** ESF AC and vital DC power sources for 15 minutes or longer

EAL:

SG1.2 General Emergency

Loss of **all** offsite and **all** onsite AC power to DIV I and DIV II ESF 4.16 KV buses for ≥ 15 min. (Note 1)

AND

Indicated voltage is < 105 VDC on vital 125 VDC buses 11DA and 11DB for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

Vital DC buses 11DA and 111DB feed the Division 1 and Division 2 loads respectively. The Division 1 and Division 2 batteries each have 61 cells with a design minimum of 1.72 volts/cell. These cell voltages yield minimum design bus voltages of 104.92 VDC (rounded to 105 VDC) (ref. 6, 7).

This IC addresses a concurrent and prolonged loss of both emergency ESF AC and Vital DC power. A loss of all emergency ESF AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling,

containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. Mitigative strategies using other power sources (HPCS DIV III diesel generator, FLEX generators, etc.) may be effective in supplying power to these buses. These power sources must be controlled in accordance
with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines) and must be capable (alone or in combination) of supplying power for long term decay heat removal systems. In particular, suppression pool cooling systems would be essential subsequent to a station blackout. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both emergency ESF AC and Vital DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

Reference(s):

1. UFSAR Figure 8.1-001 Main One Line Diagram
2. UFSAR section 8.1 Electric Power Introduction
3. UFSAR section 8.3 Onsite Power
4. UFSAR section 8A Loss of all AC Power
5. 05-1-02-I-4 Loss of AC Power
6. Calculation No: EC-Q1111-14001 Station Division I Battery 1A3 and Division II Battery 1B3 Discharge Capacity during Extended Loss of AC Power
7. UFSAR 8.3.2.1.1 Station DC Power
8. NEI 99-01 SG8

Category: S – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

Initiating Condition: Loss of **all** vital DC power for 15 minutes or longer

EAL:

SS2.1 Site Area Emergency

Indicated voltage is < 105 VDC on vital 125 VDC buses 11DA and 11DB for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

Vital DC buses 11DA and 11DB feed the Division 1 and Division 2 loads respectively. The Division 1 and Division 2 batteries each have 61 cells with a design minimum of 1.72 volts/cell. These cell voltages yield minimum design bus voltages of 104.92 VDC (rounded to 105 VDC) (ref. 1, 2).

This IC addresses a loss of vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Escalation of the emergency classification level would be via IC AG1, FG1 or SG1.

This EAL is the hot condition equivalent of the cold condition EAL CU4.1.

Reference(s):

1. Calculation No: EC-Q1111-14001 Station Division I Battery 1A3 and Division II Battery 1B3 Discharge Capacity during Extended Loss of AC Power
2. UFSAR 8.3.2.1.1 Station DC Power
3. NEI 99-01 SS8

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer

EAL:

SU3.1 Unusual Event

An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-2 Safety System Parameters
• Reactor power • RPV water level • RPV pressure • Containment pressure • Suppression Pool water level • Suppression Pool temperature

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

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:

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital or recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, RPV water level and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RPV water level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via EAL SA3.1.

Reference(s):

1. UFSAR 7.5 Safety-Related Display Instrumentation
2. NEI 99-01 SU2

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress

EAL:

SA3.1 Alert

An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

Any significant transient is in progress, Table S-3

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Table S-2 Safety System Parameters
• Reactor power • RPV water level • RPV pressure • Containment pressure • Suppression Pool water level • Suppression Pool temperature

Table S-3 Significant Transients
• Reactor scram • UNPLANNED drop in reactor thermal power > 25% • Electrical load rejection > 25% electrical load • ECCS injection • Thermal power oscillations > 10%

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

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:

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital or recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, RPV water level and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RPV water level cannot be determined from the indications and recorders on a main control board,

the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of indication.

Escalation of the emergency classification level would be via IC FS1 or AS1.

Reference(s):

1. UFSAR 7.3 Engineered Safety Features Systems
2. NEI 99-01 SA2

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: RCS activity greater than Technical Specification allowable limits

EAL:

SU4.1 Unusual Event
Offgas Pretreatment radiation monitor high-high alarm

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

The Offgas Pretreatment monitors radioactivity in the Offgas system downstream of the Offgas condenser. The monitor detects the radiation level that is attributable to the fission gases produced in the reactor and transported with steam through the turbine to the condenser. The Hi-Hi alarm, if alarming, indicates that the radioactivity present at the recombiner effluent discharge is at or above the Technical Specification 3.7.5 limit of 380 millicuries per second of Noble Gases. (ref. 1)

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via IC FA1 or the Recognition Category A ICs.

I~~O~~n the event that the Offgas Pretreatment Radiation Monitor High-High Alarm is out of service, the use of offgas~~s~~ flowrates and Offgas Pretreatment Radiation monitor readings is a viable ~~contigiency~~ contingency action to classify the EAL~~IC~~. See chart in 04-1-02-1H13-P601-19A-D7 Alarm Response Instruction for OG PRE-TREAT RAD HI_HI alarm.

Reference(s):

1. Alarm Response Instruction 04-1-02-1H13-P601-19A-D7
2. UFSAR 11.5 Process and Effluent Radiological Monitoring and Sampling Systems
3. Technical Specification 3.7.5 Main Condenser Offgas
4. 05-1-02-II-2 Offgas Activity High
5. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.2 Unusual Event

Coolant activity > 0.2 µCi/gm dose equivalent I-131 for > 48 hours

OR

Coolant activity > 4.0 µCi/gm dose equivalent I-131 instantaneous

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via IC FA1 or the Recognition Category A ICs.

Reference(s):

1. Technical Specification B3.4.8, RCS Specific Activity bases
2. UFSAR Section 15.6.4 Steam System Piping Break Outside Containment
3. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1 Unusual Event

RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. (Note 1)

OR

RCS identified leakage > 25 gpm for ≥ 15 min. (Note 1)

OR

Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min. (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed an additional 15 minutes to declare after the time limit is exceeded.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

UNISOLABLE - An open or breached system line that **cannot** be isolated, remotely or locally.

Basis:

Failure to isolate the leak within 15 minutes, or if known that the leak cannot be isolated within 15 minutes, from the start of the leak requires immediate classification.

Identified leakage is leakage into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a collecting sump; or leakage into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage.

Unidentified leakage is all leakage into the drywell that is not identified leakage (ref. 2, 3).

Pressure boundary leakage is leakage through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall (ref. 2, 3).

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

The first and second EAL conditions are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications). The third condition addresses an RCS

mass loss caused by an UNISOLABLE leak through an interfacing system. These conditions thus apply to leakage into the containment, or a location outside of containment.

The leak rate values for each condition were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). The first condition uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. A stuck-open Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.

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 A or F.

Reference(s):

1. UFSAR Section 5.2.5, Detection of Leakage Through Reactor Coolant Pressure Boundary
2. Technical Specification Definitions Section 1.1
3. Technical Specification 3.4.5
2. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor

EAL:

SU6.1 Unusual Event

An automatic scram did **not** shut down the reactor as indicated by reactor power > 5% after **any** RPS setpoint is exceeded

AND

A subsequent automatic scram or manual scram action taken at the reactor control console (Mode Switch, Manual PBs, ARI/RPT) is successful in shutting down the reactor as indicated by reactor power ≤ 5% (APRM downscale) (Note 8)

Note 8: A manual scram action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation, 2 - Startup

Definition(s):

IMMINENT - The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram 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.

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function. A reactor scram is automatically initiated by the RPS when certain continuously monitored parameters exceed predetermined setpoints (ref. 1).

A successful scram has occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power to or below the APRM downscale setpoint of 5%.

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., Mode Switch, manual scram pushbuttons, or ARI/RPT initiation). Reactor shutdown achieved by use of alternate control rod insertion methods (i.e., EP-2A step Q-1) does not constitute a successful manual scram (ref. 2).

Following any automatic RPS scram signal, operating procedures (e.g., EP-2) prescribe insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved. Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Unusual Event (ref. 3).

Taking the Mode Switch to Shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

For the purposes of this EAL, a successful automatic initiation of ARI/RPT that reduces reactor power to ≤ 5% is not considered a successful automatic scram. If automatic initiation of ARI/RPT has occurred and caused reactor shutdown, the automatic RPS scram must have failed. ARI/RPT is a backup means of inserting control rods in the unlikely event that an automatic RPS scram signal exists but the reactor continues to generate significant power. However, a successful automatic or manual initiation of ARI/RPT is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

In the event that the operator identifies a reactor scram is IMMINENT and initiates a successful manual reactor scram before the automatic scram setpoint is reached, no declaration is required. The successful manual scram of the reactor before it reaches its automatic scram setpoint or reactor scram signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss.

If by procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

Following the failure of an automatic reactor scram, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram using a different switch). Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles". Taking the Reactor Mode Switch to Shutdown is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram 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 EAL SA6.1. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

Should a reactor scram signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal generated as a result of plant work causes a plant transient that results in a condition that should have included an automatic reactor scram and the RPS fails to automatically shutdown the reactor, then this IC and associated EALs are applicable, and should be evaluated.
- If the signal generated as a result of plant work does not cause a plant transient and the scram failure is determined through other means (e.g., assessment of test results), then this IC and associated EALs are not applicable and no classification is warranted.

