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10 CFR 50.90

August 13, 2018
Serial: HNP-18-004

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

Shearon Harris Nuclear Power Plant (HNP), Unit 1
Docket No. 50-400
Renewed License No. NPF-63

Subject: License Amendment Request to Change Shearon Harris Nuclear Power Plant, Unit 1, Emergency Plan Emergency Action Level Scheme

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Duke Energy Progress, LLC, (Duke Energy) is requesting approval of proposed changes in the Emergency Plan Emergency Action Levels (EALs) used at the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed changes to the EAL scheme will correct deficiencies identified and bring the site into alignment with the approved EAL methodology, Nuclear Energy Institute (NEI) 99-01 Revision 6, “Development of Emergency Action Levels for Non-Passive Reactors.”

Enclosure 1 provides an evaluation of the proposed changes to the EAL scheme. Enclosure 2 provides the impacted pages of the EAL Technical Bases Document (clean version) for HNP. Enclosures 3 provides a markup version of the impacted pages of the EAL Technical Bases Document. Enclosure 4 provides the supporting calculation for the changes made to the containment radiation monitor EAL threshold values listed in the HNP EAL Table F-1, “Fission Product Barrier Threshold Matrix.” Enclosure 5 provides the supporting calculation for the change made to the radiation monitor reading for core uncovery during refueling that is used for cold shutdown EAL thresholds. Enclosure 6 provides a markup version of the HNP EAL Wallcharts for the proposed changes.

Duke Energy requests Nuclear Regulatory Commission review and approval of this license amendment request within one year of acceptance. The amendment shall be implemented within 180 days following approval.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated North Carolina State Official.

This document contains no new regulatory commitments.

Should you have any questions regarding this submittal, please contact Jeff Robertson, HNP Regulatory Affairs Manager, at (919)-362-3137.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on August 13 , 2018.

Sincerely,

Tanya M. Hamilton

Tanya M. Hamilton

Enclosures:

1. Evaluation of the Proposed Changes to the Emergency Plan Emergency Action Level Scheme
2. Proposed Harris Nuclear Plant Emergency Action Level Technical Bases Document Changes, EP-EAL (Clean)
3. Proposed Harris Nuclear Plant Emergency Action Level Technical Bases Document Changes, EP-EAL (Markup)
4. Calculation for Containment Radiation EAL Threshold Values
5. Calculation for Radiation Monitor Readings for Core Uncovery during Refueling
6. Proposed Harris Nuclear Plant Emergency Action Level Wallchart Changes (Markup)

cc: J. Zeiler, NRC Sr. Resident Inspector, HNP
W. L. Cox, III, Section Chief, N.C. DHSR
M. Barillas, NRC Project Manager, HNP
NRC Regional Administrator, Region II

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SERIAL HNP-18-004

ENCLOSURE 1

EVALUATION OF THE PROPOSED CHANGES TO THE EMERGENCY PLAN EMERGENCY ACTION LEVEL SCHEME

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

1.0 SUMMARY DESCRIPTION

In accordance with the provisions of 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), is proposing three changes to the Shearon Harris Nuclear Power Plant, Unit 1 (HNP), Emergency Plan Emergency Action Level (EAL) scheme. The changes are associated with the Fission Product Barrier (FPB) Degradation EAL thresholds and the Cold Shutdown/Refueling System Malfunction EAL thresholds.

Duke Energy proposes the following three changes.

1. **Hot Operating Mode Loss of Reactor Coolant System Threshold:** The HNP EAL Technical Bases Document, EP-EAL, includes declarations during hot operating modes (Modes 1 through 4) based on FPB integrity described in Table F-1, “Fission Product Barrier Loss/Potential Loss Matrix and Bases.” EP-EAL, Table F-1, includes a condition for determining ‘Loss of Reactor Coolant System’ (RCS) based on a noble gas radiation monitor (REM-1LT-3502A-SA) used for containment leak detection. The EP-EAL basis description states that the listed threshold “readings assume the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the containment atmosphere.” The design of the noble gas monitor does not support this basis statement. Instantaneous release of all reactor coolant would trigger a safety injection (SI) and would isolate containment, which isolates flow to the radiation monitor rendering it unavailable. The current threshold does not adversely impact EAL declaration since a safety injection signal would alert decision makers at HNP of a RCS loss. Duke Energy proposes a change to the 'Loss of RCS loss' EAL that is more appropriate for an instantaneous release of all reactor coolant.

2. **Hot Operating Mode Loss of Fuel Clad and Containment Thresholds:** The EP-EAL, Table F-1, includes a ‘Loss of Fuel Clad Barrier (FC)’ declaration and a ‘Potential Loss of Containment (CNMT) Barrier’ declaration based on CNMT radiation levels as measured at the Containment High-Range Radiation Monitors (CHRRMs), RM-1CR-3589SA or RM-1CR-3590SB. The existing threshold values were developed in accordance with the methodology described in NUREG-0654/FEMA-REP-1, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," November 1980 (Reference 6.5). The methodology approved for use at HNP is Nuclear Energy Institute (NEI) 99-01 Revision 6, “Development of Emergency Action Levels for Non-Passive Reactors," as identified in Reference 6.2. Duke Energy proposes a change to the CHRRM threshold values for 'Loss of FC' and 'Potential Loss of CNMT' in Table F-1 to values developed in accordance with NEI 99-01, Revision 6, methodology.

3. **Cold Shutdown Loss of RCS Inventory Threshold:** The EP-EAL includes declarations for core uncovery during cold shutdown and refueling modes under EALs CS1.3 and CG1.2 that are based upon CNMT radiation levels as measured at the CHRRMs. The current threshold value for EALs CS1.3 and CG1.2 is above radiation levels anticipated at the CHRRMs if core uncovery were to occur. In addition, due to the CHRRMs location and distance from the core, the CHRRMs are not ideal for monitoring loss of water shielding from the reactor cavity due to conditions leading to core uncovery. Alternate thresholds remain available to ensure core uncovery would result in the appropriate EAL declaration. However, establishing an appropriate method for monitoring radiation levels

in CNMT would restore diversity for declarations associated with core uncovery. Thus, Duke Energy proposes a change in the method for measuring CMNT radiation levels from a method utilizing the CHRRMs to a method more capable of determining expected CNMT radiation levels in the event of core uncovery during cold shutdown or refueling.

2.0 DETAILED DESCRIPTION

2.1 Hot Operating Mode Loss of RCS Threshold

2.1.1 System Design and Operation

As part of the Emergency Plan, Duke Energy maintains the ability to systematically declare EALs per the pre-planned scheme described in EP-EAL at HNP. The purpose of the EP-EAL document is to provide an explanation and rationale for each EAL. Decision makers use this document as a technical reference in support of EAL interpretation.

The EAL scheme includes ‘Category F’ for FPB degradation. This EAL category is applicable in hot operating modes (RCS temperature > 200°F, which is applicable to Modes 1 through 4) and represent threats to the defense-in-depth design concept that precludes the release of highly radioactive fission products to the environment. This EAL scheme evaluates threats to the FC, RCS, and CNMT.

Assessment of each FPB’s ‘Loss’ or ‘Potential Loss’ is performed per the Fission Product Barrier Threshold Matrix, Table F-1. ‘Loss’ and ‘Potential Loss’ signify the relative damage and threat of damage to a FPB. ‘Loss’ means the FPB no longer assures containment of radioactive materials. ‘Potential Loss’ means integrity of the FPB is threatened and could be lost if conditions continue to degrade. The number of FPBs that are lost or potentially lost determine the appropriate emergency classification level per the following criteria:

- Alert: Any loss or any potential loss of either FC or RCS
- Site Area Emergency (SAE): Loss or potential loss of any two barriers
- General Emergency (GE): Loss of any two barriers and loss or potential loss of third barrier

Table F-1 includes thresholds for determining ‘Loss of RCS’ based on the readings from the containment leak detection noble gas monitor, REM-1LT-3502A-SA. The purpose of this monitor is to identify a RCS release into CNMT with reactor coolant noble gas and iodine inventory at normal operating levels. Because of the very high fuel clad integrity, only small amounts of noble gases would be dissolved in the primary coolant. Thus, the REM-1LT-3502A-SA threshold value is lower than thresholds used to identify ‘Loss of FC’.

2.1.2 Current Emergency Plan Requirements

The current Emergency Plan EAL scheme would declare a ‘Loss of RCS’ whenever REM-1LT-3502A-SA detected noble gas concentrations greater than 8.3E-3 microcuries per milliliter (μCi/ml). This threshold value would lead to an Alert declaration, since at this point in an accident, the RCS would be considered lost, but FC and CNMT would not be impacted. This value is consistent with a significant leak of RCS with reactor coolant activity equal to Technical Specification limits.

2.1.3 Reason for Proposed Change

The current EAL threshold for determining 'Loss of RCS' should be based on the expected radiation readings if there were an instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory into CNMT with reactor coolant activity equal to Technical Specification limits. This is the method specified per NEI 99-01, Revision 6 (Reference 6.3), which is the current EAL methodology approved by the NRC for use at HNP. However, the design of REM-1LT-3502A-SA does not support operation during an instantaneous loss of all RCS inventory. The inlet to this radiation monitor is equipped with a CNMT isolation valve designed to close on SI initiation. A SI would be expected with instantaneous loss of all RCS, rendering this monitor unavailable.

Additionally, the current threshold values are a single number, meant to be representative of a wide array of events over a wide array of time periods. Duke Energy has identified that expected conditions in CNMT can vary greatly as an accident progresses, with dosage in CNMT changing relative to time from the accident. Thus, Duke Energy proposes using multiple thresholds for determining 'Loss of RCS' based on the time from initiation of the accident. While not required, multiple thresholds will enhance the overall accuracy of the scheme and reduce the possibility of unnecessary declarations.

2.1.4 Description of Proposed Change

Duke Energy proposes implementing a new method for declaring 'Loss of RCS' in cases where no FC damage is present. Instead of using REM-1LT-3502A-SA, Duke Energy will use the Containment Ventilation Isolation (CVI) radiation monitors, as these monitors are located inside containment, at the operating deck level around the refueling cavity, and will continuously monitor containment radiation levels in the event of a safety injection. The thresholds used will be consistent with a loss of all RCS inventory assuming that reactor coolant activity equals Technical Specification allowable limits. This method is also consistent with the Reference 6.3 specified threshold of, "Containment Radiation monitor reading greater than (site specific value)."

Thus, the current threshold in the Fission Product Barrier Matrix Table F-1 for 'Loss of RCS' will be changed as shown below:

> Current threshold for declaring 'Loss of RCS': "Containment Leak Detection Monitor Noble Gas (REM-1LT-3502A-SA) > 8.3E-3 μCi/ml."
>
> Proposed threshold for declaring 'Loss of RCS': "A Containment Ventilation Isolation Radiation Monitor (RM-1CR-3561A-SA, RM-1CR-3561B-SB, RM-1CR-3561C-SA, or RM-1CR-3561D-SB):
>
> > 1.37E+03 mR/hr [millirem per hour] from T=0 hr to T=1 hr,
> > 1.12E+03 mR/hr from T>1 hr to T=2 hrs,
> > 6.35E+02 mR/hr from T>2 hrs to T=8 hrs
> > 1.37E+02 mR/hr from T> 8 hrs.

The proposed changes to the EAL threshold description and bases are shown in Enclosure 2 (clean version of EP-EAL that incorporates changes) and Enclosure 3 (redline and strikeout version of EP-EAL changes). Enclosure 6 shows the proposed changes to the HNP EAL Wallcharts.

The proposed EAL threshold is within the CVI radiation monitor's design operating range of 10^1 - 10^7 mR/hr (Reference 6.11).

2.2 Hot Operating Mode Loss of Fuel Clad and Containment Thresholds

2.2.1 System Design and Operation

The HNP EAL scheme includes Category F for declaring FPB degradation. This EAL scheme includes the Fission Product Barrier Threshold Matrix, Table F-1, which is a matrix of thresholds for determining event classification based on conditions affecting FPBs. This matrix includes thresholds for determining FPB ‘Loss’ and ‘Potential Loss’ based on CNMT radiation levels. Two of these thresholds include ‘Loss of FC’ declared when CHRRMs register radiation levels above 150 Rem per hour (R/hr), and ‘Potential Loss of CNMT’ declared when CHRRMs register radiation levels above 600 R/hr.

2.2.2 Current Emergency Plan Requirements

The current Emergency Plan EAL scheme requires a SAE declared when the CHRRMs identify radiation levels over 150 R/hr and a GE declared when radiation levels are over 600 R/hr. Note, a declaration of ‘Loss of RCS’ is assumed, since without a loss of RCS it would not be possible to achieve the FC damage required to drive radiation levels up to 150 R/hr or 600 R/hr. Thus, 150 R/hr would result in a classification of a loss of two barriers, RCS and FC, and an SAE would be declared. The 600 R/hr radiation level may only occur after the loss of RCS and FC, and adds a ‘Potential Loss of CNMT’ declaration, resulting in a GE.

The current CHRRM threshold values were developed in accordance with the methodology described in NUREG-0654 (Reference 6.5).

2.2.3 Reason for Proposed Change

The current licensing basis for the HNP EAL scheme is NEI 99-01, Revision 6 (Reference 6.3), as approved by the NRC in a letter dated April 13, 2016 (Reference 6.2). The EAL scheme transitioned from a NUREG-0654 (Reference 6.5) based scheme to an NEI 99-01, "Methodology for Development of Emergency Action Levels," Revision 5, based scheme following NRC approval in a letter dated April 25, 2010 (Reference 6.1). The CHRRM thresholds contained within the current EAL scheme are based upon calculation methodologies used for the NUREG-0654 (Reference 6.5) based scheme instead of NEI 99-01, Revision 6 (Reference 6.3), EAL scheme guidance. Duke Energy proposes changing the CHRRM threshold values in Table F-1 to values developed in accordance with the methodology described in NEI 99-01, Revision 6 (Reference 6.3), to correct this condition.

Additionally, the current threshold values are a single number, meant to be representative of a wide array of events over a wide array of time periods. Duke Energy has identified that expected conditions in CNMT can vary greatly as an accident progresses. The dosage at the CHRRMs for a given amount of FC damage will change relative to time from the accident. Thus, Duke Energy proposes using multiple thresholds for determining a SAE or GE at the CHRRMs based on the time from initiation of the accident. While not required, multiple thresholds will enhance the overall accuracy of the scheme and reduce the possibility of unnecessary declarations.

2.2.4 Description of the Proposed Change

The current thresholds in the Fission Product Barrier Matrix Table F-1 for CNMT Radiation / RCS Activity will be revised as described below:

Current threshold for declaring ‘Loss of FC’: > 150 R/hr at the CHRRMs
Future threshold for declaring ‘Loss of FC’:
> 130 R/hr at the CHRRMs from T=0 hr to T=1 hr,
> 110 R/hr at the CHRRMs from T >1 hrs to T=2 hrs,
> 70 R/hr at the CHRRMs from T >2 hrs to T=8 hrs,
> 21 R/hr at the CHRRMs for T > 8 hrs.

Current threshold for declaring ‘Potential Loss of CNMT’: > 600 R/hr at the CHRRMs
Future threshold for declaring ‘Potential Loss of CNMT’:
> 2360 R/hr at the CHRRMs from T=0 hr to T=1 hr,
> 2000 R/hr at the CHRRMs from T >1 hr to T=2 hrs,
> 1300 R/hr at the CHRRMs from T >2 hrs to T=8 hrs,
> 390 R/hr at the CHRRMs for T > 8 hrs.