Reference(s):

1. Technical Specification Table 3.3.1.1-1 Reactor Protection System Instrumentation
2. 05-S-01-EP-2A ATWS RPV Control
3. 05-S-01-EP-2 RPV Control
4. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor

EAL:

SU6.2 Unusual Event

A manual scram did **not** shut down the reactor as indicated by reactor power > 5% after **any** manual scram action was initiated

AND

A subsequent automatic scram or manual scram action taken at the reactor control console (Mode Switch, Manual PBs, ARI/RPT) is successful in shutting down the reactor as indicated by reactor power ≤ 5% (APRM downscale) (Note 8)

Note 8: A manual scram action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation, 2 - Startup

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram 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.

This EAL addresses a failure of a manually initiated scram in the absence of having exceeded an automatic RPS trip setpoint and a subsequent automatic or manual scram is successful in shutting down the reactor (reactor power ≤ 5%) (ref. 1).

A successful scram has occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power to or below the APRM downscale setpoint of 5%.

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., Mode Switch, manual scram pushbuttons, or ARI/RPT initiation). Reactor shutdown achieved by use of alternate control rod insertion methods (i.e., EP-2A step Q-1) does not constitute a successful manual scram (ref. 2).

Taking the Mode Switch to Shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

Successful automatic or manual initiation of ARI/RPT is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

If both subsequent automatic and subsequent manual reactor scram actions in the Control Room fail to reduce reactor power below the power associated with the SAFETY SYSTEM design (≤ 5%) following a failure of an initial manual scram, the event escalates to an Alert under EAL SA6.1.

Following the failure of an automatic reactor scram, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram) using a different switch. Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

Taking the Reactor Mode Switch to Shutdown is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles

are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

Should a reactor scram signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal generated as a result of plant work causes a plant transient that results in a condition that should have included an automatic reactor scram and the RPS fails to automatically shutdown the reactor, then this IC and associated EALs are applicable, and should be evaluated.
- If the signal generated as a result of plant work does not cause a plant transient and the scram failure is determined through other means (e.g., assessment of test results), then this IC and associated EALs are not applicable and no classification is warranted.

Reference(s):

1. Technical Specification Table 3.3.1.1-1 Reactor Protection System Instrumentation
2. 05-S-01-EP-SA ATWS RPV Control
3. 05-S-01-EP-2 RPV Control
4. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are **not** successful in shutting down the reactor

EAL:

SA6.1 Alert

An automatic or manual scram fails to shut down the reactor as indicated by reactor power > 5%

AND

Manual scram actions taken at the reactor control console (Mode Switch, Manual PBs, ARI/RPT) are **not** successful in shutting down the reactor as indicated by reactor power > 5% (Note 8)

Note 8: A manual scram action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation, 2 - Startup

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action

taken away from the reactor control consoles since this event entails a significant failure of the RPS.

This EAL addresses any automatic or manual reactor scram signal that fails to shut down the reactor followed by subsequent manual scram actions that fail 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%).

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., Mode Switch, manual scram pushbuttons, or ARI/RPT initiation). Reactor shutdown achieved by use of alternate control rod insertion methods (i.e., EP-2A step Q-1) does not constitute a successful manual scram (ref. 2).

For the purposes of this EAL, a successful automatic initiation of ARI/RPT that reduces reactor power to or below 5% is not considered a successful automatic scram. If automatic actuation of ARI/RPT has occurred and caused reactor shutdown, the automatic RPS scram must have failed. ARI/RPT is a backup means of inserting control rods in the unlikely event that an automatic RPS scram signal exists but the reactor continues to generate significant power. However, a successful automatic initiation of ARI/RPT is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control consoles (e.g., locally opening breakers). Actions taken at back panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

Taking the Reactor Mode Switch to Shutdown is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram) 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 RPV water level or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6 or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

Reference(s):

1. Technical Specification Table 3.3.1.1-1 Reactor Protection System Instrumentation
2. 05-S-01-EP-2A ATWS RPV Control
3. 05-S-01-EP-2 RPV Control
4. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to RPV water level or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual scram fails to shut down the reactor as indicated by reactor power > 5%

AND

All actions to shut down the reactor are **not** successful as indicated by reactor power > 5%

AND EITHER:

RPV water level **cannot** be restored and maintained > -191 in.

OR

Heat Capacity Temperature Limit (HCTL) exceeded (EP Figure 1)

Mode Applicability:

1 - Power Operation, 2 - Startup

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition;

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Basis:

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

This EAL addresses the following:

- Any automatic reactor scram signal followed by subsequent manual scram actions 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 (EAL SA6.1), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

Reactor shutdown achieved by use of control rod insertion methods in EP-2A step Q-1 are also credited as a successful shutdown provided reactor power can be reduced to or below the APRM downscale trip setpoint before indications of an extreme challenge to either core cooling or heat removal exist. (ref. 1)

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS barriers.

Indication that core cooling is extremely challenged is manifested by inability to restore and maintain RPV water level above the Minimum Steam Cooling RPV Water Level (MSCRWL) (ref. 1). The MSCRWL is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500°F. This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core spray cooling could be a precursor of a core melt sequence (ref 2).

The Heat Capacity Temperature Limit (HCTL, EP Figure 1) is the highest suppression pool water temperature from which Emergency RPV Depressurization will not raise suppression pool temperature above the maximum design suppression pool temperature.

The HCTL is a function of RPV pressure and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant. This threshold is met when the final step of section SPT in EP-3, Containment Control, is reached (ref. 3). This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature.

In some instances, the emergency classification resulting from this 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 EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

Escalation of the emergency classification level would be via IC AG1 or FG1.

Reference(s):

1. 05-S-01-EP-2A, ATWS RPV Control
2. 05-S-01-EP-5, RPV Flooding
3. 05-S-01-EP-3, Containment Control
4. NEI 99-01 SS5

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

SU7.1 Unusual Event

Loss of **all** Table S-4 onsite communication methods

OR

Loss of **all** Table S-4 State and local agency communication methods

OR

Loss of **all** Table S-4 NRC communication methods

Table S-4 Communication Methods

System	Onsite	State/ Local	NRC
Station Radio System	X		
GGNS Plant Phone System	X		
Public Address System	X		
Emergency Notification System (ENS)			X
Commercial Telephone System		X	X
Satellite Phones		X	X
~~INFORM Notification System~~		~~X~~	
Operational Hotline		X	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

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 State and local agencies and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of on-site information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all State and local agencies of an emergency declaration. The State and local agencies referred to here are the Mississippi Emergency Management Agency, Claiborne County Civil Defense, Mississippi Highway Safety Patrol, Claiborne County Sheriff's Department, Louisiana Department of Environmental Quality, Tensas Parish Sheriff's Office, and the Louisiana Governor's Office of Homeland Security and Emergency Preparedness.

The third EAL condition addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

Reference(s):

1. GGNS Emergency Plan Section 7.5, Communications Systems
2. 04-S-01-R61-1 Plant Communications
3. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 8 – Hazardous Event Affecting Safety Systems

Initiating Condition: Hazardous event affecting SAFETY SYSTEMS needed for the current operating mode

EAL:

SA8.1 Alert

The occurrence of **any** Table S-5 hazardous event

AND

Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode

AND EITHER:

- Event damage has caused indications of degraded performance to the second train of the SAFETY SYSTEM needed for the current operating mode
- Event damage has resulted in VISIBLE DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode

(Notes 9, 10)

Note 9: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then emergency classification is **not** warranted.

Note 10: If the hazardous event **only** resulted in VISIBLE DAMAGE, with **no** indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is **not** warranted.

Table S-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should **not** automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

FIRE - Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do **not** constitute fires. Observation of flame is preferred but is **not** required if large quantities of smoke and heat are observed.

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10CFR50.2):

> Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:
>
> (1) The integrity of the reactor coolant pressure boundary;
>
> (2) The capability to shut down the reactor and maintain it in a safe shutdown condition;
>
> (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

VISIBLE DAMAGE - Damage to a SAFETY SYSTEM train 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 SAFETY SYSTEM train.

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues.

Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

Indications of degraded performance addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

VISIBLE DAMAGE addresses damage to a SAFETY SYSTEM train that is not in service/operation and that potentially could cause performance issues. Operators will make a determination of VISIBLE DAMAGE 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. This VISIBLE DAMAGE should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

Escalation of the emergency classification level would be via IC FS1 or AS1.

This EAL is the hot condition equivalent of the cold condition EAL CA6.1.