The proposed EAL thresholds are within the CHRRM's design operating range of 10^0 - 10^8 R/hr (Reference 6.11). The proposed changes to the EAL threshold descriptions and bases are shown in Enclosure 2 (clean version of EP-EAL that incorporates changes) and Enclosure 3 (redline and strikeout version of EP-EAL changes). Enclosure 6 shows the proposed changes to the HNP EAL Wallcharts.

2.3 Cold Shutdown Loss of RCS Inventory Threshold

2.3.1 System Design and Operation

The EAL scheme includes ‘Category C’ for declarations based on the status of safety system functions associated with cold shutdown (Mode 5) or refueling/defueled. This EAL category represents the performance capabilities of malfunctioning systems with consideration given to RCS integrity, CNMT closure, and fuel clad integrity for the operating mode. Subcategory ‘1’ pertains to RCS level, as reactor vessel or RCS water level is directly related to the status of adequate core cooling and fuel clad integrity in cold operating modes.

‘CS1.3’ and ‘CG1.2’ are used to declare a SAE and a GE based on site conditions that include indications of RCS level being degraded to the point of core uncovery. These EALs use multiple, redundant indications for declaring core uncovery, including the associated sump or tank level, erratic source range monitor indication, and CNMT radiation levels as measured at the CHRRMs.

2.3.2 Current Emergency Plan Requirements

CS1.3 and CG1.2 use CNMT radiation levels as an indicator, with the threshold being CHRRM readings greater than 10,000 R/hr. From the EAL basis description, “In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed area radiation monitors (RM-1CR-3589SA or RM-1CR-3590SB). If these radiation monitors reach and exceed 10,000 R/hr, a loss of inventory with potential to uncover the core is likely to have occurred.” The cited

reference in EP-EAL for this threshold is AOP-31-BD, “Loss of Refueling Cavity Integrity- Basis Document.”

2.3.3 Reason for Proposed Change

AOP-031-BD states, “The dose rate to personnel on the edge of the Refueling Cavity is estimated to be as high as 10,000 R/hr with only six inches of one assembly extended above the water.” The CHRRMs do not measure radiation levels at the edge of the refueling cavity, as their location is some distance away within CNMT. Thus, it was determined that the threshold listed in the associated EALs does not have an appropriate technical basis to serve as prompt indication of core uncovery. Also, due to the CHRRMs location and distance from the core, the CHRRMs are not ideal for monitoring loss of water shielding from the reactor cavity. The proposed change will implement a more appropriate method and threshold for monitoring CNMT radiation levels for detecting a loss of refueling cavity level.

2.3.4 Description of Proposed Change

Duke Energy proposes changing the method for measuring CNMT radiation levels from a method utilizing the CHRRMs to a method more capable of monitoring CNMT radiation levels based on loss of refueling cavity level. Duke Energy proposes utilizing the CVI radiation monitors, as these monitors have improved sensitivity and the location of these monitors allows them to detect significant changes in the loss of water shielding from the reactor cavity, which can be used to detect conditions leading to core uncovery.

The threshold proposed is a CVI radiation monitor value greater than 2.6E+04 mR/hr; however, single calculated radiation monitor reading does not represent core uncovery for all event scenarios due to multiple variables that can affect the source term (such as number of fuel assemblies in the core, fuel burn up in each assembly, and time after shutdown). The proposed threshold of 2.6E+04 mR/hr for the CVI radiation monitors will be low enough to be a valid indication of fuel uncovery, but high enough so that inadvertent classifications are not caused by maintenance events or other conditions that may increase radiation levels without a coincident loss of reactor vessel water level.

> Current threshold for declaring fuel uncovery per Category C: “Containment radiation > 10,000 R/hr (RM-1CR-3589-SA or RM-1CR-3590-SB)”
>
> Future threshold for declaring fuel uncovery per Category C: “A Containment Ventilation Isolation Radiation Monitor > 2.6E+04 mR/hr (RM-1CR-3561A-SA, RM-1CR-3561B-SB, RM-1CR-3561C-SA, or RM-1CR-3561D-SB).”

The proposed EAL threshold is within the CVI radiation monitor's design operating range of 10^1 - 10^7 mR/hr (Reference 6.11). The proposed changes to the EAL threshold descriptions and bases are shown in Enclosure 2 (clean version of EP-EAL that incorporates changes) and Enclosure 3 (redline and strikeout version of EP-EAL changes). Enclosure 6 shows the proposed changes to the HNP EAL Wallcharts.

3.0 TECHNICAL EVALUATION

3.1 Hot Operating Mode Loss of RCS Threshold

The proposed method and threshold value for declaring a Loss of RCS (Alert) per HNP EP-EAL, Table F-1, Fission Product Barrier Matrix, Category C for CNMT Radiation / RCS Activity, was developed in accordance with Reference 6.3, Table 9-F-3, PWR EAL Fission Product Barrier Table, Thresholds for LOSS or POTENITIAL LOSS of Barriers, RCS Activity / CNMT Radiation, Loss 3.A.

The Table 9-F-3 entry for PWR RCS Barrier Thresholds, RCS Activity / CNMT Radiation, Loss 3.A states the following: “The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold 3.A since it indicates a loss of the RCS Barrier only.”

The CVI radiation monitors are already considered equipment important to Emergency Preparedness, as they are already used in the HNP EAL scheme for declaring initiating conditions related to abnormal radiation levels caused by an irradiated fuel event (Category R, Subcategory 2 per the HNP EAL scheme).

The proposed thresholds will provide an indication of an instantaneous release of all reactor coolant mass with reactor coolant activity equal to Technical Specification allowable limits (with no FC damage). Duke Energy performed a calculation for the containment radiation EAL threshold values in accordance with the guidance specified in Reference 6.3, which is provided in Enclosure 4 of this submittal.

<u>Source Term for Thresholds</u>
Duke Energy used NUREG-1940, Revision 0, “RASCAL 4: Description of Models and Methods,” December 2012 (Reference 6.6), to develop the source terms used for the analysis. Guidance contained in Reference 6.6 is considered representative of the wide spectrum of possible events that establish the planning basis of emergency preparedness and provides radiological consequence assessment methods that are acceptable to the NRC. Additionally, the source term used to develop the HNP effluent EAL thresholds and in the Unified RASCAL Interface/Radiological Assessment System for Consequence Analysis (URI/RASCAL) dose assessment model is from Reference 6.6. Thus, Reference 6.6 is consistent with NRC guidance and with other source term bases used within the HNP Emergency Preparedness Program.

Consistent with the graphs in Reference 6.6, the instantaneous release of the RCS to the containment is assumed to occur one hour after the damage event / reactor scram to account for damage progression, dispersion of activity and decay of the very short half-life isotopes.

Enclosure 4 lists the HNP calculation that equates CHRRM readings to CVI radiation monitor readings for RM-1CR-3561A-SA using the ratio of 12.2 R/hr CVI to 17.5 R/hr CHRRM or approximately 70%. RM-1CR-3561A-SA is located on the inner wall of containment at the end of the refueling cavity, and is furthest away from the centerline of the reactor core. RM-1CR-3561B-SB and RM-1CR-3561C-SA are located on the inboard side of the 'A' Steam Generator Missile shield, which is adjacent to the Reactor Refueling Cavity. RM-1CR-3561D-SB is located on the inboard side of the 'C' Steam Generator Missile shield which is adjacent to the Reactor Refueling Cavity. NUREG/BR-0150, "Response Technical Manual (RTM-96)," Volume 1,

Revision 4, Method A.4, "Evaluation of Containment Radiation," establishes a basis to use a radiation monitor for a core damage assessment as: "Confirm that the containment radiation monitor "sees" more than 50% of the shaded area shown in Fig. A-3 (PWR)." Based upon plant drawings, all four CVI radiation monitors satisfy the NUREG/BR-0150 Fig. A-3 (PWR) guidelines. All four CVI radiation monitors would be immersed within the same plume and will "see" a similar volume of the containment building; therefore, all four CVI radiation monitors are equivalent and can be used for EAL Classification.

Conclusion

Utilizing the CVI radiation monitors at the proposed thresholds for ‘Loss of RCS’ declaration is consistent with the intent and methodology prescribed by NEI 99-01, Revision 6 (Reference 6.3), Table 9-F-3, PWR EAL Fission Product Barrier Table, Thresholds for LOSS or POTENTIAL LOSS of Barriers. The thresholds were derived using source terms per Reference 6.6. This method provides the level of redundancy and diversity in the HNP EAL scheme, Table F-1, Fission Product Barrier Matrix, that is prescribed from the guidance of Reference 6.3.

3.2 Hot Operating Mode Loss of Fuel Clad and Containment Thresholds

The proposed values for declaration of Loss of FC (SAE) and Potential Loss of CNMT (GE) were developed in accordance with the methodology described in NEI 99-01, Revision 6 (Reference 6.3), Table 9-F-3, PWR EAL Fission Product Barrier Table, Thresholds for LOSS or POTENITIAL LOSS of Barriers.

Loss of Fuel Clad (SAE) Barrier

The new value for Loss of FC corresponds to the Reference 6.3, Table 9-F-3 entry for PWR Fuel Clad Barrier Thresholds, RCS Activity / CNMT Radiation, Loss 3.A. This table entry specifies the following:

> “The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.
>
> The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold 3.A since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the emergency classification level to a Site Area Emergency.”

Duke Energy performed a calculation for the containment radiation EAL threshold values in accordance with the guidance specified in Reference 6.3, which is provided in Enclosure 4 of this submittal. The FC FPB threshold value is based on an instantaneous release of reactor coolant into the CNMT at the percentage (%) of FC damage equivalent to 300 μCi/cubic centimeter (cc) dose equivalent I-131 RCS activity. Enclosure 4 identifies that 300μCi/cc is equivalent to 1.08% FC damage. This % FC damage value is multiplied as a ratio to the expected CNMT radiation reading for 100% fuel clad damage to establish the FC FPB threshold value in R/hr.

Potential Loss of CNMT Barrier

The new value for Potential Loss of CNMT corresponds to the Reference 6.3, Table 9-F-3 entry for PWR Containment Barrier Thresholds, RCS Activity / Containment Radiation, Potential Loss 3.A. This table entry specifies the following:

> "The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.
>
> NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment, which would then escalate the emergency classification level to a General Emergency….
>
> NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, provides the basis for using the 20% fuel cladding failure value. Unless there is a site-specific analysis justifying a different value, the reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with 20% fuel clad failure into the containment atmosphere."

Duke Energy performed a calculation for the containment radiation EAL threshold values in accordance with the guidance specified in Reference 6.3, which is provided in Enclosure 4 of this submittal. The CNMT FPB threshold value is based on an instantaneous release of reactor coolant into the CNMT at an equivalent of 20% FC damage. The 20% FC damage value is multiplied as a ratio to the expected CNMT radiation reading for 100% FC damage to establish the CNMT FBP threshold value in R/hr. Enclosure 4 identifies that 300μCi/cc is equivalent to approximately 1.08% FC damage. Therefore, the ratio between the radiation reading proposed for FC damage and the expected CNMT radiation reading for 20% FC damage is approximately 1.08/20.

Source Term for Thresholds

Duke Energy used Reference 6.6 to develop the source terms used for the analysis. Guidance contained in Reference 6.6 is considered representative of the wide spectrum of possible events that establish the planning basis of emergency preparedness and provides radiological consequence assessment methods that are acceptable to the NRC. Additionally, the source term used to develop the HNP effluent EAL thresholds and in the Unified RASCAL Interface/Radiological Assessment System for Consequence Analysis (URI/RASCAL) dose assessment model is from Reference 6.6. Thus, Reference 6.6 is consistent with NRC guidance and with other source term bases used within the HNP Emergency Preparedness Program.

Consistent with the graphs in Reference 6.6, the instantaneous release of the RCS to the containment is assumed to occur one hour after the damage event / reactor scram to account for damage progression, dispersion of activity and decay of the very short half-life isotopes.

Conclusion
The proposed values for declaration of 'Loss of FC' and 'Potential Loss of CNMT' developed in accordance with the methodology established in NEI 99-01, Revision 6 (Reference 6.3), Table 9-F-3, PWR EAL Fission Product Barrier Table, Thresholds for LOSS or POTENITIAL LOSS of Barriers, use source terms derived in accordance with Reference 6.6. This method is consistent with NRC-approved guidance for deriving the specified EAL threshold values. Thus, these threshold values are appropriate for use in classifying and declaring initiating conditions affecting the site's FPBs.

3.3 Cold Shutdown Loss of RCS Inventory Threshold

The current basis for declaring core uncovery during cold shutdown with the reactor vessel head removed utilizes the CHRRMs with a threshold of 10,000 R/hr. Duke Energy proposes replacing this method with a method that utilizes the CVI radiation monitors, using a reading of greater than 2.6E+04 mR/hr as the threshold. The CVI radiation monitors are positioned to be able to detect a loss of refueling cavity level. The use of the CVI radiation monitor value of 2.6E+04 mR/hr as the threshold will allow decision makers at HNP to distinguish normal CNMT radiation level fluctuations from a significant issue with the potential to lead to core uncovery.

The proposed threshold will provide an indication of core uncovery. Duke Energy performed a calculation for the proposed radiation monitor reading for core uncovery during refueling in accordance with the guidance specified in Reference 6.3, which is provided in Enclosure 5 of this submittal.

Source Term for Thresholds
Duke Energy used NUREG-1940, Revision 0, "RASCAL 4: Description of Models and Methods," December 2012 (Reference 6.6), to develop the source terms used for the analysis. Guidance contained in Reference 6.6 is considered representative of the wide spectrum of possible events that establish the planning basis of emergency preparedness and provides radiological consequence assessment methods that are acceptable to the NRC. Additionally, the source term used to develop the HNP effluent EAL thresholds and in the Unified RASCAL Interface/Radiological Assessment System for Consequence Analysis (URI/RASCAL) dose assessment model is from Reference 6.6. Thus, Reference 6.6 is consistent with NRC guidance and with other source term bases used within the HNP Emergency Preparedness Program.

As described in Enclosure 5, the CVI radiation monitors do not have a direct line of site to the core. Therefore, the threshold value is calculated by backscatter from the containment dome. Design inputs for containment geometry, radiation monitor detector location, source materials weight and geometry, source geometry and receptor geometry were used to calculate results for each of the CVI radiation monitors. Based upon monitor accuracy/readability, human factors, and the similarity of results between monitors, the CS1.3 and CG1.2 EAL thresholds are established at a value greater than 2.6E+04 mR/hr.

A single calculated radiation monitor reading does not represent core uncovery for all event scenarios due to multiple variables that can affect the source term (such as number of fuel assemblies in the core, fuel burn up in each assembly, and time after shutdown). The proposed threshold of a reading greater than 2.6E+04 mR/hr on the CVI radiation monitors will be low enough to be a valid indication of fuel uncovery, but high enough so that inadvertent classifications are not caused by maintenance events or other conditions that may increase radiation levels without a coincident loss of reactor vessel water level.