Reference(s):

1. EP FAQ 2016-002
2. NEI 99-01 SA9

Background

NEI 99-01 Revision 6 ICs AA3 and HA5 prescribe declaration of an Alert based on impeded access to rooms or areas (due to either area radiation levels or hazardous gas concentrations) where equipment necessary for normal plant operations, cooldown or shutdown is located. These areas are intended to be plant operating mode dependent. Specifically the Developers Notes for AA3 and HA5 states:

> *The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.*
>
> *The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).*

Further, as specified in IC HA5:

> *The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.*

GGNS Table A-3 and H-2 Bases

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:

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
IOI 03-1-01-2 Power Operations			
LOWER Power by reducing Recirculation flow until 62.2% core flow (70 mlbm/hr) is reached.	MCR	1	
INSERT Control Rods per Control Rod Movement Sequence.	MCR	1	
TECH SPEC TRIGGER (SR 3.3.2.1.2, SR 3.3.2.1.4) IF Reactor power has been reduced below the HPSP OR the LPSP, THEN PERFORM one of the following: Required Surveillances or enter LCO for TS 3.3.2.1	MCR	1	
CHECK OPEN the following valves on 1H13-P870-6C: a. N11-F029A, HP TURB EXTR To MSR A 1ST STG RHT b. N11-F029B, HP TURB EXTR To MSR B 1ST STG RHTIF N11-F029A OR N11-F029B are NOT open, THEN RETURN MSR 1ST Stage Reheaters to service per SOI 04-1-01-N11-1.	MCR	1	
CHECK OPEN the following valves on panel 1H13-P870-6C: a. N36-F010A, EXTR STM SPLY TO FW HTR 5A b. N36-F010B, EXTR STM SPLY TO FW HTR 5B c. N36-F011A, EXTR STM SPLY TO FW HTR 6A d. N36-F011B, EXTR STM SPLY TO FW HTR 6B TAKE handswitches for the following valves to OPEN position on panel 1H13-P870-6C: a. N36-F013A, FW HTR 5A EXTR STM BTV b. N36-F013B, FW HTR 5B EXTR STM BTV c. N36-F012A, FW HTR 6A EXTR STM BTV d. N36-F012B, FW HTR 6B EXTR STM BTV	MCR	1	
NOTIFY the following of the power reduction: • Load Dispatcher (Woodlands) • *Duty Manager (IF unexpected power reduction) • (SMEPA)(1-601-261-2318 OR 1-601-261-2313) • * (SMEPA) Site Representative • Radwaste • Radiation Protection • Chemistry • *NRC Resident Inspector	MCR	1	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
*These notifications Must be made by Shift Manager			
IF LP Turbine inlet temperature is >491°F, and N11-F028A and N11-F028B are open, THEN SIMULTANEOUSLY THROTTLE the following valves on 1H22-P177 to CONTROL LP Inlet Temperatures within a band of 470° F to 490° F while monitoring LP Turbine Inlet differential temperatures within 30° F (comparing A side to B side). • N11-F028A • N11-F028B •I IF LP Turbine inlet temperature is >491°F, and N11-F028A and N11-F028B are closed, THEN SLOWLY, SIMULTANEOUSLY LOWER MSR-A/B HTG STM FEED CONT manual setpoint to CONTROL LP Inlet Temperatures within a band of 470° F to 490° F while monitoring LP Turbine Inlet differential temperatures within 30° F (comparing A side to B side).	TURB BLDG ELEV 133 AREA 4 ROOM 1T325	1	Not Required
LOWER Reactor power by INSERTING control rods to specified Control Rod in-sequence position per 17-S-02-400.	MCR	1	
At approximately 48% Reactor power, PERFORM the following on panel 1H13-P601. VERIFY the following valves Open: • B21-F033 INBD MSL DR SOL TO MN CNDSR • B21-F069 OTBD MSL DR SOL TO MN CNDSR • OPEN B21-F016	MCR	1	
At approximately 50% Reactor Power, PERFORM the following: **SHUTDOWN 1 Reactor Feed Pump per SOI 04-1-01-N21-1.**			
VERIFY RFPT B is operating normally on master controller.	MCR	1	
RAISE FW MASTER LVL CONT setpoint to approximately 39"	MCR	1	
TRANSFER the RFPT A SP CONT to MAN.	MCR	1	
SLOWLY LOWER speed of RFPT A USING RFPT A SP CONT by DEPRESSING the OUT □ pushbutton. OBSERVE speed of RFPT B raises to maintain RPV water level, OR control it manually	MCR	1	
FURTHER REDUCE speed of RFPT A using RFPT A SP CONT in MAN until it reaches low speed stop.	MCR	1	
TRANSFER speed control of RFPT A to SPEED AUTO by DEPRESSING the OBSERVE the FW AUTO pushbutton extinguishes AND the SPEED AUTO, RAISE, AND LOWER pushbuttons backlight.	MCR	1	
FURTHER REDUCE RFPT A speed using RFPT A LOWER pushbutton.	MCR	1	
WHEN RFPT A speed reaches 1100 rpm, THEN TRIP RFPT A by DEPRESSING the RFPT A MAN TRIP pushbutton	MCR	1	
CHECK F014A, RFP A DISCH VLV starts to close.	MCR	1	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
REOPEN F014A, RFP A DISCH VLV			
WHEN RFPT A coasts down to zero speed, THEN RESET turning gear by pressing A TURN GEAR OPER RESET pushbutton. OBSERVE turning gear engages automatically, unless RFPT A is rolling on min flow.	MCR	1	
IF turning gear fails to engage, THEN MANUALLY ENGAGE the turning gear locally by PRESSING DOWN the manual engaging lever.	TURB BLDG ELEV 133 AREA 3 ROOM 1T307, 1T309	1	Not Required
CHECK OPEN/OPEN the following Drain valves on 1H22-P175: 1N11-F019A, RFPT A HP IN DR VLV 1N11-F023A, RFPT A HP IN DR VLV 1N11-F018A, RFPT A IP IN DR VLV 1N11-F021A, RFPT A IP IN DR VLV 1N11-F042A, RFPT A IP IN DR VLV 1N33-F021A, RFPT A ABOVE SEAT DR 1N33-F022A, RFPT A ABOVE SEAT DR 1N33-F023A, RFPT A BELOW SEAT DR 1N33-F024A, RFPT A BELOW SEAT DR	N/A	N/A	These steps are not required to be performed to Shut down and Cooldown the plant.
RETURN FW MASTER LVL CONT setpoint to approximately 36"	MCR	1	
IF desired, RESET RFPT A trip using the RFPT A TRIP RESET pushbutton	MCR	1	
SHUTDOWN 1 Circulating Wtr Pump per SOI 04-1-01-N71-1			
CHECK that CTCS balls are collected AND system shut down.			
DEPRESS the BALL CATCH FLAP CATCH pushbutton on P001A (B) MIMIC AND OBSERVE the flap rotates to the CATCH position.	Turb Bldg 113' Area 4 (1T203)	1	Not Required
OBSERVE ball collection starts by a rising number of balls in ball collector tank.	Turb Bldg 113' Area 4 (1T203)	1	Not Required
After 10 minutes STOP Ball Recirculation pump by DEPRESSING RECIRC PUMP OFF pushbutton on P001A(B) MIMIC	Turb Bldg 113' Area 4 (1T203)	1	Not Required
CLOSE Pump Discharge Valve F323A(B).	Turb Bldg 113' Area 4 (1T203)	1	Not Required
PLACE Screens #1 AND #2 in BACKWASH position by DEPRESSING	Turb Bldg	1	Not

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
SCREEN BACKWASH pushbutton on P001A (B) MIMIC AND OBSERVE screens rotate to BACKWASH position.	113' Area 4 (1T203)		Required
PRESS the CIRC WTR PMP A(B) STOP pushbutton on 1H13-P680.	MCR	1	
CHECK that F002A(B) Circulating Water Pump Discharge valve closes on 1H13-P680	MCR	1	
ENSURE that A(B) Circulating Water pump has shut down USING pump indication light on 1H13-P680 WHEN its discharge valve is CLOSED.	MCR	1	
OPEN OR CHECK OPEN F001 USING CIRC WTR LOOP A/B XTIE handswitch on 1H13-P870.	MCR	1	
CLOSE OR CHECK CLOSED F040A (B) Acid Feed Valve.	N/A	N/A	Not required to be performed to Shut down and Cooldown the plant.
CLOSE OR CHECK CLOSED LV-F513 A(B), Blowdown valve	MCR	1	
OPEN F039A(B), CIRC WTR PUMP A(B) COLUMN VENT	N/A	N/A	Not required to be performed to Shut down and Cooldown the plant.
ENSURE Condenser vacuum is maintained > 23.8" Hg	MCR	1	
SHUTDOWN one Heater Drain Pump per SOI 04-1-01-N23-1			
JOG CLOSED N23-F051A(B), HTR DR PMP A(B) DISCH VLV on 1H13-P680 for desired pump.	MCR	1	
STOP HTR DR PMP A(B) on 1H13-P680.	MCR	1	
WHEN Reactor power has been reduced < 40%, SHUTDOWN 2nd Heater Drain Pmp per SOI 04-1-01-N23-1			
Before securing second Heater Drain Pump, PLACE N23-LK-R053, HTR DR TK DR, in Manual AND Slowly REDUCE output to 0%.	MCR	1	
ENSURE Heater Drain Tank level is maintained by Dump Valves N23-LV-F518A-E	MCR	1	
JOG CLOSED N23-F051B(A) HTR DR PMP B(A) DISCH VLV on 1H13-P680 for second pump.	MCR	1	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
STOP Heater Drain Pump HTR DR PMP B(A) on 1H13-P680.	MCR	1	
WHEN BOTH Heater Drain Pumps are shutdown, THEN CLOSE N23-F054, HTR DR PMP COMMON DISCH VLV on 1H22 P175	TURB BLDG ELEV 133 AREA 6 ROOM 1T327	1	Not Required
IOI 03-1-01-2 Continued			
SHIFT the Reactor Recirculation Pump(s) to slow speed as follows: INSERT Control Rods until Load Line is between 50 AND 65% VERIFY Control Rods are in sequence of the Control Rod Pattern Controller.	MCR	1	
BEFORE entering Controlled Entry Region of Figure 3, PERFORM the following WHEN TS 3.3.1.1, Action J.1 is in effect: VERIFY Fraction of Core Boiling Boundary (FCBB) is ≤ 1.0 per 06 RE-1J11-V-0002. IMPLEMENT TS 3.3.1.1, Action J.2, within 12 hours of entry AND J3 within 90 days.	MCR	1	
IF any APRM gain is out of tolerance, THEN ADJUST gain per 06-RE-1C51-W-0001 prior to downshift of Recirculation Pumps.	MCR	1	
CLOSE Both Recirculation A AND B Flow Control Valves (FCV's) to Min Ed position using RECIRC A(B) FLO CONT on 1H13-P680	MCR	1	
TRANSFER Both Reactor Recirculation Pumps to slow speed per SOI 04-1-01-B33-1	MCR	1	
CONTINUE Reactor Power reduction to 25 - 30% by insertion of Control Rods	MCR	1	
SHUTDOWN Hydrogen Water Chemistry Injection per **SOI 04-1-01-P73-1.**			
At H13-P845, momentarily DEPRESS HWC SHUTDOWN pushbutton AND OBSERVE the following: HWC SHUTDOWN pushbutton 1P73-M602 Will be flashing as H2 AND O2 flows ramp down to 0. O2 isolation valves Will Close WHEN O2 levels remain at normal levels with no O2 injection for at least 5 minutes. HWC SHUTDOWN pushbutton Will be in solid WHEN all control valves AND isolation valves are fully Closed. HWC RUNNING pushbutton extinguishes.	MCR	1	
CLOSE P73-F107, H2 Inj Sply Line Man Line Shutoff valve. After O2 valves F515 AND F512 (as indicated by white dots on red cap being perpendicular to pipe) have Closed, CLOSE OR CHECK CLOSED Both F207 AND F208, O2 Rack Sply Isol to OG Preheater A(B).	N/A	N/A	Not required to be performed to Shut