This proposed method for declaring core uncovery during cold shutdown is consistent with the guidance issued per Reference 6.3, Table C-1, Recognition Category “C” Initiating Condition Matrix, for declaring SAE and GE (CS1 and CG1). The EAL Developer Notes state, “As water level in the reactor vessel lowers, the dose rate above the core will increase. Enter a “site-specific radiation monitor” that could be used to detect core uncovery and the associated “site-specific value” indicative of core uncovery.” The Duke Energy proposed method will both utilize a site-specific radiation monitor and a site-specific value, and will allow the site to identify conditions indicative of core uncovery.

This method of determining elevated risk of core uncovery is consistent with current EAL methodology for declaring CS1.3 and CG1.2. Currently, the 10,000 R/hr is credited as indicative of “a loss of inventory with potential to uncover the core….” An alternate method for declaring CS1.3 and CG1.2 is “UNPLANNED increase in any C-1 sump or tank of sufficient magnitude to indicate core uncovery.”

Conclusion

Thus, the proposed method and threshold value will properly classify and declare conditions that indicate core uncovery during a refueling shutdown with the reactor vessel head removed. This change is consistent with the current EAL scheme and will improve the means to detect core uncovery during a refueling shutdown.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Requirements for the Emergency Plan’s EAL Scheme:
The regulations in 10 CFR 50.54(q) provide direction to licensees seeking to revise their Emergency Plan. The requirements related to nuclear power plant Emergency Plans are provided in the standards in 10 CFR 50.47, "Emergency Plans," and the requirements of Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR 50.

Paragraph (a)(1) of 10 CFR 50.47 states in part that, “no initial operating license for a nuclear power reactor will be issued unless a finding is made by the NRC that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.” Section 50.47 establishes standards that onsite and offsite emergency response plans must meet for the NRC staff to make such a positive finding. One of these standards, 10 CFR 50.47(b)(4), stipulates that Emergency Plans include a standard emergency classification and action level scheme.

10 CFR 50.47(b)(4) states, "A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures."

Table B17-1, “Conformance with QA Regulatory Guides and Industry Standards” of the Duke Energy Quality Assurance Program Document DUKE-QAPD-001 -A-, Amendment 43, confirms HNP will follow a format for emergency procedures in accordance with 10 CFR 50, Appendix E. 10 CFR 50 Appendix E, Section IV. Content of Emergency Plans, Item B, Assessment Actions states:

1. "The means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within and outside the site boundary to protect health and safety. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite and offsite monitoring. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite and offsite monitoring…"

2. "A licensee desiring to change its entire emergency action level scheme shall submit an application for an amendment to its license and receive NRC approval before implementing the change. Licensees shall follow the change process in 10 CFR 50.54(q) for all other emergency action level changes."

NRC Regulatory Guide 1.101, Revision 4, endorsed Nuclear Management and Resources Council, Inc./National Environmental Studies Project (NUMARC/NESP)-007, Revision 2, issued in January 1992, and NEI 99-01, Revision 4, EAL guidance as acceptable alternatives to the guidance provided in NUREG-0654 for development of EALs to comply with 10 CFR 50.47 and Appendix E to 10 CFR Part 50. NRC RIS 2005-02, Revision 1, also discusses that a change in an EAL scheme to incorporate the improvements provided in NUMARC/NESP-007 or NEI 99-01 would not decrease the overall effectiveness of the Emergency Plan. However, due to the potential safety significance of the change, the change needs prior NRC review and approval.

In a letter dated March 28, 2013 (Reference 6.4), the NRC staff concluded that the guidance contained in NEI 99-01, Revision 6, provides an acceptable method to develop an EAL scheme in accordance with the requirements of Appendix E to 10 CFR Part 50. In a letter dated April 13, 2016 (Reference 6.2), the NRC staff approved the HNP EAL scheme change to implement Reference 6.3 guidance, which is the basis for the current HNP EAL scheme.

As evaluated per Section 3.0 of this submittal, proposed changes to the HNP Emergency Plan's EAL scheme will restore or enhance compliance to the applicable regulations and guidance.

4.2 Precedent

Other operating plants have submitted LARs to address deficiencies in their EAL scheme. The following examples involve EAL changes that improved the licensee's ability to declare a condition based upon expected plant conditions as described in the NEI 99-01 guidance:

(1.) Vogtle Electric Generating Plant, Units 1 and 2: Southern Company determined that the range of the credited main steam radiation monitors in their Category R declaration scheme was insufficient to cover the entire range of the EAL thresholds and submitted a LAR that requested removing the affected monitors from their EAL scheme and relying on existing alternate means for making effluent declarations (Reference 6.7). The amendment was issued by the NRC in the Safety Evaluation (Reference 6.8) dated September 30, 2014.

(2.) Prairie Island Nuclear Generating Plant, Units 1 and 2: Xcel Energy determined that the effluent threshold established for an Alert classification level in their Category R declaration scheme was not appropriate – the emergency classification level exceeded the radiation monitor's indicated range. They also determined that for the loss of fuel clad barrier, the RCS

letdown line radiation monitor value used in their scheme was inconsistent with the intent of the loss criteria established in the NEI 99-01 guidance. A LAR was submitted to revise the radiation monitor value used for the effluent threshold and to remove the RCS letdown line monitor from the loss of fuel clad barrier EAL (Reference 6.9). The amendment was issued by the NRC in the Safety Evaluation (Reference 6.10) dated January 25, 2014.

4.3 Significant Hazards Consideration

In accordance with the provisions of 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), is proposing changes to the Shearon Harris Nuclear Power Plant, Unit 1 (HNP), Emergency Plan Emergency Action Level (EAL) scheme.

Duke Energy has evaluated whether or not a significant hazards consideration (SHC) is warranted with the proposed amendment by addressing the three criterion set forth in 10 CFR 50.92(c) as discussed below.

(1) *Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?*

The proposed changes affect the HNP Emergency Plan EAL scheme and do not alter any of the requirements of the Operating License or the Technical Specifications. The proposed changes do not reduce the effectiveness of the HNP Emergency Plan or the HNP Emergency Response Organization. The proposed changes do not modify any plant equipment and do not impact any failure modes that could lead to an accident. Additionally, the proposed changes do not impact the consequence of any analyzed accident since the changes do not affect any equipment related to accident mitigation. Based on this discussion, the proposed amendment does not increase the probability or consequences of an accident previously evaluated.

(2) *Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes affect the HNP Emergency Plan EAL scheme and do not alter any of the requirements of the Operating License or the Technical Specifications. These changes do not modify any plant equipment and there is no impact on the capability of the existing equipment to perform their intended functions. No new failure modes are introduced by the proposed changes. The proposed amendment does not introduce any accident initiator or malfunctions that would cause a new or different kind of accident. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) *Does the proposed amendment involve a significant reduction in a margin of safety?*

These changes affect the HNP Emergency Plan EAL scheme and do not alter any of the requirements of the Operating License or the Technical Specifications. The proposed changes do not affect any of the assumptions used in the accident analysis, nor do they affect any operability requirements for equipment important to plant safety. Therefore, the proposed changes will not result in a significant reduction in the margin of safety.

Based on the above, Duke Energy concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c). Accordingly, a finding of “no significant hazards consideration” is justified.

4.4 Conclusion

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 the proposed change in the HNP Emergency Plan, (2) operation of HNP will continue to 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

Duke Energy has determined that the proposed amendment would not change requirements with respect to use of a facility component located within the restricted area, as defined by 10 CFR 20, nor would it change inspection or surveillance requirements. Duke Energy has evaluated the proposed change and has determined that the change does not involve:

I. A Significant Hazards Consideration,
II. A significant change in the types or significant increase in the amounts of any effluent that may be released off site, or
III. A significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets 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 needs be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Letter from the NRC to Carolina Power & Light Company, Shearon Harris Nuclear Power Plant, Unit 1 - Changes to the Emergency Action Level Scheme (TAC No. ME1227), dated April 25, 2010 (ADAMS Accession No. ML100610685)

2. Letter from the NRC to Duke Energy, Shearon Harris Nuclear Power Plant, Unit 1 - Issuance of Amendment to Adopt Emergency Action Level Scheme Pursuant to NEI 99-01, Revision 6, “Development of Emergency Action Levels for Non-Passive Reactors" (CAC No. MF6196), dated April 13, 2016 (ADAMS Accession Number ML16057A838)

3. NEI 99-01, Revision 6, Development of Emergency Action Levels for Non-Passive Reactors, dated November 2012 (ADAMS Accession Number ML12326A805)

4. Letter from NRC to NEI, U.S. Nuclear Regulatory Commission Review and Endorsement of NEI 99-01, Revision 6, Dated November, 2012, dated March 28, 2013 (ADAMS Accession Number ML12346A463)

5. NRC and Federal Emergency Management Agency, NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," dated November 30, 1980 (ADAMS Accession No. ML040420012)

6. NRC, NUREG-1940, Revision 0, “RASCAL 4: Description of Models and Methods,” dated December 2012 (ADAMS Accession No. ML13031A448)

7. Letter from Southern Nuclear Operating Company, Inc. to NRC, License Amendment Request to Revise the Vogtle Electric Generating Plant Emergency Plan, dated August 20, 2013 (ADAMS Accession No. ML13233A112)

8. Letter from NRC to Southern Nuclear Operating Company, Inc., Vogtle Electric Generating Plant, Units 1 and 2 - Issuance of Amendments to the Emergency Plan (TAC Nos. MF2594 and MF2595), dated September 30, 2014 (ADAMS Accession No. ML14170A911)

9. Letter from Xcel Energy to the NRC, License Amendment Request (LAR) to Revise Emergency Plan (EP) Emergency Action Levels (EALs): RA1.2 and Fuel Clad Barrier Loss Criteria," dated December 13, 2012 (ADAMS Accession No. ML14170A911)

10. Letter from the NRC to Xcel Energy, Prairie Island Nuclear Generating Plant, Units 1 and 2 - Issuance of Amendments Re: Emergency Plan Changes (TAC Nos. MF0379 and MF0380), dated January 25, 2014 (ADAMS Accession No. ML13270A279)

11. HNP Final Safety Analysis Report Update (FSAR), Table 12.3.4-1, Area Radiation Monitors, Amendment 61

SERIAL HNP-18-004

ENCLOSURE 2

PROPOSED HARRIS NUCLEAR PLANT EMERGENCY ACTION LEVEL TECHNICAL BASES DOCUMENT CHANGES, EP-EAL (CLEAN)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.3 Site Area Emergency

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

AND

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

- UNPLANNED increase in **any** Table C-1 sump or tank of sufficient magnitude to indicate core uncovery
- A Containment Ventilation Isolation Radiation Monitor > 2.6E+04 mR/hr (RM-1CR-3561A-SA, RM-1CR-3561B-SB, RM-1CR-3561C-SA, or RM-1CR-3561D-SB)
- Erratic source range monitor indication

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

Table C-1 Sumps / Tanks
• Containment sumps
• PRT
• RCDT
• CCW surge tank
• RAB sumps
• RWST
• RMWST
• Recycle Holdup Tank

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

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

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

Basis:

In this EAL, all RCS water level indication would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications. Sump level increases must be evaluated against other potential sources of leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Sumps and tanks where RCS leakage may accumulate are listed in listed in Table C-1(ref. 1, 2).

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this loss of water shielding from the reactor cavity will result in indications on the Containment Ventilation Isolation (CVI) area radiation monitors. If these radiation monitors reach and exceed 2.6E+04 mR/hr, with RCS water level indication unavailable for greater than 30 minutes, then a loss of inventory with potential to uncover the core is likely to have occurred.

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

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

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

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

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

EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading.

A single calculated radiation monitor reading does not represent core uncovery for all event scenarios due to multiple variables that can affect the source term (i.e., number of fuel assemblies in the core, fuel burn up in each assembly, and time after shutdown). Therefore, consideration has been given so that the threshold value of > 2.6E+04 mR/hr on the CVI radiation monitors will be low enough to be a valid indication of fuel uncovery, but high enough so that inadvertent classifications are not caused by maintenance events or other conditions that may increase radiation levels without a coincident loss of reactor vessel water level.

HNP utilizes CVI area radiation monitors RM-1CR-3561A-SA, 1RM-1CR-3561B-SB, RM-1CR-3561C-SA, and RM-1CR-3561D-SB for evaluating conditions in containment. The sensitivity and location of these monitors allows them to detect significant changes in the loss of water shielding from the reactor cavity, which can be used to detect core uncovery.

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

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

HNP Basis Reference(s):

1. GP-001, Reactor Coolant System Fill and Vent Mode 5
2. GP-008, Draining the Reactor Coolant System
3. GP-009, Refueling Cavity Fill, Refueling and Drain of the Refueling Cavity Modes 5-6-5
4. CSD-EP-HNP-0101-06, Radiation Monitor Readings for Core Uncovery during Refueling
5. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CG1.2 General Emergency

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

AND

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

- UNPLANNED increase in **any** Table C-1 sump or tank of sufficient magnitude to indicate core uncovery
- A Containment Ventilation Isolation Radiation Monitor > 2.6E+04 mR/hr (RM-1CR-3561A-SA, RM-1CR-3561B-SB, RM-1CR-3561C-SA, or RM-1CR-3561D-SB)
- Erratic source range monitor indication

AND

Any Containment Challenge indication, Table C-2

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

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

Table C-1 Sumps / Tanks
• Containment sumps • PRT • RCDT • CCW surge tank • RAB sumps • RWST • RMWST • Recycle Holdup Tank

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

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions. As applied to HNP, Containment Closure is established when containment penetration closure is established in accordance with Technical Specifications 3/4.9.4.

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 this EAL, all RCS water level indication would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications. Sump level increases must be evaluated against other potential sources of leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Sumps and tanks where RCS leakage may accumulate are listed in listed in Table C-1 (ref. 1, 2).

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to core uncover will result in indications on the Containment Ventilation Isolation (CVI) area radiation monitors. If these radiation monitors reach and exceed 2.6E+04 mR/hr, with RCS water level indication unavailable for greater than 30 minutes, then a loss of inventory with potential to uncover the core is likely to have occurred.

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

Three 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 an explosive mixture exists. An explosive mixture can be formed when hydrogen gas concentration in the containment atmosphere is greater than 4% by volume in the presence of oxygen.
- Any unplanned increase in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of containment closure capability. Unplanned containment pressure increases indicates containment closure cannot be assured and the containment cannot be relied upon as a barrier to fission product release.

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

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

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

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

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

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

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

EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading.

A single calculated radiation monitor reading does not represent core uncovery for all event scenarios due to multiple variables that can affect the source term (i.e., number of fuel assemblies in the core, fuel burn up in each assembly, and time after shutdown). Therefore, consideration has been given so that the threshold value of > 2.6E+04 mR/hr on the CVI radiation monitors will be low enough to be a valid indication of fuel uncovery, but high enough so that inadvertent classifications are not caused by maintenance events or other conditions that may increase radiation levels without a coincident loss of reactor vessel water level.

HNP utilizes CVI area radiation monitors RM-1CR-3561A-SA, 1RM-1CR-3561B-SB, RM-1CR-3561C-SA, and RM-1CR-3561D-SB for evaluating conditions in containment. The sensitivity and location of these monitors allows them to detect significant changes in the loss of water shielding from the reactor cavity, which can be used to detect core uncovery.

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*.