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
CLOSE 1P73-F209, O2 injection to Condensate pumps			down and Cooldown the plant.
IF Drywell entry is scheduled, WHEN Reactor Power has been reduced to less than 30%, THEN PERFORM the following: PERFORM the following for 1D21-K607, DRWL PERS HATCH ARM: DIRECT I&C to CONNECT Canon plug to the plug labeled "ALARM" AND "J3" at the back of 1D21K607. PLACE Function Selector switch on front of 1D21K607 (DRWL PERS HATCH ARM) to OPERATE position. PERFORM EPI 04-1-03-D21-1 for 1D21K607.	N/A	N/A	Not required to be performed to Shut down and Cooldown the plant.
IOI 03-1-01-2 Continued **REMOVE Both Second Stage MSR Reheaters from service per SOI 04-1-01-N11-1.**			
OBSERVE PDS Computer Points N11N044A,B,C AND N11N045A,B, C to monitor LP Turbine Inlet Temperature □T during removal of Second Stage Reheaters from service.	MCR	1	
ENSURE Both MSR HTG STM FEED CONT are in MANUAL on 1H13-P680.	MCR	1	
CLOSE the following MSR 2ND STG HTG STM valves on 1H13-P680: • N11-F304C • N11-F304D	MCR	1	
SIMULTANEOUSLY CLOSE the following MSR 2ND STG HTG STM valves on 1H13-P680: • N11-F304A • N11-F304B	MCR	1	
LOWER the manual outputs on Both MSR HTG STM FEED CONT to minimum on 1H13-P680 to close the temperature control valves.	MCR	1	
CLOSE the following MSR SUPPLY VLVS valves on 1H22-P177. • N11- F028A • N11- F028B	TURB BLDG ELEV 133 AREA 4 ROOM 1T325	1	Not Required
VERIFY the following valve lineup on local panels: • N35-F015A Closed, HS-M003A • N35-F015B Closed, HS-M003B • N35-F018A Closed, HS-M007A • N35-F018B Closed, HS-M007B	N/A	N/A	Not required to be performed to Shut down and Cooldown the plant.

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
IF Feedwater Heater 6A/B are being supplied from extraction steam (i.e., IF 1N36-F010A/B AND 1N36-F011A/B on 1H13-P870 are open), THEN CLOSE the following valves on 1H22-P177: • N35-F008A, 2ND STG RHTR DR TK A TO HTR 6A • N35-F008B, 2ND STG RHTR DR TK B TO HTR 6B	TURB BLDG ELEV 133 AREA 4 ROOM 1T325	1	Not Required
REMOVE Both First Stage MSR Reheaters from service per SOI 04-1-01-N11-1.			
OPEN the following valves on 1H13-P870: • N11-F005A, MSR 1ST STG RHT RO BYP DR VLVS • N11-F005B, MSR 1ST STG RHT RO BYP DR VLVS	MCR	1	
SIMULTANEOUSLY CLOSE the following valves on 1H13-P870: • N11-F029A, HP TURB EXTR TO MSR A • N11-F029B, HP TURB EXTR TO MSR B	MCR	1	
CLOSE the following valves by taking its respective handswitch to TEST: • N11-F003A, MSR A 1ST STG RHT EXTR STM BTV (1H13-P870) • N11-F003B, MSR B 1ST STG RHT EXTR STM BTV (1H13-P870)	MCR	1	
REMOVE Condensate Precoat filters from service per SOI 04-1-01-N22-1, IF in service.			Not required to be performed to Shut down and Cooldown the plant.
OPEN the following BSCV UPSTRM DR VLV's: a. N33-F300A b. N33-F300B c. N33-F300C	MCR	1	
At approximately 23 – 26 % Reactor Power, RAISE the SPEED DEMAND setpoint to approximately 35%, as monitored on PDS Computer point N32K246, by DEPRESSING the SP DEMAND RAISE AND REL pushbuttons.	MCR	1	
SIMULTANEOUSLY DEPRESS LOAD REF OFF AND REL pushbuttons on 1H13-P680-9C to turn off load demand Control AND VERIFY OFF light is illuminated.	MCR	1	
LOWER load by DEPRESSING SPEED DEMAND LOWER AND REL pushbutton. (Expected value 150 – 175 MWe)	MCR	1	
OBTAIN Shift Manager permission for Manual Scram	MCR	1/2	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
NOTIFY the following that Main Generator is being disconnected from the grid: • Entergy Load Dispatcher (Woodlands) • (SMEPA) 1-601-261-2318 OR 1-601-261-2313) • Entergy Mississippi Dispatcher • Duty Manager	MCR	1/2	
VERIFY Switchyard lineup is acceptable for trip of J5228 AND J5232	MCR	1/2	
INSERT IRMs	MCR	1/2	
NOTIFY the following personnel/departments that a manual scram is being initiated: • Radwaste • Chemistry • Radiation Protection ANNOUNCE over plant pager that manual Scram is being initiated.	MCR	1/2	
TAKE initial temperature data per Attachment III, Data Sheet I of IOI 03-1-01-3 prior to scram	MCR	1/2	
Manually SCRAM the Reactor using the MANUAL SCRAM pushbuttons. a. VERIFY all Control Rods are fully inserted. b. VERIFY Reactor Power is decreasing. c. IF Pressure Control System is maintaining reactor pressure greater than 850 psig, THEN PLACE Reactor Mode switch to SHUTDOWN. d. VERIFY Reactor Recirculation pumps are running in slow speed.	MCR	1/2/3	
ENSURE Main Turbine and Generator trip. (Reverse power 15 seconds time delay, 5 seconds time delay IF turbine has already tripped.). a. VERIFY the Generator Output Breakers open. b. VERIFY the Turbine Stop and Control Valves close.	MCR	3	
WHEN reactor water level Can be restored AND maintained above 11.4 inches, THEN PERFORM the following to prevent Reactor water level from reaching Level 9 RFPT trip setpoint (58 in.): IF Reactor pressure is dropping rapidly, THEN SELECT SPEED AUTO OR MANUAL on the running Reactor Feed Pump AND LOWER Reactor Feed Pump discharge pressure to MAINTAIN Reactor level below 58 inches. TRANSFER Feedwater Control to Start-Up Level Control per SOI 04-1-01-N21-1. (Attachment VII of SOI 04-1-01-N21-1 May be used.)	MCR	3	
ENSURE Scram Discharge Volume Vent AND Drain valves closed	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
IOI 03-1-01-4 SCRAM Recovery			
INSERT all SRM's AND VERIFY response on SRM recorders.	MCR	3	
SWITCH IRM/APRM LVL recorders to IRM AND VERIFY neutron monitoring established on IRM's	MCR	3	
IF scram signal Can be cleared AND Reactor level AND pressure are stable, THEN RESET scram AND RETURN CRD System to normal as follows: BYPASS Scram Instrument Volume High Level signal by PLACING CRD DISCH VOL HI TRIP BYP switches RPS Div 1, 2, 3, 4 to BYPASS. RESET scram by PLACING SCRAM RESET handswitches RPS Div 1, 2, 3, 4 to RESET. VERIFY all CRDs settle into Position '00'. IF any Control Rod is NOT at the '00' position, THEN PERFORM one notch insert to attempt to force the rod to settle into the '00' position. WHEN "CRD DISCH VOL WTR LVL HI TRIP" annunciator is clear, THEN RETURN CRD DISCH VOL HI TRIP BYP switches to NORMAL. VERIFY that the HCU scram accumulators have been recharged by OBSERVING the ACCUM FAULT indicating lights on 1H13-P680 are out.	MCR	3	
THROTTLE G33-F102 to raise bottom head drain flow AND limit Bottom Head Drain Line Heatup/Cooldown to < 100°F/HR. Bottom head drain flow greater than 250 gpm May be required.	MCR	3	
IF Reactor water level is high, THEN REJECT water to Main Condenser per SOI 04-1-01-G33-1 to MAINTAIN level band.			
PLACE NSSSS OTBD MOV TEST handswitch on 1H13-P601-19B to the TEST position. VERIFY that "RX DIV 1 ISOL SYS OOSVC" annunciator (1H13-P601-19A-H3) Alarms.	MCR	3	
PLACE NSSSS INBD MOV TEST handswitch on 1H13-P601-18B to the TEST position. VERIFY that "RX DIV 2 ISOL SYS OOSVC" annunciator (1H13- P601-19A-G3) Alarms.	MCR	3	
ADJUST F033, RWCU SYS BLWDN F/D CONT VLV is ~ 10% Open.	MCR	3	
OPEN OR CHECK OPEN the following valves: F028 RWCU BLWDN CTMT INBD ISOL 1H13-P680 F034, RWCU BLWDN CTMT OTBD ISOL1H13-P680	MCR	3	
IF rejecting to main condenser, OPEN OR CHECK OPEN in the following order: F046 RWCU BLWDN TO MN CNDSR 1H13-P680	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
F041 RWCU BLWDN TO MN CNDSR BYP 1H13-P680 F235 RWCU BLWDN TO MN CNDSR 1H13-P870-3C F234 RWCU BLWDN TO MN CNDSR 1H13-P870-9C			
IF desired, while rejecting during depressurized OR low pressure conditions, F031, RWCU BLWDN ORF BYP VLV May be Open to allow maximum flow	MCR	3	
Begin rejecting by SLOWLY OPENING F033, RWCU SYS BLWDN FLO CONT valve, AND IF necessary THROTTLING CLOSED F042	MCR	3	
OBSERVE FI-R602, RWCU BLWDN FLO indicator on 1H13-P680	MCR	3	
MONITOR reactor water level, blowdown flow AND area/room temperature indication while reject is in progress.	MCR	3	
ENSURE Bypass valves are maintaining Reactor pressure	MCR	3	
IF proceeding to Cold Shutdown, THEN PERFORM Cooldown per Attachment II of IOI 03-1-01-3 concurrent with remaining steps of this attachment.	MCR	3	
DEPRESS the MHC START DVC "LOWER" pushbutton on 1H13-P680-9C to reduce the MHC START DVC to Zero.	MCR	3	
CONFIRM the following Bleeder Trip valves are Closed: a. N36-F013A, FW HTR 5A EXTR STM BTV b. N36-F013B, FW HTR 5B EXTR STM BTV c. N36-F012A, FW HTR 6A EXTR STM BTV d. N36-F012B, FW HTR 6B EXTR STM BTV e. N11-F003A, MSR A 1ST STG RHT EXTR STM BTV f. N11-F003B, MSR B 1ST STG RHT EXTR STM BTV	MCR	3	
ENSURE Seal Steam Pressure AND Reactor Feed Pump operation maintained by main steam	MCR	3	
CLOSE the following valves as soon as possible following Turbine trip at Gas Rack 1N44D001-N to isolate Hydrogen Pressure Regulators N44-PCV-F505 AND F506: a. N44-FA20 b. N44-FA21	N/A	N/A	Not required to be performed to Shut down and Cooldown the plant.
OBSERVE the following actions occur: Field amps AND generator output voltage indicate 0. Generator field breaker Will trip on a generator/transformer lockout condition (including reverse power) AND the TVR feeder switch Will open IF a lockout was NOT initiated WHEN the Turbine speed drops to ~1620 rpm. TURB AUX OIL PMPS A, B OR C starts at about 1335 rpm. AUX PW CIRC PUMP starts at about 815 rpm. (Locally)	MCR	3	Aux Primary Water Circ Pump can be verified running by computer

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
TURB SHAFT LIFT OIL PMP starts at about 510 rpm. TURB GEAR OIL VLVs N34-FE01/FE02 open at about 210 rpm.			point in the MCR.
THROTTLE P43-F053 to maintain Main Turbine Lube Oil temp between 90-119□F.	N/A	N/A	Not required to be performed to Shut down and Cooldown the plant.
WHEN fast speed trend recording is no longer necessary AND vessel level is greater than 11.4" AND vessel pressure is less than 1064.7 psig., THEN PERFORM the following: DEPRESS the POST ACC MON HI SP RESET pushbutton for POST ACC MON B21-R623A on 1H13-P601-20B. DEPRESS the POST ACC MON HI SP RESET pushbutton for POST ACC MON B21-R623B on 1H13-P601-17B.	MCR	3	
OPEN the Generator motor operated air break GEN DISC J5230. PLACE Red Tag on the Control Room handswitch for J5230 in open position. (This step May be performed after step 9.30.3)	MCR	3	
AFTER GEN DISC J5230 is opened, THEN PERFORM the following: IF tripped, THEN RESET the following Generator reverse power relays by PRESSING the relay reset rod upwards: a. 432/G12 (1N41-M752) b. 432/UT11 (1N41-M756) AFTER Generator reverse power relays are reset, THEN RESET the following Generator Lockout relays, IF tripped: a. 486-1/G12 (1N41-M769) b. 486-2/G12 (1N41-M770) c. 786-1/UT11 (1N41-M759) d. 786-2/UT11 (1N41-M760)	MCR	3	
AFTER all Generator Lockout relays are reset AND "GEN UNIT TRIP" annunciator clears on 1H13-P680-9A-A8, THEN OBTAIN Entergy Mississippi dispatcher's permission AND PERFORM the following to close breakers J5228 AND J5232 from 1H13-P680 panel: PLACE SYNC CONT BRKR J5228 switch to ON position. CLOSE 500 KV BRKR J5228. PLACE SYNC CONT BRKR J5228 switch to OFF position PLACE SYNC CONT BRKR J5232 switch to ON position. CLOSE 500 KV BRKR J5232. PLACE SYNC CONT BRKR J5232 switch to OFF position.	MCR	3	
IF all Generator Lockout relays Will NOT reset, THEN PERFORM the	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
following: CONTACT Electrical Maintenance to investigate reason any other Generator relays other than reverse power May have tripped. REQUEST Entergy Mississippi dispatcher to open disconnects to de-energize breaker(s) J5228 AND J5232.			
DEPRESS EHC SP DEMAND LOWER AND REL pushbutton on 1H13-P680-9C to reduce SP DEMAND indicator to 0 percent. WAIT for SP LTD meter to decrease to 0 percent.	MCR	3	
At each Main Transformer Control Cabinet (Phase A, Phase B, AND Phase C), VERIFY lead cooler group fans are OFF	Outside at MN XFMRs	3	Not Required
SECURE the following steam loads to limit plant cooldown: • **SJAE per SOI 04-1-01-N62-1**			
CLOSE Recombiner Drain Valves N64-F264 AND F265 (N64-F268 AND F269)	93' OG Preheater A/B Rooms 1T109 1T110	3	Not Required
CLOSE N64-F007A(B) Preheater Inlet Drain using handswitch on N64-P001.	113' Turb Area 1 1T202	3	Not Required
OPEN RECOMBINER AIR PURGE A(B) Manual Valve 1N64-F004A(B) Train A(B) Purge Air Sply Sol Byp for the corresponding recombiner train to ESTABLISH a purge flow of approximately 60 scfm through the recombiner train.	93' OG Preheater A/B Rooms 1T109 1T110	3	Not Required
CLOSE N62-F003A(B) CNDSR AIR TO 1 STG SJAE A(B) locally at 1H22 P176 OBSERVE that F003A(B) CNDSR AIR TO 1 STG SJAE A(B) indicates Closed before continuing to the next step.	133' Turb Area 1/4 1T305, 1T324	3	Not Required
DEPRESS N62-F003A(B) SJAE A(B) 1ST STG SUCT VLV CLOSE pushbutton on 1H13-P680 [10C].	MCR	3	
CHECK the indication on 1H13-P680 and the following valves Close: SJAE A(B) 1ST STG STM INL VLV, N62-F024A(B) SJAE ICNDSR DR VLV, N62-F011A(B) SJAE A(B) 2ND STG SUCT VLV, N62-F006A(B) SJAE A(B) MN STM SPLY VLV, N62-F001A(B) SJAE A(B) EXH VLV, N62-F012A(B) SJAE A(B) SEP DR VLV, N62-F002A(B)	MCR	3	
REDUCE setpoint of N62-PIC-R010A(B) to zero 0 psi	113' Turb Area 1 1T202	3	Not Required