HNP Basis Reference(s):

1. GP-001, Reactor Coolant System Fill and Vent Mode 5
2. GP-008, Draining the Reactor Coolant System
3. GP-009, Refueling Cavity Fill, Refueling and Drain of the Refueling Cavity Modes 5-6-5
4. CSD-EP-HNP-0101-06, Radiation Monitor Readings for Core Uncovery during Refueling
5. NEI 99-01 CG1

Table F-1 Fission Product Barrier Threshold Matrix						
	Fuel Clad (FC) Barrier		**Reactor Coolant System (RCS) Barrier**		**Containment (CNMT) Barrier**	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A **RCS or SG Tube Leakage**	None	None	1. An automatic or manual ECCS (SI) actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	1. CSFST Integrity-RED Path entry conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B **Inadequate Heat Removal**	1. CSFST Core Cooling-RED Path entry conditions met	1. CSFST Core Cooling-ORANGE PATH entry conditions met 2. CSFST Heat Sink-RED Path entry conditions met **AND** Heat sink is required	None	1. CSFST Heat Sink-RED Path entry conditions met **AND** Heat sink is required	None	1. CSFST Core Cooling-RED Path entry conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C **CNMT Radiation / RCS Activity**	1. (RM-1CR-3589SA or RM-1CR-3590SB) > Table F-2 Column FC Barrier Loss 2. Dose equivalent I-131 coolant activity > 300 µCi/gm	None	1. (RM-1CR-3561A-SA, RM-1CR-3561B-SB, RM-1CR-3561C-SA, or RM-1CR-3561D-SB) > Table F-2 Column RCS Barrier Loss	None	None	1. (RM-1CR-3589SA or RM-1CR-3590SB) > Table F-2 Column CNMT Potential Loss
D **CNMT Integrity or Bypass**	None	None	None	None	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on Emergency Coordinator judgment • UNISOLABLE pathway from Containment to the environment exists 2. Indications of RCS leakage outside of containment	1. CSFST Containment-RED Path entry conditions met 2. Containment hydrogen concentration > 4% 3. Containment pressure > 10 psig with < one full train of depressurization equipment operating (one CNMT spray pump and two CNMT fan coolers) per design for ≥ 15 min. (Note 1)
E **EC Judgment**	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the fuel clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the fuel clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the containment barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the containment barrier

Table F-2 Containment Radiation			
Time After S/D (Hours)	**FC Barrier Loss R/hr**	**RCS Barrier Loss mR/hr**	**CNMT Potential Loss R/hr**
0 - 1	130	1.37E+03	2360
1 - 2	110	1.12E+03	2000
2 - 8	70	6.35E+02	1300
> 8	21	1.37E+02	390

Barrier: Fuel Clad

Category: C. CNMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. RM-1CR-3589SA or RM-1CR-3590SB > Table F-2, Column FC Barrier Loss

Table F-2 Containment Radiation			
Time After S/D (Hours)	FC Barrier Loss R/hr	RCS Barrier Loss mR/hr	CNMT Potential Loss R/hr
0 - 1	130	1.37E+03	2360
1 - 2	110	1.12E+03	2000
2 - 8	70	6.35E+02	1300
> 8	21	1.37E+02	390

Definition(s):

None

Basis:

Containment radiation monitor readings greater than Table F-2, FC Barrier Loss indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the Containment. The radiation monitor reading is derived assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/cc dose equivalent I-131 into the Containment atmosphere (ref. 1).

Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage.

RM-1CR-3589SA and RM-1CR-3590SB are the Containment High Range Monitors that provide indication of radiation levels in Containment during and after postulated accidents (ref. 2).

Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold C.1 since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the ECL to a Site Area Emergency.

HNP Basis Reference(s):

1. EP-EALCALC-HNP-1701, Containment Radiation EAL Threshold Values
2. DBD-304, Radiation Monitoring System and Gross Failed Fuel Monitor
3. NEI 99-01 CNMT Radiation / RCS Activity Fuel Clad Loss 3.A

Barrier: Reactor Coolant System

Category: C. CNMT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

1. A Containment Ventilation Isolation Radiation Monitor (RM-1CR-3561A-SA, RM-1CR-3561B-SB, RM-1CR-3561C-SA, or RM-1CR-3561D-SB) > Table F-2, Column RCS Barrier Loss

Table F-2 Containment Radiation			
Time After S/D (Hours)	FC Barrier Loss R/hr	RCS Barrier Loss mR/hr	CNMT Potential Loss R/hr
0 - 1	130	1.37E+03	2360
1 - 2	110	1.12E+03	2000
2 - 8	70	6.35E+02	1300
> 8	21	1.37E+02	390

Definition(s):

None

Basis:

A Containment Ventilation Isolation radiation monitor reading greater than Table F-2, RCS Barrier Loss (ref. 1) indicates the release of reactor coolant into containment. The readings assume the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the Containment atmosphere. Because of the very high fuel clad integrity, only small amounts of noble gases would be dissolved in the primary coolant.

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold C.1 since it indicates a loss of the RCS Barrier only.

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

HNP Basis Reference(s):

1. EP-EALCALC-HNP-1701, Containment Radiation EAL Threshold Values
2. DBD-304, Radiation Monitoring System and Gross Failed Fuel Monitor
3. NEI 99-01 CNMT Radiation / RCS Activity RCS Loss 3.A

Barrier: Containment

Category: C. CNMT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

1. RM-1CR-3589SA or RM-1CR-3590SB > Table F-2, Column CNMT Potential Loss

Table F-2 Containment Radiation			
Time After S/D (Hours)	FC Barrier Loss R/hr	RCS Barrier Loss mR/hr	CNMT Potential Loss R/hr
0 - 1	130	1.37E+03	2360
1 - 2	110	1.12E+03	2000
2 - 8	70	6.35E+02	1300
> 8	21	1.37E+02	390

Definition(s):

None

Basis:

Containment radiation monitor readings on RM-1CR-3589SA or RM-1CR-3590SB > Table F-2 column CNMT Potential Loss indicate significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier (ref. 1).

The readings are higher than that specified for Fuel Clad Loss C.1 and RCS Loss C.1. Containment radiation readings at or above the Containment barrier Potential Loss threshold, therefore, signify a loss of two fission product barriers and Potential Loss of the third, indicating the need to upgrade the emergency classification to a General Emergency.

RM-1CR-3589SA and RM-1CR-3590SB are the Containment High Range Monitors that provide indication of radiation levels in Containment during and after postulated accidents (ref. 2).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the ECL to a General Emergency.

HNP Basis Reference(s):

1. EP-EALCALC-HNP-1701, Containment Radiation EAL Threshold Values
2. DBD-304, Radiation Monitoring System and Gross Failed Fuel Monitor
3. NEI 99-01 CNMT Radiation / RCS Activity Containment Potential Loss 3.A

SERIAL HNP-18-004

ENCLOSURE 3

PROPOSED HARRIS NUCLEAR PLANT EMERGENCY ACTION LEVEL TECHNICAL BASES DOCUMENT CHANGES, EP-EAL (MARKUP)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.3 Site Area Emergency

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

AND

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

- UNPLANNED increase in **any** Table C-1 sump or tank of sufficient magnitude to indicate core uncovery
- ~~Containment radiation > 10,000 R/hr (RM-1CR-3589-SA or RM-1CR-3590-SB)~~
- Erratic source range monitor indication

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

Replace with:
"A Containment Ventilation Isolation Radiation Monitor > 2.6E+04 mR/hr (RM-1CR-3561A-SA, RM-1CR-3561B-SB, RM-1CR-3561C-SA, or RM-1CR-3561D-SB)"

Table C-1 Sumps / Tanks
• Containment sumps • PRT • RCDT • CCW surge tank • RAB sumps • RWST • RMWST • Recycle Holdup Tank

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

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

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

Basis:

In this EAL, all RCS water level indication would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications. Sump level increases must be evaluated against other potential sources of leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Sumps and tanks where RCS leakage may accumulate are listed in listed in Table C-1(ref. 1, 2).

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. ~~The dose rate due to this core shine should result in indications on installed area radiation monitors (RM-1CR-3589 SA or RM-1CR-3590 SB). If these radiation monitors reach and exceed 10,000 R/hr, a loss of inventory with potential to uncover the core is likely to have occurred (ref. 4).~~

Replace with INSERT A

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

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

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

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

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

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

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

Add INSERT B

HNP Basis Reference(s):

1. GP-001, Reactor Coolant System Fill and Vent Mode 5
2. GP-008, Draining the Reactor Coolant System
3. GP-009, Refueling Cavity Fill, Refueling and Drain of the Refueling Cavity Modes 5-6-5
4. ~~AOP-031-BD, Loss of Refueling Cavity Integrity- Basis Document~~
5. NEI 99-01 CS1

Replace with: "CSD-EP-HNP-0101-06, Radiation Monitor Readings for Core Uncovery during Refueling"

INSERT A:

The dose rate due to this loss of water shielding from the reactor cavity will result in indications on the Containment Ventilation Isolation (CVI) area radiation monitors. If these radiation monitors reach and exceed 2.6E+04 mR/hr, with RCS water level indication unavailable for greater than 30 minutes, then a loss of inventory with potential to uncover the core is likely to have occurred.

INSERT B:

EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. A single calculated radiation monitor reading does not represent core uncovery for all event scenarios due to multiple variables that can affect the source term (i.e., number of fuel assemblies in the core, fuel burn up in each assembly, and time after shutdown). Therefore, consideration has been given so that the threshold value of > 2.6E+04 mR/hr on the CVI radiation monitors will be low enough to be a valid indication of fuel uncovery, but high enough so that inadvertent classifications are not caused by maintenance events or other conditions that may increase radiation levels without a coincident loss of reactor vessel water level. HNP utilizes CVI area radiation monitors RM-1CR-3561A-SA, RM-1CR-3561B-SB, RM-1CR-3561C-SA, and RM-1CR-3561D-SB for evaluating conditions in containment. The sensitivity and location of these monitors allows them to detect significant changes in the loss of water shielding from the reactor cavity, which can be used to detect core uncovery.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CG1.2 General Emergency

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

AND

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

- UNPLANNED increase in **any** Table C-1 sump or tank of sufficient magnitude to indicate core uncovery
- ~~Containment radiation > 10,000 R/hr (RM-1CR-3589-SA or RM-1CR-3590-SB)~~
- Erratic source range monitor indication

AND

Any Containment Challenge indication, Table C-2

Replace with:
"A Containment Ventilation Isolation Radiation Monitor > 2.6E+04 mR/hr (RM-1CR-3561A-SA, RM-1CR-3561B-SB, RM-1CR-3561C-SA, or RM-1CR-3561D-SB)"

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

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

Table C-1 Sumps / Tanks
• Containment sumps
• PRT
• RCDT
• CCW surge tank
• RAB sumps
• RWST
• RMWST
• Recycle Holdup Tank

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

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined actions taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions. As applied to HNP, Containment Closure is established when containment penetration closure is established in accordance with Technical Specifications 3/4.9.4.

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 this EAL, all RCS water level indication would be unavailable for greater than 30 minutes, and the RCS inventory loss must be detected by indirect leakage indications. Sump level increases must be evaluated against other potential sources of leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Sumps and tanks where RCS leakage may accumulate are listed in listed in Table C-1 (ref. 1, 2).

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. ~~The dose rate due to this core shine should result in indications on installed area radiation monitors (RM-1CR-3589-SA or RM-1CR-3590-SB). If these radiation monitors reach and exceed 10,000 R/hr, a loss of inventory with potential to uncover the core is likely to have occurred (ref. 4).~~ Add INSERT C

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

Three 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 an explosive mixture exists. An explosive mixture can be formed when hydrogen gas concentration in the containment atmosphere is greater than 4% by volume in the presence of oxygen.
- Any unplanned increase in containment pressure in the Cold Shutdown or Refueling mode indicates a potential loss of containment closure capability. Unplanned containment pressure increases indicates containment closure cannot be assured and the containment cannot be relied upon as a barrier to fission product release.

INSERT C:

The dose rate due to core uncover will result in indications on the Containment Ventilation Isolation (CVI) area radiation monitors. If these radiation monitors reach and exceed 2.6E+04 mR/hr, with RCS water level indication unavailable for greater than 30 minutes, then a loss of inventory with potential to uncover the core is likely to have occurred.

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

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

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

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

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

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

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

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

Add INSERT D

HNP Basis Reference(s):

1. GP-001, Reactor Coolant System Fill and Vent Mode 5
2. GP-008, Draining the Reactor Coolant System
3. GP-009, Refueling Cavity Fill, Refueling and Drain of the Refueling Cavity Modes 5-6-5
4. ~~AOP-031-BD, Loss of Refueling Cavity Integrity- Basis Document~~
5. NEI 99-01 CG1

Replace with: "CSD-EP-HNP-0101-06, Radiation Monitor Readings for Core Uncovery during Refueling"

INSERT D:

EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading.

A single calculated radiation monitor reading does not represent core uncovery for all event scenarios due to multiple variables that can affect the source term (i.e., number of fuel assemblies in the core, fuel burn up in each assembly, and time after shutdown). Therefore, consideration has been given so that the threshold value of > 2.6E+04 mR/hr on the CVI radiation monitors will be low enough to be a valid indication of fuel uncovery, but high enough so that inadvertent classifications are not caused by maintenance events or other conditions that may increase radiation levels without a coincident loss of reactor vessel water level.

HNP utilizes CVI area radiation monitors RM-1CR-3561A-SA, 1RM-1CR-3561B-SB, RM-1CR-3561C-SA, and RM-1CR-3561D-SB for evaluating conditions in containment. The sensitivity and location of these monitors allows them to detect significant changes in the loss of water shielding from the reactor cavity, which can be used to detect core uncovery.

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad (FC) Barrier		Reactor Coolant System (RCS) Barrier		Containment (CNMT) Barrier	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A **RCS or SG Tube Leakage**	None	None	1. An automatic or manual ECCS (SI) actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	1. Operation of a standby charging pump is required by **EITHER:** • UNISOLABLE RCS leakage • SG tube leakage 2. CSFST Integrity-RED Path entry conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B **Inadequate Heat Removal**	1. CSFST Core Cooling-RED Path entry conditions met	1. CSFST Core Cooling-ORANGE PATH entry conditions met 2. CSFST Heat Sink-RED Path entry conditions met **AND** Heat sink is required	None	1. CSFST Heat Sink-RED Path entry conditions met **AND** Heat sink is required	None	1. CSFST Core Cooling-RED Path entry conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C **CNMT Radiation / RCS Activity**	~~1. Containment radiation >150 R/hr (RM-1CR-3589-SA or RM-1CR-3590-SB)~~ 2. Dose equivalent I-131 coolant activity > 300 µCi/gm	None	~~1. Containment Leak Detection Monitor Noble Gas (REM-1LT-3502A-SA) > 8.3E-3 µCi/ml~~	None	None	~~1. Containment radiation >600 R/hr (RM-1CR-3589-SA or RM-1CR-3590-SB)~~
D **CNMT Integrity or Bypass**					1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on Emergency Coordinator judgment • UNISOLABLE pathway from Containment to the environment exists 2. Indications of RCS leakage outside of containment	1. CSFST Containment-RED Path entry conditions met 2. Containment hydrogen concentration > 4% 3. Containment pressure > 10 psig with < one full train of depressurization equipment operating (one CNMT spray pump and two CNMT fan coolers) per design for ≥ 15 min. (Note 1)
E **EC Judgment**	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the fuel clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the fuel clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the containment barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the containment barrier

Revise to
"1. (RM-1CR-3589SA or
RM-1CR-3590SB) > Table F-2
Column CNMT Potential Loss"

Revise to
"1. (RM-1CR-3589SA or
RM-1CR-3590SB) > Table F-2
Column FC Barrier Loss"

Revise to
1. " (RM-1CR-3561A-SA,
RM-1CR-3561B-SB,
RM-1CR-3561C-SA, or
RM-1CR-3561D-SB) > Table F-2
Column RCS Barrier Loss"

Add INSERT H

INSERT H:

Table F-2 Containment Radiation			
Time After S/D (Hours)	FC Barrier Loss R/hr	RCS Barrier Loss mR/hr	CNMT Potential Loss R/hr
0 - 1	130	1.37E+03	2360
1 - 2	110	1.12E+03	2000
2 - 8	70	6.35E+02	1300
> 8	21	1.37E+02	390

Barrier: Fuel Clad

Category: C. CNMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

Replace with 'INSERT E' from the following page:

~~1. Containment radiation >150 R/hr (RM-1CR-3589-SA or RM-1CR-3590-SB)~~

~~**Definition(s):**~~

~~None~~

~~**Basis:**~~

~~Containment radiation monitor readings greater than 150.3 R/hr, rounded to 150 R/hr for readability, indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the Containment. The reading is derived assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/cc dose equivalent I-131 into the Containment atmosphere.~~

Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage ~~(approximately 5% clad failure depending on core inventory and RCS volume). (ref. 1)~~

RM-1CR-3589SA and RM-1CR-3590SB

~~RM-1CR-3589-SA and RM-1CR-3590-SB~~ are the Containment High Range Monitors that provide indication of radiation levels in Containment during and after postulated accidents. ~~The Alert alarms are set at 6.5 R/hr and the High alarms are set at 17.5 R/hr.~~ (ref. 2~~, 3~~).