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
ENSURE OPEN following handswitches on 1H22-P176: N62-F004A the COND AIR TO 1 STG SJAE A N62-F004B, the COND AIR TO 1 STG SJAE B	133' Turb Area 1/4 1T305, 1T324	3	Not Required
ENSURE OPEN N62-F034 A, B, C, DISCH PIPE DRN VLV for draining discharge piping.	113' Turb A MVP Area 1T218	3	Not Required
WHEN discharge piping has drained, THEN CLOSE N62-F034 A, B, C, DISCH PIPE DRN VLV.	113' Turb A MVP Area 1T218	3	Not Required
OPEN N62-F014 MECH VAC PUMPS COM SUCT VLV, at 1H22-P176.	133' Turb Area 1/4 1T305, 1T324	3	Not Required
ENSURE proper mechanical vacuum pump oil level (>50%), THEN Prelube with manual oiler as follows: ENGAGE manual oiler pump handle ROTATE for a minimum of 60 seconds. DEPRESS each plunger 5 times CHECK oil flow visible from each oil return line.	113' Turb A MVP Area 1T218	3	Not Required
CLOSE P44-F348 A(B,C) MECH VAC PMP COOLER DRAIN. OPEN P44-F109 A(B,C) MECH VAC PMP PSW INL ISOL. OPEN P44-F344 A(B,C) MECH VAC PMP PSW DISCH ISOL. BLOW DOWN strainer as follows: (1) OPEN P44-F316 A(B,C), MECH VAC PMPA(B)(C) STR DR. (2) WHEN blowdown has been completed, THEN CLOSE P44-F316 A(B,C) MECH VAC PMPA(B)(C) STR DR.	113' Turb A MVP Area 1T218	3	Not Required
START MECH VAC PMP A(B)(C) with START pushbutton on 1H13 P680.	MCR	3	
CHECK proper vacuum pump operation for each running pump by OBSERVING the following: Cooling Water Inlet Valve 1P44-SV-F514A, B, OR C has opened by MOMENTARILY OPENING drain valve 1P44-F348A, B, C MECH VAC PMP A(B)(C) CLR DR. OBSERVING pressurized water flow, THEN CLOSE drain valve1P44-F348A, B, C MECH VAC PMP A(B)(C) CLR DR. IF 1P44-SV-F514A, B, OR C did NOT open, THEN OPEN respective MECH VAC PMP A(B)(C) PSW SPLY BYP valve 1P44-F347A,B,C to provide cooling as needed for operation of Mechanical Vacuum Pump. Suction Drain Valve SV-F507A, B OR C has Closed by OBSERVING	113' Turb A MVP Area 1T218	3	Not Required

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
no air suction flow. Mechanical Vacuum Pump Inlet Valve F007A, B, OR C has Opened. Proper oiler operation by OBSERVING oil flow from each oil return line.			
Secure Seal Steam Generator per SOI 04-1-01-N33-1			
PLACE Controller PK-R617 in MANUAL on 1H13-P878, AND CLOSE F506 as necessary to control reactor cooldown. The turbine Can be sealed with seal steam header pressure as low as 15 psig, PI-R622.	MCR	3	
Secure Reactor Feed Pump per SOI 04-1-01-N21-1			All areas previously addressed in securing 1st RFPT
IOI 03-1-01-4 Continued			
Offgas Preheater by placing controllers 1N64-R009A and 1N64-R009B in manual and reducing output to 0 percent	Turbine Building 93' Area 1 (1T113)	3	Not Required
Main Steam Isolation valves AND/OR Main Steam Line	MCR	3	
SHUTDOWN a Condensate Booster Pump AND CLOSE respective discharge valve per SOI 04-1-01-N19-1, leaving one Condensate Booster Pump in service	MCR	3	
SHUTDOWN a Condensate Pump AND CLOSE respective discharge valve per SOI 04-1-01-N19-1, leaving one Condensate Pump in service	MCR	3	
CLOSE B21-F069 OPEN the following MSIV drain valves: a. B21-F067A b. B21-F067B c. B21-F067C d. B21-F067D	MCR	3	
OPEN the following valves: a. B21-F033 b. B21-F068	MCR	3	
ISOLATE extraction steam to the HP Feedwater heaters as follows: CLOSE the following valves: a. N36-F010A EXTR STM SPLY TO FW HTR 5A b. N36-F010B EXTR STM SPLY TO FW HTR 5B c. N36-F011A EXTR STM SPLY TO FW HTR 6A d. N36-F011B EXTR STM SPLY TO FW HTR 6B	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
OBSERVE the following drain valves open: N36-F008A FW HTR 6A EXTR STM RO BYP DR VLV N36-F008B FW HTR 6B EXTR STM RO BYP DR VLV	MCR	3	
OPEN the following drain valves: OPEN HP Stop AND Control Valve Drain Valves by DEPRESSING each of the following MSCV UPSTRM DR VLV "JOG OPEN" pushbuttons: a. N33-F078A b. N33-F078B c. N33-F078C d. N33-F078D OPEN Left Side Crossover piping drains by DEPRESSING each of the following XOVER PIPE LS DR VLV "JOG OPEN" pushbuttons: a. N11-F043A (FR IST) b. N11-F036A (FR 2ST) c. N11-F044A (RE IST) d. N11-F038A (RE 2ST) OPEN Right Side Crossover piping drains by DEPRESSING each of the following XOVER PIPE RS DR VLV "JOG OPEN" pushbuttons: a. N11-F044B (FR IST) b. N11-F038B (FR 2ST) c. N11-F043B (RE IST) d. N11-F036B (RE 2ST) OPEN N11-F015, MSCV A/B DNSTRM DR VLV. OPEN the following MSR 2ND STG STM DR VLVS: a. N11-F301 b. N11-F302	MCR	3	
OPEN the following drain valves unless required closed to minimize cooldown: OPEN Main Steam Line Drain Valves N11-F056, F055, F009, F011, F049, AND F050 by DEPRESSING MSL DR LINE ISOL VLVS "OPEN" pushbutton. OPEN MSL Bypass Drain valves (N11-F002A, F002B, F002C, F002D, F010, F007, F052A, F052B, F057) using MSL DR VLVS DR LINE BYP VLV "OPEN" pushbutton.	MCR	3	
DEPRESS Both NSSSS INBD ISOL RESET pushbutton (1H13-P601-18B) AND NSSSS OTBD ISOL RESET pushbutton (1H13-P601-19B) to reset logic AND re-energize RHR Logic lights on 1H13-P622 AND 1H13-P623 panels.	MCR	3	
TRANSFER to startup level control IF NOT already in service	MCR	3	
TRANSFER the RFPT A(B) SP CONT to MAN.	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
IF MSIV's are open with Main Condenser available, THEN INITIATE AND MAINTAIN cooldown at ≤ 90°F/hr with one of the following methods: CONTROL Reactor cooldown with Manual Bypass Jack on 1H13-P680-9C	MCR	3	
At approximately 200 psig Reactor pressure, SHUTDOWN one RWCU Pump per SOI 04-1-01-G33-1, IF Both are running.			
SLOWLY OPEN 1G33-F044, RWCU FLTR DMIN BYP VLV on 1H13-P680 while reducing F/D flow with flow controller 1G36-FC-R022A(B) on 1G36-P002.	MCR CTMT 185' RWCU Panel (1A509)	3	Not Required
MAINTAIN a nearly constant system flow rate, (450-500 gpm Is recommended), as indicated on 1G33-FI-R609, RWCU INL FLO, on1H13-P680.	MCR	3	
On 1G36-P002, OBSERVE that holding pump comes on WHEN F/D flow is < 80%.	CTMT 185' RWCU Panel (1A509)	3	Not Required
WHEN filter flow is < 20%, TURN Filter/Hold switch A(B) on 1G36-P002 to HOLD position. OBSERVE the following valves fully Close: • G36-F001A(B) F/D Inlet • G36-F002A(B) F/D Inlet • G36-F003A(B) F/D Outlet • G36-F004A(B) F/D Outlet	CTMT 185' RWCU Panel (1A509)	3	Not Required
OBSERVE HOLD light on AND FILTER light out on 1G36-P002	CTMT 185' RWCU Panel (1A509)	3	Not Required
PLACE the MANUAL/AUTO selector on controller 1G36-FC-R022A (B) in MANUAL position with controller output at 0% output.	CTMT 185' RWCU Panel (1A509)	3	Not Required
REPEAT Steps 4.6.2a AND 4.6.2b for second F/D.	CTMT 185' RWCU Panel (1A509)	3	Not Required
LOWER system flow rate to < 280 gpm by THROTTLING 1G33F044 as indicated on 1G33FI-R609, RWCU INL FLO, on 1H13-P680.	MCR	3	
TRIP one of the running RWCU pumps	MCR	3	
ESTABLISH 90 to 300 gpm flow as indicated on 1G33-FI-R609, RWCU INL FLO, on 1H13-P680 by THROTTLING the Bypass Valve 1G33F044	MCR	3	
WHEN Reactor pressure is reduced to < 135 psig, THEN at			