~~The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage.~~ Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold C.1 since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the ECL to a Site Area Emergency.

HNP Basis Reference(s):

~~1. Calculation 3-B-12-022, DHRAM- Response to a Fuel and RCS Breach~~

2. DBD-304, Radiation Monitoring System and Gross Failed Fuel Monitor

~~3. HP-500, Radiation Monitoring System Data Base Manual~~

3. ~~4.~~ NEI 99-01 CNMT Radiation / RCS Activity Fuel Clad Loss 3.A

Revise:
"EP-EALCALC-HNP-1701, Containment Radiation EAL Threshold Values"

INSERT E:

1. RM-1CR-3589SA or RM-1CR-3590SB > Table F-2, Column FC Barrier Loss

Table F-2 Containment Radiation			
Time After S/D (Hours)	FC Barrier Loss R/hr	RCS Barrier Loss mR/hr	CNMT Potential Loss R/hr
0 - 1	130	1.37E+03	2360
1 - 2	110	1.12E+03	2000
2 - 8	70	6.35E+02	1300
> 8	21	1.37E+02	390

Definition(s):

None

Basis:

Containment radiation monitor readings greater than Table F-2, FC Barrier Loss indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the Containment. The radiation monitor reading is derived assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 µCi/cc dose equivalent I-131 into the Containment atmosphere (ref. 1).

Barrier: Reactor Coolant System
Category: C. CNMT Radiation/ RCS Activity
Degradation Threat: Loss
Threshold:

Replace with 'INSERT F' from the following page:

1. ~~Containment Leak Detection Monitor Noble Gas (REM-1LT-3502A-SA) > 8.3E-3 µCi/ml~~

~~**Definition(s):**~~

~~None~~

~~**Basis:**~~

~~Containment radiation monitor readings on REM-1LT-3502A-SA noble gas channel greater than 8.3E-3 µCi/ml (ref. 1) indicate the release of reactor coolant to the Containment.~~ The readings assume the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the Containment atmosphere. Because of the very high fuel clad integrity, only small amounts of noble gases would be dissolved in the primary coolant.

~~The Containment High Range Monitors (RM-1CR-3589-SA or RM-1CR-3590-SB) are bugged to read at least 1 R/hr and are not capable of detecting this radiation level (ref. 2, 3)~~.

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold C.1 since it indicates a loss of the RCS Barrier only.

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

HNP Basis Reference(s):

1. ~~Calculation HNP-M/MECH-1074, Alternate Source Term Effect on REM-3502A Response to RCS Breach with Non-Failed Fuel~~
2. DBD-304, Radiation Monitoring System and Gross Failed Fuel Monitor
3. ~~HP-500, Radiation Monitoring System Data Base Manual~~

3. ~~4.~~ NEI 99-01 CNMT Radiation / RCS Activity RCS Loss 3.A

Revise:
"EP-EALCALC-HNP-1701, Containment Radiation EAL Threshold Values"

INSERT F:

1. A Containment Ventilation Isolation Radiation Monitor (RM-1CR-3561A-SA, RM-1CR-3561B-SB, RM-1CR-3561C-SA, or RM-1CR-3561D-SB) > Table F-2, Column RCS Barrier Loss

Table F-2 Containment Radiation			
Time After S/D (Hours)	FC Barrier Loss R/hr	RCS Barrier Loss mR/hr	CNMT Potential Loss R/hr
0 - 1	130	1.37E+03	2360
1 - 2	110	1.12E+03	2000
2 - 8	70	6.35E+02	1300
> 8	21	1.37E+02	390

Definition(s):

None

Basis:

A Containment Ventilation Isolation radiation monitor reading greater than Table F-2, RCS Barrier Loss (ref. 1) indicates the release of reactor coolant into containment.

Barrier: Containment

Category: C. CNMT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

Replace with 'INSERT G' from the following page:

~~1. Containment radiation > 600 R/hr (RM-1CR-3589-SA or RM-1CR-3590-SB)~~

~~**Definition(s):**~~

~~None~~

~~**Basis:**~~

~~Containment radiation monitor readings greater than 601.2 R/hr, rounded to 600 R/hr for readability, (ref. 1) indicate significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier.~~

The readings are higher than that specified for Fuel Clad Loss C.1 and RCS Loss C.1. Containment radiation readings at or above the Containment barrier Potential Loss threshold, therefore, signify a loss of two fission product barriers and Potential Loss of the third, indicating the need to upgrade the emergency classification to a General Emergency.

~~RM-1CR-3589-SA and RM-1CR-3590-SB~~ RM-1CR-3589SA and RM-1CR-3590SB are the Containment High Range Monitors that provide indication of radiation levels in Containment during and after postulated accidents. ~~The Alert alarms are set at 6.5 R/hr and the High alarms are set at 17.5 R/hr.~~ (ref. 2, ~~3~~).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the ECL to a General Emergency.

HNP Basis Reference(s):

~~1. Calculation 3-B-12-022 DHRAM, Response to a Fuel and RCS Breach~~

2. DBD-304, Radiation Monitoring System and Gross Failed Fuel Monitor

~~3. HP-500, Radiation Monitoring System Data Base Manual~~

3. ~~4.~~ NEI 99-01 CNMT Radiation / RCS Activity Containment Potential Loss 3.A

Revise:
"EP-EALCALC-HNP-1701, Containment Radiation EAL Threshold Values"

INSERT G:

1. RM-1CR-3589SA or RM-1CR-3590SB > Table F-2, Column CNMT Potential Loss

Table F-2 Containment Radiation			
Time After S/D (Hours)	FC Barrier Loss R/hr	RCS Barrier Loss mR/hr	CNMT Potential Loss R/hr
0 - 1	130	1.37E+03	2360
1 - 2	110	1.12E+03	2000
2 - 8	70	6.35E+02	1300
> 8	21	1.37E+02	390

Definition(s):

None

Basis:

Containment radiation monitor readings on RM-1CR-3589SA or RM-1CR-3590SB > Table F-2 column CNMT Potential Loss indicate significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier (ref. 1).

SERIAL HNP-18-004

ENCLOSURE 4

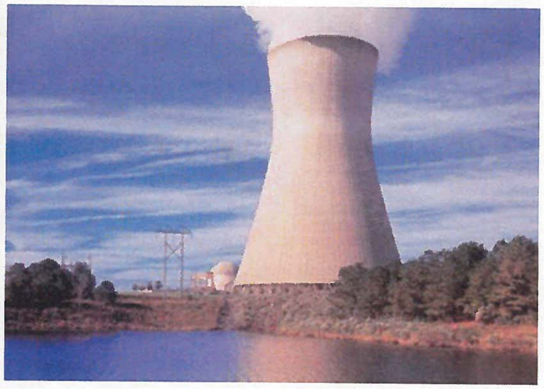
CALCULATION FOR CONTAINMENT RADIATION EAL THRESHOLD VALUES

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

DUKE ENERGY®



Harris Nuclear Plant (HNP)

Containment Radiation EAL Threshold Values

EP-EALCALC-HNP-1701
Revision 2

Revision 2 Author: Caryl D. Ingram

Technical Reviewer: William P. Cerame

EPM Reviewer: David A. Thompson CHP

Revision 2 Author: 7/2/2018

Technical Reviewer: 7/2/2018

EP Reviewer: 7/2/2018

List of Affected Pages

Calculation Number:	EP-EALCALC-HNP-1701	
Revision Number:	2	

Body of Calculation (including appendices)				Supporting Documents				
Rev. #	*Pages Revised*	*Pages Deleted*	*Pages Added*	*Rev. #*	*Type*	*Pages Revised*	*Pages Deleted*	*Pages Added*
0			1 - 15					
1		1 - 15	1 - 21					
2		1 - 21	1 - 25					

Revision Summary Form

Calculation Number:	**EP-EALCALC-HNP-1701**
Revision Number:	**2**

Revision Summary

Revision	Summary
0	Original Issue.
1	Provided interpolation of Containment High Range Radiation Monitor threshold values between 1 hr and 24 hrs.
2	Provided EAL Table F-2 RCS Barrier Loss containment radiation monitor threshold values using the Containment Ventilation Isolation monitors (CVIs) in lieu of the Containment High Range Radiation Monitor's minimum instrument range value of 5 R/hr.

Table of Contents

1. PURPOSE

The Shearon Harris Nuclear Plant (HNP) 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 FPB Containment High Range Radiation Monitor (CHRRM) readings developed in accordance with the guidance in NEI 99-01 Revision 6.

Documentation of the assumptions, calculations and results are provided for the values associated with the NEI 99-01 Revision 6 Table 9-F-3, PWR EAL Fission Product Barrier Table, thresholds listed below.

- NEI Fuel Clad Loss 3.A
- NEI Reactor Coolant Loss 3.A
- NEI Containment Potential Loss 3.A

2. DEVELOPMENT METHODS AND BASES

2.1. Fuel Clad Loss

Guidance Criteria

Per NEI 99-01 Revision 6, this radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals 300 μCi/gm dose equivalent I-131. 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 since it indicates a loss of both the fuel clad barrier and the RCS barrier.

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 μCi/gm dose equivalent I-131, into the primary containment atmosphere.

HNP Basis

The fuel clad FPB threshold value is based on an instantaneous release of reactor coolant into the containment at that percent fuel clad damage equivalent to 300 μCi/cc DEI-131 RCS activity. That percent fuel clad damage value is ratioed to a containment radiation reading for 100% fuel clad damage to establish the fuel clad FBP threshold value in R/hr.

2.2. Reactor Coolant System Loss

Guidance Criteria

Per NEI 99-01 Revision 6, this radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for fuel clad barrier loss threshold since it indicates a loss of the RCS barrier only.

The reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory, with RCS activity at Technical Specification allowable limits, into the primary containment atmosphere. RCS activity at this level will typically result in primary containment radiation levels that can be more readily detected by primary containment radiation monitors, and more readily differentiated from those caused by piping or component “shine” sources. If desired, a plant may use a lesser value of RCS activity for determining this value.

In some cases, the site-specific physical location and sensitivity of the containment radiation monitor(s) may be such that radiation from a cloud of released RCS gases cannot be distinguished from radiation emanating from piping and components containing elevated reactor coolant activity. If so, determine if an alternate indication is available.

HNP Basis

The HNP higher technical specification stabilized value for DEI-131 is 60 μCi/g. This activity would yield a containment radiation monitor reading approximately 5x lower than the fuel clad loss fission product barrier containment radiation reading equivalent to 300 μCi/g. NUREG-1940 Figure 1-1 provides estimates for standard plant containment radiation based on spiked RCS activity, which is slightly less than half the value obtained by the 300 μCi/g to 60 μCi/g DEI-131 ratio.

NUREG-1940 Figure 1-1 models a spiked RCS activity that is lower than the RCS activity equivalent to 60 μCi/g DEI-131 described above (the NUREG-1940 graph is based on a release into containment of 100 times the non-noble gas fission products normally found in the coolant). This is the preferred value for the RCS loss threshold as it provides for a containment monitor escalation of approximately one decade between fission product barrier thresholds at the 1 hour point. NEI 99-01 guidance criteria allows the use of a lesser value for RCS activity (see guidance criteria section above).

The HNP RCS FPB threshold value is based on NUREG-1940 standard plant containment radiation readings for an instantaneous release of spiked reactor coolant, which is lower than 60 μCi/g DEI-131, and is adjusted for the site specific power rating.

2.3. Containment Potential Loss

Guidance Criteria

Per NEI 99-01 Revision 6, this radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous fuel clad and RCS barrier loss thresholds.

NUREG-1228 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 and fuel clad barriers. 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.

NUREG-1228 provides the basis for using the 20% fuel cladding failure value. Unless there is a site-specific analysis justifying a different value, the reading should be determined assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with 20% fuel clad failure into the primary containment atmosphere.

HNP Basis

The containment FPB threshold value is based on an instantaneous release of reactor coolant into the containment at an equivalent of 20% fuel clad damage. The 20% fuel clad damage value is ratioed to a containment radiation reading for 100% fuel clad damage to establish the containment FBP threshold value in R/hr.

2.4. Source Term

Guidance Criteria

NEI 99-01 does not specify a basis for source term activity or reduction factors.

RG 1.183 provides assumptions for a LOCA used as a reference for FSAR design basis event analysis. Per RG 1.183 Section 1.1.4, Emergency Preparedness Applications:

Requirements for emergency preparedness at nuclear power plants are set forth in 10 CFR 50.47, "Emergency Plans." Additional requirements are set forth in Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50. The planning basis for many of these requirements was published in NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants". This joint effort by the Environmental Protection Agency (EPA) and the NRC considered the principal characteristics (such as nuclides released and distances) likely to be involved for a spectrum of design basis and severe (core melt) accidents. No single accident scenario is the basis of the required preparedness. The objective of the planning is to provide public protection that would encompass a wide spectrum of possible events with a sufficient basis for extension of response efforts for unanticipated events. These requirements were issued after a long period of involvement by numerous stakeholders, including the Federal Emergency Management Agency, other Federal agencies, local and State governments (and in some cases, foreign governments), private citizens, utilities, and industry groups.

Although the AST provided in this guide was based on a limited spectrum of severe accidents, the particular characteristics have been tailored specifically for DBA analysis use. The AST is not representative of the wide spectrum of possible events that make up the planning basis of emergency preparedness. Therefore, the AST is insufficient by itself as a basis for requesting relief from the emergency preparedness requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50.

Thus, RG 1.183 is not used as a source basis for the containment radiation monitor thresholds.

Guidance contained in NUREG-1940 is considered representative of the wide spectrum of possible events that make up the planning basis of emergency preparedness and provides radiological consequence assessment methods which are acceptable to the NRC. Additionally, the source term used to develop the HNP effluent EAL thresholds and in the Unified RASCAL Interface/Radiological Assessment System for Consequence Analysis (URI/RASCAL) dose assessment model is from NUREG-1940. Thus, NUREG-1940 has been selected as a source term basis for the fission product barrier containment radiation thresholds for conformance to NRC guidance and consistency with other source term bases used within the HNP emergency preparedness program.

HNP Basis

2.4.1. The NUREG-1940 source term inputs used for the fission product barrier containment radiation thresholds are as follows:

Fuel Clad Damage Equivalent to 300 µCi/g DEI-131 – NUREG-1940 Table 1-1 equilibrium core activity, in conjunction with the NUREG-1940 Table 1-5 non-noble gas release fraction, is used to develop the site specific iodine source term.

Fuel Clad and Containment Barrier Thresholds – NUREG-1940 Figure 1-1 for cladding failure is used as a basis to establish this threshold.

RCS Barrier Threshold – NUREG-1940 Figure 1-1 for spiked coolant is used as a basis to establish this threshold.

2.4.2. NUREG-1940 source term is based on a generic plant with a power rating of 3000 MWt. Per FSAR Section 1.1.5, Core Thermal Power, and Renewed License No. NPF-63 Amendment No. 152, Section 2.C.(1) Maximum Power Level, the HNP site specific source term is taken from the Unit 1 licensed core thermal power output of 2948 megawatts (consistent with the posting on the NRC website).

The EAL thresholds are not based on the FSAR Chapter 15 rated power assumption of 2958, which includes 0.34% uncertainty.

2.4.3. Dose equivalent iodine 131 (DEI-131) dose conversion factors are derived from values provided in EPA-400-R-92-001. EPA-400 is used as the source for emergency preparedness related dose conversion factors and is the basis for the protective action guidelines.

The DEI-131 dose conversion factors are not based on the FSAR Chapter 15 or other 10 CFR 20 reference sources as those are not reflective of the assumptions used within the EPA guidance for emergency preparedness use.

2.5. Decay Considerations

Guidance Criteria

Fission product barrier thresholds and their associated EALs are applicable only when the plant is in Power Operation, Hot Standby, Startup, or Hot Shutdown modes (known as the hot operating modes, or modes 1-4).

Per NEI 99-01, the events for these thresholds correspond to an instantaneous release of all reactor coolant mass into the primary containment.

HNP Basis

Consistent with the NUREG-1940 graphs, the instantaneous release of the RCS to the containment is assumed to occur one hour after the damage event / reactor scram to account for damage progression, dispersion of activity and decay of the very short half-life isotopes.

NUREG-1940 provides additional containment radiation results for 24 hours after the damage event / reactor scram. Although NEI 99-01 does not require the use of multiple thresholds for the EAL, HNP will include multiple thresholds for consistency with fleet EALs.

3. DESIGN INPUTS

3.1. Constants and Conversion Factors

3.1.1. 453.592 g per lbm water conversion factor

3.2. Plant Inputs

3.2.1. Rated Power

- Standard Plant (NUREG-1940 Section 1.2.4) ....3,000 MWt
- HNP (FSAR 1.1.5) ....2,948 MWt

3.2.2. Reactor Coolant Mass (HNP-M/MECH-1020 Table 3)

- RCS mass ....4.41619 E+05 lbm

3.2.3. Standard Plant Containment Radiation Reading (NUREG-1940 Figure 1-1)

Spray Off

- 100% fuel clad damage .... 60,000 R/hr (@ 1 hr after shutdown)
- 100% fuel clad damage .... 30,000 R/hr (@ 24 hr after shutdown)
- 100% spiked coolant.. 50 R/hr (@ 1 hr after shutdown)
- 100% spiked coolant.. 6 R/hr (@ 24 hr after shutdown)

Spray On

- 100% fuel clad damage .... 12,000 R/hr (@ 1 hr after shutdown)
- 100% fuel clad damage .... 2,000 R/hr (@ 24 hr after shutdown)
- 100% spiked coolant.. 2 R/hr (@ 1 hr after shutdown)
- 100% spiked coolant... 0.2 R/hr (@ 24 hr after shutdown)

3.3. Source Term

3.3.1. Source Term Activity (NUREG-1940 Table 1-1)

	Core Activity (Ci/MWt)
I-131	2.67E+04
I-132	3.88E+04
I-133	5.42E+04
I-134	5.98E+04
I-135	5.18E+04

3.3.2. Release Fractions – RF_{Core} (NUREG-1940 Table 1-5)

- Non-Noble Gasses (I, Cs, Rb) – Fuel Clad Damage 0.05 (5%)

3.3.3. Iodine Dose Conversion Factors – DCF (EPA-400 Table 5-2)

	Rem/hr per µCi/cc
I-131	1.3E+06
I-132	7.7E+03
I-133	2.2E+05
I-134	1.3E+03
I-135	3.8E+04

4. CALCULATIONS

4.1. Fuel Clad Damage Estimate Based on 300 µCi/g DEI-131

4.1.1. 100% Core Activity Equivalent Reactor Coolant Iodine Concentrations

$$100\%\ Core\ RC\ Activity_i(\mu Ci/g) = \frac{Core\ Activity_i(Ci/MWt) \times Unit(MWt) \times 10^6}{RC\ Mass\ (g)}$$

	Coolant Activity (µCi/g)
I-131	3.93E+05
I-132	5.71E+05
I-133	7.98E+05
I-134	8.80E+05
I-135	7.62E+05
Total	3.40E+06

4.1.2. 100% Fuel Clad Activity Equivalent Reactor Coolant Iodine Concentrations

$$100\%\ Clad\ RC\ Activity_i(\mu Ci/g) = 100\%\ Core\ RC\ Activity_i(\mu Ci/g) \times RF_{Core}$$

	Coolant Activity (µCi/g)
I-131	1.96E+04
I-132	2.86E+04
I-133	3.99E+04
I-134	4.40E+04
I-135	3.81E+04
Total	1.70E+05

4.1.3. 100% Fuel Clad Activity Equivalent Reactor Coolant DEI-131 Concentrations

$$100\%\ DEI\ RC\ Activity\ (\mu Ci/g) = \sum 100\%\ Clad\ RC\ Activity_i(\mu Ci/g) \times DEI\ DCF_i$$

The DEI-131 value for each iodine isotope is determined as follows:

$$DEI\ DCF_i = \frac{EPA - 400\ Table\ 5 - 2\ Iodine\ DCF_i(Rem/hr\ per\ \mu Ci/cc)}{EPA - 400\ Table\ 5 - 2\ Iodine\ DCF_{I-131}(Rem/hr\ per\ \mu Ci/cc)}$$

	Coolant Activity (µCi/g)
I-131	1.96E+04
I-132	1.69E+02
I-133	6.75E+03
I-134	4.40E+01
I-135	1.11E+03
Total	2.77E+04

4.1.4. % Fuel Clad Activity Equivalent Reactor Coolant at 300 µCi/g DEI-131

$$\%\ Clad\ Damage = \frac{300\ \mu Ci/g}{100\%\ DEI\ RC\ Activity\ (\mu Ci/g)}$$

300 µCi/ml DEI-131 = 1.08% Fuel Clad Damage

See Attachment 1 for the spreadsheet calculations that develop the fuel clad source term activity and the % fuel clad damage estimates.

4.2. Fission Product Barrier Thresholds

4.2.1. Containment Potential Loss (20% Fuel Clad Damage)

$$RM_{20\%\ Clad}\ (R/hr) = Std\ Plant_{100\%\ Clad}\ (R/hr) \times \frac{MWt_{HNP}}{MWt_{Std\ Plant}} \times 20\%$$

Time After Shutdown	**Spray Off Containment Potential Loss**	**Spray On Containment Potential Loss**
1 hour	1.18E+04 R/hr	2.36E+03 R/hr
24 hours	5.90E+03 R/hr	3.93E+02 R/hr

See Attachment 2 for the spreadsheet calculations that develop the 20% fuel clad damage containment radiation monitor reading.

4.2.2. Fuel Clad Loss (300 µCi/g DEI-131 Equivalent Clad Damage)

$$RM_{DEI\ Damage}\ (R/hr) = Std\ Plant_{100\%\ Clad}\ (R/hr) \times \frac{MWt_{HNP}}{MWt_{Std\ Plant}} \times \%_{DEI\ Damage}$$

Time After Shutdown	Spray Off Fuel Clad Loss	Spray On Fuel Clad Loss
1 hour	6.38E+02 R/hr	1.28E+02 R/hr
24 hours	3.19E+02 R/hr	2.13E+01 R/hr

See Attachment 2 for the spreadsheet calculations that develop the 300 µCi/g DEI-131 equivalent fuel clad damage containment radiation monitor reading.

4.2.3. RCS Loss (Spiked Coolant)

$$RM_{Spiked}\ (R/hr) = Std\ Plant_{Spiked}\ (R/hr) \times \frac{MWt_{HNP}}{MWt_{Std\ Plant}}$$

Time After Shutdown	Spray Off RCS Loss	Spray On RCS Loss
1 hour	4.91E+01 R/hr	1.97+00 R/hr
24 hours	5.90E+00 R/hr	1.97E-01R/hr

See Attachment 2 for the spreadsheet calculations that develop the spiked RCS containment radiation monitor reading.

4.3. Extension and Interpolation of Data Points

The CHRM response to activity disbursed instantaneously and homogeneously throughout containment during a postulated LOCA event is influenced by multiple factors. Photon energy and abundance of each isotope are the major contributors with decay, daughter ingrowth, washout, settling and plate-out being additional factors when considering the effects of time upon the monitor reading from the onset.

While development of decay curves with complex consideration of the above influences is possible, thresholds for the EAL fission product barriers consistent with the values provided at 1 and 24 hours in NUREG-1940 is desired. Thus, the derivation of a relative decay factor based on the 1 hour and 24 hour data points is the method chosen to interpolate containment monitor reading at hourly intervals out to 72 hours (the time at which the modes applicable to the fission product barrier matrix are no longer likely).

See Attachment 4 for the equations and results for the derived containment high range radiation monitor hourly readings from 1 to 72 hours.

5. CONCLUSIONS

5.1. 300 µCi/g DEI-131 is equivalent to 1.08% Fuel Clad Damage

5.2. Time dependent containment high range radiation monitor fission product barrier values both with and without sprays are provided in Attachment 4.

5.3. Table F-2 presents selected values at 1, 2, 8 and 24 hours after shutdown (rounded for readability) as they will appear on the EAL Wallchart (see below). Values for sprays on were chosen based on operational strategies at Harris Nuclear Plant that actuate sprays during adverse containment conditions.

Table F-2 Containment Radiation (RM-1CR-3589SA, RM-1CR-3590SB)

Time After S/D (Hours)	FC Barrier Loss R/hr	RCS Barrier Loss R/hr*	CNMT Potential Loss R/hr
0 - 1	130	Refer to Attachment 5 for CVI Radiation Monitor FPB Threshold	2360
1 - 2	110		2000
2 - 8	70		1300
> 8	21		390

* Attachment 5 provides EAL Table F-2 RCS Barrier Loss containment radiation monitor threshold values using the Containment Ventilation Isolation radiation monitors (CVIs) in lieu of the Containment High Range Radiation Monitors.

6. REFERENCES

6.1. NEI 99-01 Revision 6, Development of Emergency Action Levels for Non-Passive Reactors, September 2012

6.2. EPA-400-R-92-001, Manual of Protective action Guides and Protective Actions for Nuclear Incidents, May 1992

6.3. WCAP-14696-A, Westinghouse Owners Group Core Damage Assessment Guide, Revision 1

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

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

6.6. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000

6.7. Harris Nuclear Plant FSAR

- 1.1.5 Core Thermal Power

6.8. Renewed License No. NPF-63 Amendment No. 152, Section 2.C.(1) Maximum Power Level

6.9. Technical Specifications, Harris Nuclear Plant, Revision 170

- Section 3.4.8, RCS Specific Activity

6.10. HNP-M/MECH-1020, Reactor Vessel and Reactor Coolant System Water Volumes, Revision 3

6.11. EPM-601, Core Damage Assessment Technical Basis, Revision 2

	NUREG-1940 Table 1-1 Core Activity (Ci/MWt)	HNP Core Activity (Ci)	HNP RCS Activity (µCi/g 100% Core)	HNP RCS Activity (µCi/g 100% Clad)	EPA-400 Table 5-2 Iodine DCFs (rem per µCi/cc)	DEI-131 ICF	RCS Activity (µCi/g 100% Clad DEI-131)
I-131	2.67E+04	7.87E+07	3.93E+05	1.96E+04	1.3E+06	1.00E+00	1.96E+04
I-132	3.88E+04	1.14E+08	5.71E+05	2.86E+04	7.7E+03	5.92E-03	1.69E+02
I-133	5.42E+04	1.60E+08	7.98E+05	3.99E+04	2.2E+05	1.69E-01	6.75E+03
I-134	5.98E+04	1.76E+08	8.80E+05	4.40E+04	1.3E+03	1.00E-03	4.40E+01
I-135	5.18E+04	1.53E+08	7.62E+05	3.81E+04	3.8E+04	2.92E-02	1.11E+03
Total	2.31E+05	6.82E+08	3.40E+06	1.70E+05			2.77E+04

Volume Conversion (g/lbm):	453.592	**Rated Power (MWt):**	2948
Non-Noble Gas RF (%):	5.0%	**RC Liquid Mass (lbm):**	4.42E+05
		RC Liquid Mass (g):	2.00E+08
Target DEI-131 (µCi/g):	3.00E+02		
		% Clad Damage:	1.08%

	Reading for 300 µCi/cc RCS (Fuel Clad FPB R/hr)	Reading for Spiked RCS (RCS FPB R/hr)	Reading for 20% Clad Failure (Containment FPB in R/hr)
1 hour - Spray Off	6.38E+02	4.91E+01	1.18E+04
24 hours - Spray Off	3.19E+02	5.90E+00	5.90E+03
1 hour - Spray On	1.28E+02	1.97E+00	2.36E+03
24 hours - Spray On	2.13E+01	1.97E-01	3.93E+02

% Damage for 300 µCi/g DEI-131 RCS Activity:	1.08%
Standard Plant (MWt):	3000
HNP Rated Power (MWt):	2948

NUREG-1940 Figure 1-1 Containment Rad Readings

1 hr spray off 100% Clad Failure (R/hr):	6.00E+04
24 hr spray off 100% Clad Failure (R/hr):	3.00E+04
1 hr spray on 100% Clad Failure (R/hr):	1.20E+04
24 hr spray on 100% Clad Failure (R/hr):	2.00E+03
1 hr spray off Spiked Coolant (R/hr):	5.00E+01
24 hr spray off Spiked Coolant (R/hr):	6.00E+00
1 hr spray on Spiked Coolant (R/hr):	2.00E+00
24 hr spray on Spiked Coolant (R/hr):	2.00E-01