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
approximately 40 psig, PLACE one loop of RHR System in SHUTDOWN COOLING mode per SOI 04-1-01-E12-2.			
RACK OUT RHR A/B PMP Breaker, 152-1509/1606	Control Bldg. 111' SWGR Rms 0C202, 0C215	3	**Required**
SHUTDOWN RHR JOCKEY PUMP A/B on 1H13-P871.	MCR	3	
CLOSE F082A/B, RHR JCKY PMP SUCT ISOL VLV, on 1H13-P871.	MCR	3	
CLOSE F064A/B, RHR MIN FLO TO SUPP POOL.	MCR	3	
CLOSE F004A/B, RHR PMP SUCT FM SUPP	MCR	3	
ENSURE OPEN F003A/B, RHR HX OUTL VLV	MCR	3	
ENSURE OPEN F048A/B, RHR HX A BYP VLV.	MCR	3	
CLOSE F047A/B, RHR HX INL VLV.	MCR	3	
CLOSE F428A/B, PRESSURE LOCK ISOL for F024	RHR A/B Pump Rm Aux Bldg 93' (1A103/1A105)	3	**Required**
CLOSE F438A/B, PRESSURE LOCK ISOL for F064	RHR A/B Pump Rm Aux Bldg 93' (1A103/1A105)	3	**Required**
SLOWLY OPEN F020, Manual Flush Valve.	Aux Bldg 119' RCIC Rm (1A204)	3	**Required**
OPEN F006A, RHR PMP A SUCT FM SHUTDN CLG AND MONITOR RHR HR A STM press indicator for rise in pressure.	MCR	3	
VENT Shutdown Cooling suction header as follows: (a) OPEN F323. (b) OPEN F399. (c) WHEN a solid stream of water is observed out of vent line, THEN CLOSE F399. (d) CLOSE F323.	Aux Bldg 119' RCIC Rm (1A204)	3	**Required**
OPEN F073A, RHR HX A OTBD VENT VLV.	MCR	3	
OPEN F074A, RHR HX A INBD VENT VLV.	MCR	3	
VENT RHR A Heat Exchanger A as follows: (a) OPEN F400A, A RHR HX VENT.	Aux. 139' RHR A/B Rm	3	**Required**

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
(b) OPEN F401A, A RHR HX VENT. (c) WHEN water is observed from vent, THEN CLOSE F401A. (d) CLOSE F400A.	1A303, 1A304/1A306 , 1A307		
OPEN F064A/B. AFTER approximately one minute, THEN CLOSE F064A/B	MCR	3	
WHEN Conductivity as indicated on HX A/B OUT CNDCT, is as low as practical (Should be less than 2.0 µmhos/cm), THEN CLOSE F073A/B, RHR HX A/B OTBD VENT VLV.	MCR	3	
CLOSE F074A/B, RHR HX A/B INBD VENT VLV	MCR	3	
LOCK CLOSED F020, Manual Flush Valve.	Aux Bldg 119' RCIC Rm (1A204)	3	**Required**
CLOSE F048A/B	MCR	3	
OPEN F063A/B, Manual Flush Valve.	RHR A/B Pump Rm Aux Bldg 119' (1A203/1A205)	3	**Required**
OPEN F073A/B, RHR HX A/B OTBD VENT VLV	MCR	3	
OPEN F074A/B, RHR HX A/B INBD VENT VLV	MCR	3	
WHEN Conductivity as indicated on HX A/B OUT CNDCT, is as low as practical (Should be less than 2.0 µmhos/cm), THEN CLOSE F073A/B, RHR HX A/B OTBD VENT VLV.	MCR	3	
CLOSE F074A/B, RHR HX A/B INBD VENT VLV	MCR	3	
LOCK CLOSED F063A/B, Manual Flush Valve.	RHR A/B Pump Rm Aux Bldg 93' (1A203/1A205)	3	**Required**
OPEN F048A.	MCR	3	
OPEN F047A.	MCR	3	
ENSURE OPEN F003A.	MCR	3	
ENSURE Shutdown Cooling Isolation Logic is reset by PRESSING NSSSS INBD ISOL RESET pushbutton AND NSSSS OTBD ISOL RESET pushbutton on 1H13-P601.	MCR	3	
PLACE Standby Service Water A System in service to RHR A Heat Exchanger on 1H13-P870 as follows. START SSW Pump A per SOI 04-1-01-P41-1.	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
OPEN P41-F014A, SSW INL TO RHR HX A. ENSURE OPEN P41-F068A, SSW OUTL FM RHR HX A. START RHR RM A FAN COIL UNIT.			
ENSURE OPEN F010, SHUTDN CLG MAN SUCT VLV.	MCR	3	
ENSURE CLOSED F040, RHR TO RADWST OTBD SHUTOFF VLV.	MCR	3	
ENSURE CLOSED F049, RHR TO RADWST INBD SHUTOFF VLV.	MCR	3	
OPEN F020, Manual Flush Valve approximately 3 turns. Valve May be opened further IF required for level control.	Aux Bldg 119' RCIC Rm (1A204)	3	**Required**
OPEN F008, RHR SHUTDN CLG OTBD SUCT VLV	MCR	3	
OPEN F009, RHR SHUTDN CLG INBD SUCT VLV as follows; ENSURE breaker 52-163137 is CLOSE position OPEN F009, RHR SHUTDN CLG INBD SUCT VLV MONITOR Reactor water level WHILE 1E12F009 AND 1E12F020 are OPEN. PERFORM IMMEDIATELY the next step 4.1.2.b(14) IF a rise in Reactor water level is NOT desired.	MCR	3	
LOCK CLOSED F020, Manual Flush Valve.	Aux Bldg 119' RCIC Rm (1A204)	3	**Required**
NOTIFY Radwaste Operators to be prepared for Reactor water flush to Waste Surge tank.	MCR Radwaste Building 118' Radwaste Control Room (0R241)	3	**Required**
OPEN F203, RHR SYS FLUSH TO LIQ RADWST by the following handswitches to OPEN: F203 SVA-RHR SYS FLUSH TO LIQ RADWST (1H13-P870-3C) F203 SVB-RHR SYS FLUSH TO LIQ RADWST (1H13-P870-8C)	MCR	3	
ENSURE CLOSED F070A/B, Manual RHR Drain Valve	Aux Bldg 93' Corridor (1A101)	3	**Required**
OPEN F072A/B, RHR Drain Valve	RHR A/B Pump Rm Aux Bldg 93' (1A103/1A105)	3	**Required**