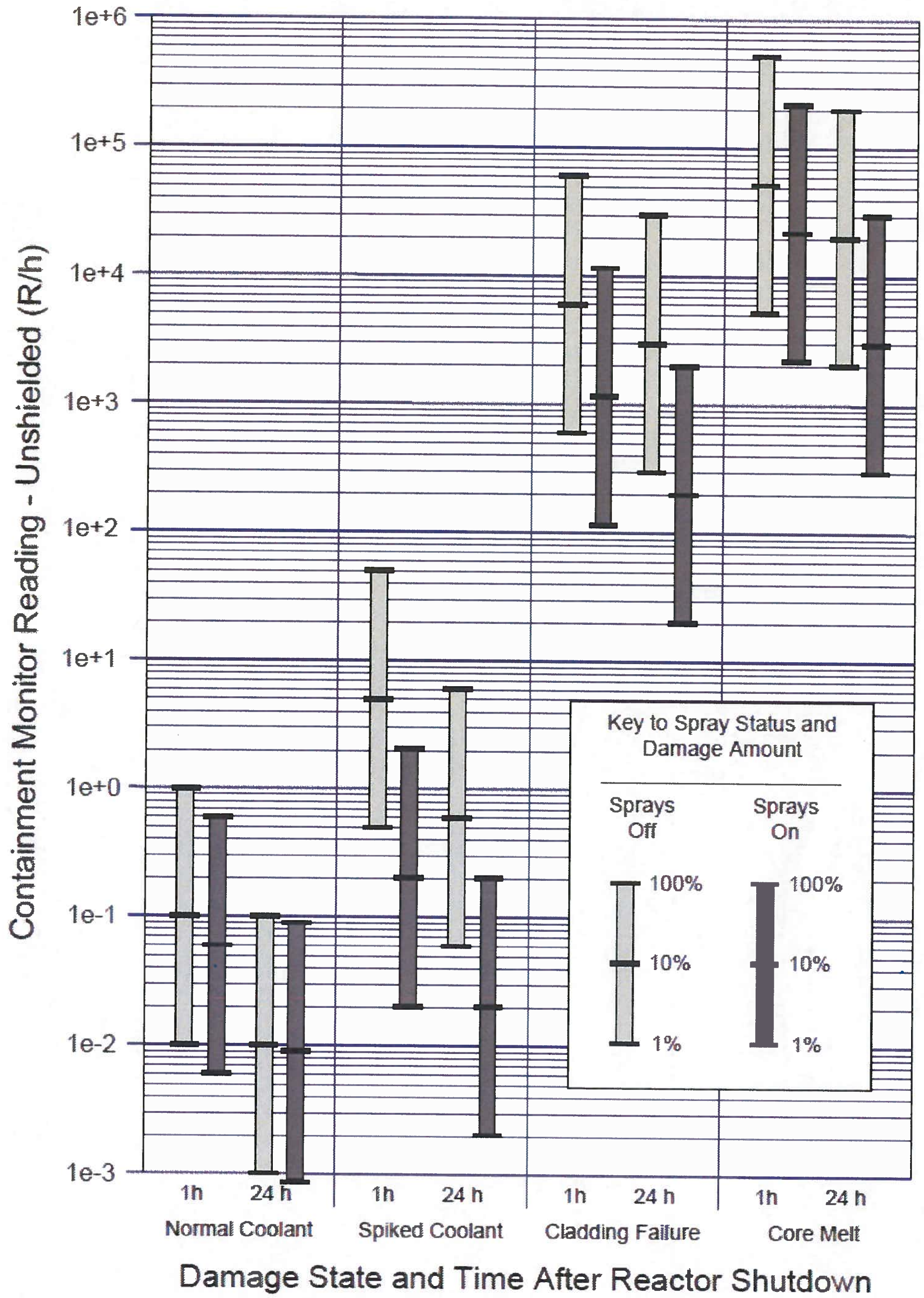


Figure 1-1 PWR containment monitor response

1-14

1) The relative decay constants related to the containment radiation monitor results from NUREG-1940 Figure 1-1 are derived as follows:

$$k = \frac{\ln\left(\frac{R_t}{R_i}\right)}{-t}$$

Where:

R_t	Monitor Reading at time interval (t)
R_i	Initial Monitor Reading
t	time interval
k	decay constant

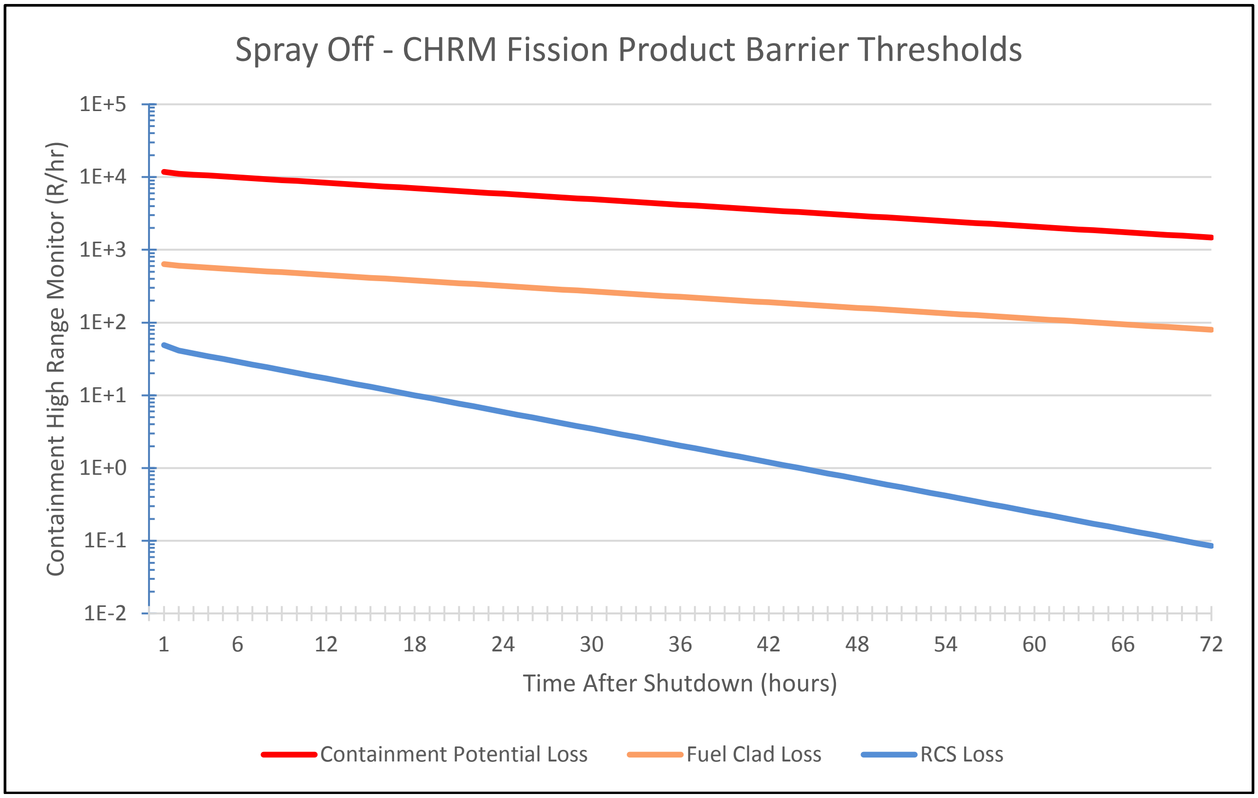
Spray Off	Ri (1 hr)	Rt (24 hr)	k
Containment Potential Loss Threshold (R/hr)	1.18E+04	5.90E+03	2.89E-02
Fuel Clad Loss Threshold (R/hr)	6.38E+02	3.19E+02	2.89E-02
RCS Loss Threshold (R/hr)	4.91E+01	5.90E+00	8.83E-02

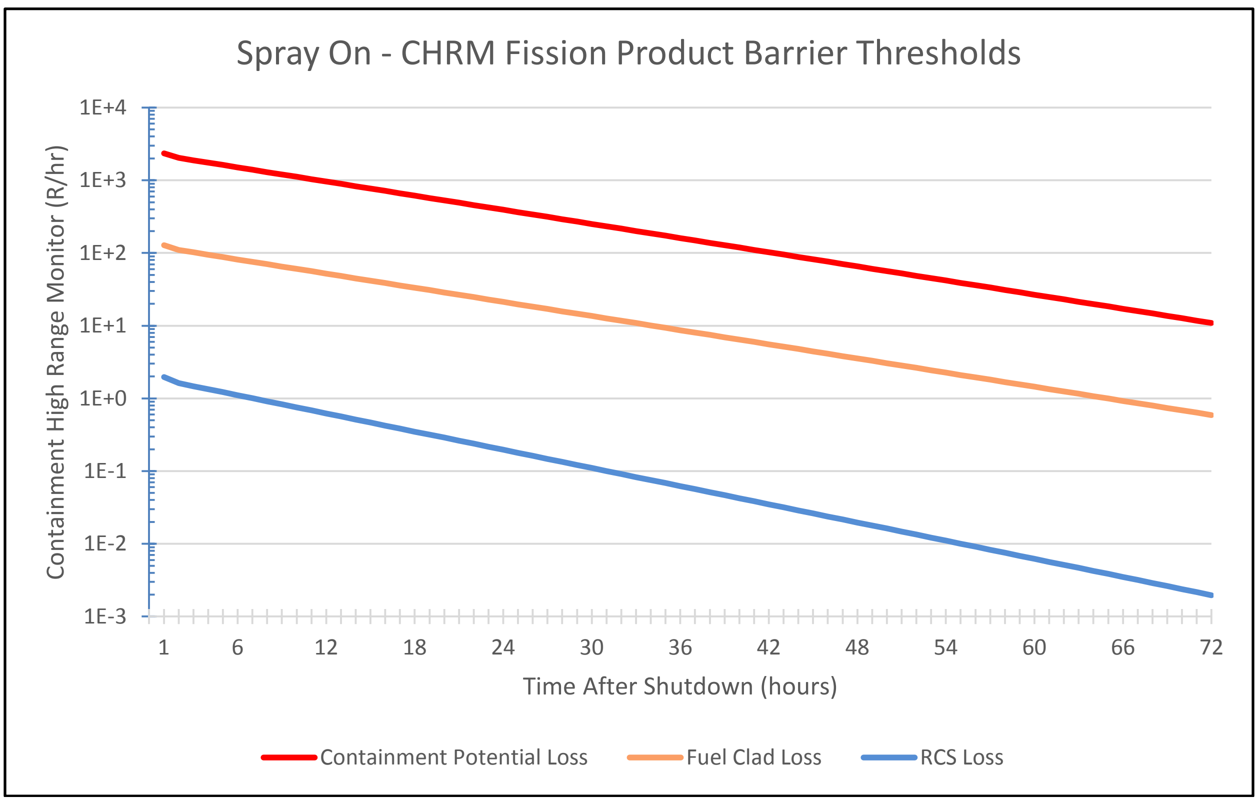
Spray On	Ri (1 hr)	Rt (24 hr)	k
Containment Potential Loss Threshold (R/hr)	2.36E+03	3.93E+02	7.47E-02
Fuel Clad Loss Threshold (R/hr)	1.28E+02	2.13E+01	7.47E-02
RCS Loss Threshold (R/hr)	1.97E+00	1.97E-01	9.59E-02

t (hours): 24

2) The containment radiation monitor readings for various times after shutdown are then determined as follows:

$$R_t = R_i \times e^{-kt}$$





Note: Values provided below are the calculation results given to illustrate trends. It is recognized that the low scale of the high range containment monitor is above the later values for the RCS loss fission product barrier and are not appropriate for use in event declaration.

Spray Off

Time After Shutdown (hours)	Containment Potential Loss Threshold (R/hr)	Fuel Clad Loss Threshold (R/hr)	RCS Loss Threshold (R/hr)
1	1.18E+04	6.38E+02	4.91E+01
2	1.11E+04	6.02E+02	4.12E+01
3	1.08E+04	5.85E+02	3.77E+01
4	1.05E+04	5.68E+02	3.45E+01
5	1.02E+04	5.52E+02	3.16E+01
6	9.92E+03	5.37E+02	2.89E+01
7	9.63E+03	5.21E+02	2.65E+01
8	9.36E+03	5.06E+02	2.42E+01
9	9.09E+03	4.92E+02	2.22E+01
10	8.83E+03	4.78E+02	2.03E+01
11	8.58E+03	4.64E+02	1.86E+01
12	8.34E+03	4.51E+02	1.70E+01
13	8.10E+03	4.38E+02	1.56E+01
14	7.87E+03	4.26E+02	1.43E+01
15	7.65E+03	4.14E+02	1.31E+01
16	7.43E+03	4.02E+02	1.20E+01
17	7.22E+03	3.90E+02	1.09E+01
18	7.01E+03	3.79E+02	1.00E+01
19	6.81E+03	3.69E+02	9.17E+00
20	6.62E+03	3.58E+02	8.40E+00
21	6.43E+03	3.48E+02	7.69E+00
22	6.25E+03	3.38E+02	7.04E+00
23	6.07E+03	3.28E+02	6.44E+00
24	5.90E+03	3.19E+02	5.90E+00
25	5.73E+03	3.10E+02	5.40E+00
26	5.57E+03	3.01E+02	4.94E+00
27	5.41E+03	2.93E+02	4.52E+00
28	5.25E+03	2.84E+02	4.14E+00
29	5.10E+03	2.76E+02	3.79E+00
30	4.96E+03	2.68E+02	3.47E+00
31	4.82E+03	2.61E+02	3.18E+00
32	4.68E+03	2.53E+02	2.91E+00
33	4.55E+03	2.46E+02	2.66E+00
34	4.42E+03	2.39E+02	2.44E+00
35	4.29E+03	2.32E+02	2.23E+00
36	4.17E+03	2.26E+02	2.04E+00

Time After Shutdown (hours)	Containment Potential Loss Threshold (R/hr)	Fuel Clad Loss Threshold (R/hr)	RCS Loss Threshold (R/hr)
37	4.05E+03	2.19E+02	1.87E+00
38	3.94E+03	2.13E+02	1.71E+00
39	3.82E+03	2.07E+02	1.57E+00
40	3.71E+03	2.01E+02	1.43E+00
41	3.61E+03	1.95E+02	1.31E+00
42	3.51E+03	1.90E+02	1.20E+00
43	3.41E+03	1.84E+02	1.10E+00
44	3.31E+03	1.79E+02	1.01E+00
45	3.21E+03	1.74E+02	9.22E-01
46	3.12E+03	1.69E+02	8.44E-01
47	3.03E+03	1.64E+02	7.73E-01
48	2.95E+03	1.60E+02	7.08E-01
49	2.86E+03	1.55E+02	6.48E-01
50	2.78E+03	1.51E+02	5.93E-01
51	2.70E+03	1.46E+02	5.43E-01
52	2.63E+03	1.42E+02	4.97E-01
53	2.55E+03	1.38E+02	4.55E-01
54	2.48E+03	1.34E+02	4.16E-01
55	2.41E+03	1.30E+02	3.81E-01
56	2.34E+03	1.27E+02	3.49E-01
57	2.27E+03	1.23E+02	3.19E-01
58	2.21E+03	1.19E+02	2.92E-01
59	2.15E+03	1.16E+02	2.68E-01
60	2.08E+03	1.13E+02	2.45E-01
61	2.03E+03	1.10E+02	2.24E-01
62	1.97E+03	1.06E+02	2.05E-01
63	1.91E+03	1.03E+02	1.88E-01
64	1.86E+03	1.00E+02	1.72E-01
65	1.80E+03	9.76E+01	1.58E-01
66	1.75E+03	9.48E+01	1.44E-01
67	1.70E+03	9.21E+01	1.32E-01
68	1.65E+03	8.95E+01	1.21E-01
69	1.61E+03	8.70E+01	1.11E-01
70	1.56E+03	8.45E+01	1.01E-01
71	1.52E+03	8.21E+01	9.27E-02
72	1.47E+03	7.98E+01	8.49E-02