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
SLOWLY OPEN F070A, RHR Drain Valve approximately one turn to start flow to Radwaste. IF "RHR A DISCH PRESS ABNORMAL" annunciator alarms while warming RHR A, THEN CLOSE F047A AND F048A to prevent draining of downstream piping.	Aux Bldg 93' Corridor (1A101)	3	**Required**
THROTTLE F070A/B to warm RHR Pump A/B at less than 100°F/hr until RHR DISCH TO RADWST ON RHR TEMP recorder is 200°F OR within 100°F of RX water temp, whichever is less.	Aux Bldg 93' Corridor (1A101)	3	**Required**
LOCK CLOSED F070A, RHR Drain Valve.	Aux Bldg 93' Corridor (1A101)	3	**Required**
LOCK CLOSED F072A, RHR Drain Valve.	RHR A/B Pump Rm Aux Bldg 93' (1A103/1A105)	3	**Required**
CLOSE F203, RHR SYS FLUSH TO LIQ RADWST by TAKING the following handswitches to CLOSE: F203 SVA-RHR SYS FLUSH TO LIQ RADWST (1H13-P870-3C) F203 SVB-RHR SYS FLUSH TO LIQ RADWST (1H13-P870-8C)	MCR	3	
RACK IN RHR A/B PMP Breaker, 152-1509/1606	Control Bldg. 111' SWGR Rms 0C202, 0C215	3	**Required**
NOTIFY Chemistry AND Radiation Protection that possibility of a crud burst Could occur due to starting of RHR pump in SDC mode	MCR	3	
START OR ENSURE running RHR RM A FAN COIL UNIT on 1H13-P870.	MCR	3	
ENSURE CLOSED F064A, RHR A MIN FLO TO SUPP POOL.	MCR	3	
ENSURE RHR JOCKEY PUMP A is shutdown.	MCR	3	
ENSURE CLOSED F082A, RHR A JCKY PMP SUCT ISOL VLV.	MCR	3	
ENSURE CLOSED F004A, RHR A SUCT FM SUPP POOL.	MCR	3	
ENSURE OPEN the following valves: (a) F010 (Concurrent Verification Required) (b) F008 (c) F009 as follows; (1) ENSURE breaker 52-163137 is CLOSE position (2) ENSURE OPEN F009, RHR SHUTDN CLG SUCT VLV (d) F006A	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
(e) F047A (f) F048A			
CLOSE F003A, RHR HX A OUTL VLV .	MCR	3	
ENSURE CLOSED B21-F065A, FW INL SHUTOFF VLV.	MCR	3	
START RHR PMP A AND IMMEDIATELY FULLY OPEN one of the following valves: (a) E12-F053A, RHR A SHUTDN CLNG RTN TO FW (b) E12-F037A, RHR A TO CTMT POOL (c) E12-F042A, RHR A INJ SHUTOFF VLV	MCR	3	
MONITOR RHR HX A differential temperature on RHR TEMPERATURE RECORDER as follows: RHR HX A Point 1(inlet) - Point 5(outlet)	MCR	3	
ESTABLISH a cool down rate of less than 90°F/hr, as follows: Slowly JOG OPEN F003A to allow flow through heat exchanger, AND MONITOR cooldown rate. THROTTLE one of the following valves to maintain RHR pump flow ~8600 gpm AND RHR heat exchanger flow ~8200 gpm: IF flow is through F053A, THEN THROTTLE F053A AS LONG AS flow through valve is maintained < 8550 gpm.	MCR	3	
IF E12-F003A is closed while in SHUTDOWN COOLING, THEN MONITOR REACTOR COOLANT TEMPERATURE using the following indications: REACTOR RECIRC LOOP A/B suction temperature (IF recirc pump(s) running) RWCU REGENERATIVE HEAT EXCHANGER INLET temperature (IF RWCU pump(s) are running.) Point 5 of RHR TEMPERATURE RECORDER. Installed thermocouple suspended above Reactor core.	MCR	3	
WHEN F003A valve is full open AND additional cooling is required, THEN SLOWLY THROTTLE CLOSE F048A as needed to establish desired cooldown rate.	MCR	3	
WHEN F048A valve is full closed, THEN, IF desired, THROTTLE F003A to MAINTAIN desired coolant temperature OR SDC flow while MAINTAINING ≥ 3000 gpm flow. F048A may be fully opened to reduce cooldown rate but CANNOT be left in a throttled position UNTIL F003A is full open.	MCR	3	
SELECT "Shutdown Cooling-RHR A" OP GUIDE on PDS computer. The guide Should be left on-screen OR icon'd WHEN the respective shutdown cooling loop is in service until Reactor Coolant has been stabilized at desired temperature so that the guide Will warn operators IF Shutdown Cooling parameters are out of range	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
LOG Reactor coolant temperature on Data Sheet I of 03-1-01-3 OR other applicable IOI. TAKE temperatures as required by 03-1-01-3 during cooldown AND CONTINUE to take readings once per hour WHEN temperature is stable.	MCR	3	
LOG temperatures for SSW/RHR HX AND reactor coolant on log similar to Attachment I to ENSURE SSW temperature does NOT exceed design temperatures. (Ref. CR1997-0282) IF SSW A auto start signal from RHR A pump running is defeated by Temporary Alteration, THEN START/STOP SSW A AND B fans as necessary to MAINTAIN SSW A Supply temp. (E12-R601, pt. 12) between 50 AND 75 deg.	MCR	3	
IF RPV level control via RWCU blowdown is unavailable, THEN RPV level control May be established by USING E12-F073A AND E12-F074A RHR heat exchanger vent to establish RPV level control, AND THROTTLE OPEN E12-F073A AND E12-F074A as required to establish AND maintain the desired RPV level. MONITOR RPV level while reject is in progress.	MCR	3	
IF desired to add water to Reactor with SDC in operation WHEN in Modes 4 OR 5, THEN PERFORM the following: THROTTLE OPEN, F020. WHEN desired Reactor Vessel Level is reached, THEN LOCK CLOSED F020.	Aux Bldg 119' RCIC Rm (1A204)	4, 5	Not Required
IOI 03-1-01-3 Continued			
At approximately 120 psig, PERFORM the following: **TRANSFER RWCU to Pre-pump mode per SOI 04-1-01-G33-1.**			
PLACE NSSSS OTBD MOV TEST handswitch on 1H13-P601-19B to the TEST position. VERIFY that "RX DIV 1 ISOL SYS OOSVC" annunciator (1H13-P601-19A-H3) Alarms.	MCR	3	
PLACE NSSSS INBD MOV TEST handswitch on 1H13-P601-18B to the TEST position. VERIFY that "RX DIV 2 ISOL SYS OOSVC" annunciator (1H13- P601-19A-G3) Alarms.	MCR	3	
SECURE RWCU blowdown flow per Section 5.1 of this instruction.	MCR	3	
STOP running RWCU pump AND leave F044, RWCU FLTR DMIN BYP VLV THROTTLED SLIGHTLY OPEN.	MCR	3	
CLOSE the following valves AND proceed to Step 4.4.2g without delay: F250 RWCU SPLY TO RWCU HXS 1H13-P870-3C F251 RWCU SPLY TO RWCU HXS 1H13-P870-9C	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
F252 RWCU HX RTN TO RWCU PMPS 1H13-P870-9C F253 RWCU HX RTN TO RWCU PMPS 1H13-P870-3C F255 RWCU FLTR/DMIN INL FM RWCU PMP 1H13-P870-5C			
OPEN OR CHECK OPEN the following valves: F004 PMP SUCT CTMT OTBD ISOL 1H13-P680 F001 PMP SUCT DRWL INBD ISOL 1H13-P680 F254 RWCU FLTR/DMIN INL FM RWCU HX 1H13-P870-5C F256 RWCU HX INL FM RWCU PMP 1H13-P870-5C	MCR	3	
CLOSE OR CHECK CLOSED F044, RWCU FLTR DMIN BYP VLV; And THEN RESTART one RWCU pump AND JOG OPEN F044 to establish flow greater than 90 gpm but less than 300 gpm.	MCR	3	
START one RWCU the pump AND THROTTLE F044 to achieve a system flow greater than 90 gpm, But less than 300 gpm as indicated on FI-R609, RWCU INL FLO. IF performing system warm-up. THEN MAINTAIN minimum flow, AVOIDING low flow trip.	MCR	3	
IF desired, START a second pump as follows: START the pump AND THROTTLE F044 to maintain 300 - 500 gpm system flow as indicated on FI-R609, RWCU INL FLO, with Both Pumps running.	MCR	3	
IF desired, ESTABLISH RWCU blowdown flow in accordance with Section of this instruction			All areas previously addressed for this evolution
IF desired, PLACE F/Ds in service in accordance with Section 4.5 of this instruction.			All areas previously addressed for this evolution
PLACE NSSSS OTBD MOV TEST handswitch on 1H13-P601-19B to the NORM position. VERIFY that "RX DIV 1 ISOL SYS OOSVC" annunciator (1H13-P601-19A-H3) Clears.	MCR	3	
PLACE NSSSS INBD MOV TEST handswitch on 1H13-P601-18B to the NORM position. VERIFY that "RX DIV 2 ISOL SYS OOSVC" annunciator (1H13- P601-19A-G3) Clears.	MCR	3	
IOI 03-1-01-3 Continued			
SHUTDOWN the running Condensate Booster Pump AND CLOSE respective discharge valve per SOI 04-1-01-N19-1.	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
IF scheduled, THEN PERFORM 06-OP-1B21-R-0010 (Att. I AND/OR II) WHEN reactor pressure is between 50 AND 100 psig			Not required to be performed to Shut down and Cooldown the plant.
At approximately 60 psig Reactor pressure, PERFORM the following: VERIFY that RCIC system isolates automatically. IMMEDIATELY NOTIFY CAS, SAS, OR Security Island that RCIC is not available (non-functional). COMPLETE shutdown of RCIC system per SOI 04-1-01-E51-1.	MCR	3	
WHEN cooldown using Bypass Valves is no longer desired AND Shutdown Cooling is in service, THEN CLOSE the Bypass Valves as follows: SET the TURB STM PRESS DEMAND setpoint approximately 100 psig above Reactor pressure using the PRESS REF "RAISE" OR "LOWER" pushbuttons on 1H13-P680-9C. DEENERGIZE the Manual Bypass Valve Controller by depressing the MAN BYP CONT "OFF" pushbutton on 1H13-P680-9C.	MCR	3	
IF MSIV's are open AND stroke time testing was NOT scheduled, THEN PERFORM the following: CLOSE the following Inboard MSIVs: • B21-F022A • B21-F022B • B21-F022C • B21-F022D WHEN Main Steam Line pressure downstream of MSIVs is near zero psig, THEN CLOSE the following Outboard MSIVs: • B21-F028A • B21-F028B • B21-F028C • B21-F028D CLOSE B21-F016 CLOSE B21-F019	MCR	3	
NOTIFY Radiation Protection that the Reactor is to be vented to Drywell sump AND REQUEST Drywell survey after Head Vent realignment.	MCR	3	
WHEN Reactor coolant temperature is less than 210°F, THEN REALIGN Reactor Head Vents on 1H13-P601 as follows: OPEN 1B21-F001, RPV OTBD VENT VLV.	MCR	3	

IOI / SOI ACTIONS	LOCATION	MODE	NOTES
PEN 1B21-F002, RPV INBD VENT VLV. CLOSE 1B21-F005, RPV VENT TO MSL A.			

Control Room ventilation systems have adequate engineered safety/design features in place to preclude a Control Room evacuation due to the release of a hazardous gas. Therefore, the Control Room is not included in this assessment or in Table H-2.

Table A-3 & H-2 Results

Table A-3 & H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode**
Control Building 111' SWGR Rms (0C202, 0C215)	3
Auxiliary Building 93' RHR A Pump Room (1A103)	3
Auxiliary Building 93' RHR B Pump Room (1A105)	3
Auxiliary Building 93' Corridor (1A101)	3
Auxiliary Building 119' Corridor (1A201)	3
Auxiliary Building 119' RHR A Pump Room (1A203)	3
Auxiliary Building 119' RHR B Pump Room (1A205)	3
Auxiliary Building 119' RCIC Room (1A204)	3
Auxiliary Building 139' RHR A Room (1A303, 1A304)	3
Auxiliary Building 139' RHR B Room (1A306, 1A307)	3
Radwaste Building 118' Radwaste Control Room (0R241)	3