Spray On

Time After Shutdown (hours)	Containment Potential Loss Threshold (R/hr)	Fuel Clad Loss Threshold (R/hr)	RCS Loss Threshold (R/hr)
1	2.36E+03	1.28E+02	1.97E+00
2	2.03E+03	1.10E+02	1.62E+00
3	1.89E+03	1.02E+02	1.47E+00
4	1.75E+03	9.47E+01	1.34E+00
5	1.62E+03	8.79E+01	1.22E+00
6	1.51E+03	8.15E+01	1.11E+00
7	1.40E+03	7.57E+01	1.00E+00
8	1.30E+03	7.02E+01	9.12E-01
9	1.20E+03	6.52E+01	8.29E-01
10	1.12E+03	6.05E+01	7.53E-01
11	1.04E+03	5.61E+01	6.84E-01
12	9.63E+02	5.21E+01	6.21E-01
13	8.94E+02	4.83E+01	5.65E-01
14	8.29E+02	4.49E+01	5.13E-01
15	7.70E+02	4.16E+01	4.66E-01
16	7.14E+02	3.86E+01	4.23E-01
17	6.63E+02	3.59E+01	3.85E-01
18	6.15E+02	3.33E+01	3.49E-01
19	5.71E+02	3.09E+01	3.18E-01
20	5.30E+02	2.87E+01	2.88E-01
21	4.92E+02	2.66E+01	2.62E-01
22	4.56E+02	2.47E+01	2.38E-01
23	4.24E+02	2.29E+01	2.16E-01
24	3.93E+02	2.13E+01	1.97E-01
25	3.65E+02	1.97E+01	1.79E-01
26	3.39E+02	1.83E+01	1.62E-01
27	3.14E+02	1.70E+01	1.47E-01
28	2.92E+02	1.58E+01	1.34E-01
29	2.71E+02	1.46E+01	1.22E-01
30	2.51E+02	1.36E+01	1.11E-01
31	2.33E+02	1.26E+01	1.00E-01
32	2.16E+02	1.17E+01	9.12E-02
33	2.01E+02	1.09E+01	8.29E-02
34	1.86E+02	1.01E+01	7.53E-02
35	1.73E+02	9.36E+00	6.84E-02
36	1.60E+02	8.68E+00	6.21E-02

Time After Shutdown (hours)	Containment Potential Loss Threshold (R/hr)	Fuel Clad Loss Threshold (R/hr)	RCS Loss Threshold (R/hr)
37	1.49E+02	8.06E+00	5.65E-02
38	1.38E+02	7.48E+00	5.13E-02
39	1.28E+02	6.94E+00	4.66E-02
40	1.19E+02	6.44E+00	4.23E-02
41	1.10E+02	5.98E+00	3.85E-02
42	1.03E+02	5.55E+00	3.49E-02
43	9.52E+01	5.15E+00	3.18E-02
44	8.83E+01	4.78E+00	2.88E-02
45	8.20E+01	4.43E+00	2.62E-02
46	7.61E+01	4.12E+00	2.38E-02
47	7.06E+01	3.82E+00	2.16E-02
48	6.55E+01	3.54E+00	1.97E-02
49	6.08E+01	3.29E+00	1.79E-02
50	5.64E+01	3.05E+00	1.62E-02
51	5.24E+01	2.83E+00	1.47E-02
52	4.86E+01	2.63E+00	1.34E-02
53	4.51E+01	2.44E+00	1.22E-02
54	4.19E+01	2.26E+00	1.11E-02
55	3.88E+01	2.10E+00	1.00E-02
56	3.61E+01	1.95E+00	9.12E-03
57	3.35E+01	1.81E+00	8.29E-03
58	3.11E+01	1.68E+00	7.53E-03
59	2.88E+01	1.56E+00	6.84E-03
60	2.67E+01	1.45E+00	6.21E-03
61	2.48E+01	1.34E+00	5.65E-03
62	2.30E+01	1.25E+00	5.13E-03
63	2.14E+01	1.16E+00	4.66E-03
64	1.98E+01	1.07E+00	4.23E-03
65	1.84E+01	9.96E-01	3.85E-03
66	1.71E+01	9.25E-01	3.49E-03
67	1.59E+01	8.58E-01	3.18E-03
68	1.47E+01	7.96E-01	2.88E-03
69	1.37E+01	7.39E-01	2.62E-03
70	1.27E+01	6.86E-01	2.38E-03
71	1.18E+01	6.37E-01	2.16E-03
72	1.09E+01	5.91E-01	1.97E-03

1. Purpose

The purpose of Attachment 5 is to provide EAL Table F-2 RCS Barrier Loss containment radiation monitor threshold values using the Containment Ventilation Isolation radiation monitors (CVIs) in lieu of the Containment High Range Radiation Monitors (CHRRMs) minimum instrument range value of 5 R/hr.

There are 4 CVIs (RM-1CR-3561A-SA, RM-1CR-3561B-SB, RM-1CR-3561C-SA, RM-1CR-3561D-SB) that are located at the operating deck level around the refueling cavity. The CVIs' primary advantage over the CHRRMs for application to the EAL Table F-2 RCS Barrier Loss threshold is that the CVIs have better low range sensitivity than the CHRRMs (10^1 - 10^7 mR/hr (CVIs) versus 10^0 - 10^8 R/hr (CHRRMs)) {3.1}.

2. Calculation

Harris calculation TDR-RC-013 {3.2} provides a method to equate CHRRM readings to CVI radiation monitor readings for RM-1CR-3561A-SA using the ratio (12.2 R/hr CVI/17.5 R/hr CHRRM) (~70%). RM-1CR-3561A-SA is located on the inner wall of containment at the end of the refueling cavity, and is furthest away from the centerline of the reactor core. RM-1CR-3561B-SB and RM-1CR-3561C-SA are located on the inboard side of the 'A' Steam Generator Missile shield which is adjacent to the Reactor Refueling Cavity. RM-1CR-3561D-SB is located on the inboard side of the 'C' Steam Generator Missile shield which is adjacent to the Reactor Refueling Cavity. NUREG/BR-0150 {3.3}, Method A.4, "Evaluation of Containment Radiation" establishes a basis to use a radiation monitor for a core damage assessment as: "*Confirm that the containment radiation monitor "sees" more than 50% of the shaded area shown in either Fig. A-3 (PWR) or Fig. A-4 (BWR)."* Based upon plant drawings CAR-2166-G-453 {3.4} and CAR-2165-G-013 {3.5}, all four CVI radiation monitors satisfy the NUREG/BR-0150 Fig. A-3 (PWR) guidelines. All four CVI radiation monitors would be immersed within the same plume and will "see" a similar volume of the containment building, therefore all four CVI radiation monitors are equivalent and can be used for EAL Classification.

Applying the 12.2/17.5 ratio to the "Spray On" CHRRM time dependent readings (R/hr) shown on Attachment 4, page 23 of this calculation and multiplying by 1000 to convert R/hr to mR/hr results in the following CVI threshold values:

Time After S/D (Hours)	CHRRMs RCS Barrier Loss R/hr	CVIs RCS Barrier Loss mR/hr
0 - 1	1.97	1.373E+03
1 - 2	1.62	1.129E+03
2 - 8	0.912	6.358E+02
> 8	0.197	1.373E+02

For example, 1.97 R/hr x 12.2/17.5 x 1000 mR/R = 1.373E+03 mR/hr CVI equivalent reading

Truncating for use in EAL Table F-2 results in the following:

Table F-2 Containment Radiation For FC Barrier Loss and CNTM Potential Loss (RM-1CR-3589SA, RM-1CR-3590SB) For RCS Barrier Loss (RM-1CR-3561A-SA, RM-1CR-3561B-SB, RM-1CR-3561C-SA, RM-1CR-3561D-SB)			
Time After S/D (Hours)	**FC Barrier Loss R/hr**	**RCS Barrier Loss mR/hr**	**CNMT Potential Loss R/hr**
0 - 1	130	1.37E+03	2360
1 - 2	110	1.12E+03	2000
2 - 8	70	6.35E+02	1300
> 8	21	1.37E+02	390

3. References

3.1 Harris UFSAR, Table 12.3.4-1, "Area Radiation Monitors," Amendment 61.

3.2 Harris Calculation TDR-RC-013, "Determination of EAL Dose Rate for Monitor RM-1CR-3561A-SA," Rev. 0.

3.3 NUREG/BR-0150, Response Technical Manual (RTM-96), US Nuclear Regulatory Commission, Vol. 1, Rev. 4.

3.4 Drawing CAR-2166-G-453 (Fusion Drawing 6-G-0453), Instrument Location Arrangement, Unit 1, Plan El. 286', Revision 16.

3.5 Drawing CAR-2165-G-013 (Fusion Drawing 5-G-013), General Arrangement Containment Building Sections Sheet 1, Revision 17.

SERIAL HNP-18-004

ENCLOSURE 5

CALCULATION FOR RADIATION MONITOR READINGS FOR CORE UNCOVERY DURING REFUELING

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

NUCLEAR DEVELOPMENT / OPERATING FLEET / DECOMMISSIONED

CSD-EP-HNP-0101-06

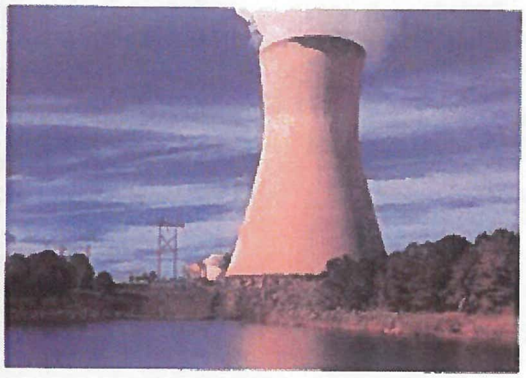
RADIATION MONITOR READINGS FOR CORE UNCOVERY DURING REFUELING

REVISION 0

Effective Dates:

N/A	N/A	N/A	06/28/2018	N/A
Brunswick	Catawba	Crystal River	Harris	Lee
N/A	N/A	N/A	N/A	N/A
Levy	McGuire	Oconee	Robinson	NGO

DUKE ENERGY®



Harris Nuclear Plant (HNP)

Radiation Monitor Readings for Core Uncovery during Refueling

CSD-EP-HNP-0101-06
Revision 0

Document Author: Scott McCain

Technical Reviewer: William Cerame

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Table of Contents

ATTACHMENTS

1 PURPOSE

The Shearon Harris Nuclear Plant (HNP) 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 cold shutdown radiation monitor readings in containment developed in accordance with the guidance in NEI 99-01 Revision 6.

Documentation of the assumptions, calculations and results are provided for the HNP Cx1 series EAL values associated the NEI 99-01 Revision 6 EALs listed below.

- NEI EAL CS1.3
- NEI EAL CG1.2

2 DEVELOPMENT METHOD AND BASES

2.1 CS1.3

Guidance Criteria

The initiating condition for CS1 is "Loss of RCS inventory affecting core decay heat removal capability." To be met, EAL CS1.3 requires two threshold conditions (inability to monitor reactor vessel RCS level and core uncovery indication).

Per NEI 99-01 Revision 6, for EAL #3.b bullet one - As water level in the reactor vessel lowers, the dose rate above the core will increase. Enter a "site-specific radiation monitor" that could be used to detect core uncovery and the associated "site-specific value" indicative of core uncovery.

It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.

To further promote accurate classification, developers should consider if some combination of monitors could be specified in the EAL to build-in an appropriate level of corroboration between monitor readings into the classification assessment.

HNP Bases

The CS1.3 radiation monitor threshold is predicated on the loss of water above the core during refueling shutdown with the reactor vessel head removed. The monitors used do not have direct line of sight. Therefore, the threshold value will be calculated by backscatter from the containment dome.

2.2 CG1.2

Guidance Criteria

The initiating condition for CG1 is "Loss of RCS inventory affecting fuel clad integrity with containment challenged." To be met, EAL CG1.2 requires three threshold conditions (inability to monitor reactor vessel RCS level, core uncovery indication and containment challenge indication).

Per NEI 99-01 Rev. 6, for EAL #2.b bullet one - As water level in the reactor vessel lowers, the dose rate above the core will increase. Enter a "site-specific radiation monitor" that could be used to detect core uncovery and the associated "site-specific value" indicative of core uncovery.

It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading. This provision notwithstanding, if the estimated/calculated monitor reading is greater than approximately 110% of the highest accurate monitor reading, then developers may choose not to include the monitor as an indication and identify an alternate EAL threshold.

To further promote accurate classification, developers should consider if some combination of monitors could be specified in the EAL to build-in an appropriate level of corroboration between monitor readings into the classification assessment.

HNP Bases

The CG1.2 radiation monitor threshold is predicated on the loss of water above the core during refueling shutdown with the reactor vessel head removed. The monitors used do not have direct line of sight. Therefore, the threshold value will be calculated by backscatter from the containment dome.

2.3 Event Conditions

Guidance Criteria

Per NEI 99-01 Revision 6, these ICs address a significant and prolonged loss of RCS inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant.

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

HNP Bases

A single calculated radiation monitor reading cannot represent the point of core uncovery for all event scenarios due to multiple variables that can affect the source term (such as number of fuel assemblies in the core, fuel burn up in each assembly and time after shutdown). Therefore, reasonable assumptions and inputs have been selected so that the calculated value will be low enough to be a valid indication of fuel uncovery, but high enough so that inadvertent classifications are not made due to maintenance events or other conditions which may increase radiation levels without a coincident loss of reactor vessel water level.

3 DESIGN INPUTS

3.1 Constants and Conversion Factors

3.1.1 16.387 cc per in^3 unit conversion factor

3.1.2 453.592 g per lbm unit conversion factor

3.2 Plant Inputs

3.2.1 Rated Power

- Standard Plant (NUREG-1940 Section 1.2.4) ....3,000 MWt
- HNP (FSAR 1.1.5) ....2,948 MWt

3.2.2 Containment Geometry

- Reactor vessel head is removed.
- Top of Active Fuel (EAL technical basis manual) ....249’
- Containment Spring Line height (Drawing CAR-2165-G-0013) ....376’
- Containment Dome radius (Drawing CAR-2165-G-0013)....................65’
- Containment Dome height (sum of spring line and dome radius) ....441’

3.3 Radiation Monitors

3.3.1 Monitor Ranges (FSAR TABLE 12.3.4-1)

- RM-1CR-3561A–SA .... 1E+1 – 1E+7 mR/hr
- RM-1CR-3561B–SB .... 1E+1 – 1E+7 mR/hr
- RM-1CR-3561C–SA .... 1E+1 – 1E+7 mR/hr
- RM-1CR-3561D–SB .... 1E+1 – 1E+7 mR/hr

3.3.2 Detector Location - Elevation (Drawing CAR-2166-G-0453)

- RE-1CR-3561–ASA ....289’ 8”
- RE-1CR-3561–BSB ....289’ 6”
- RE-1CR-3561–CSA ....289’ 6”
- RE-1CR-3561–DSB ....289’ 6”

3.3.3 Detector Location – Radial Distance from Core Center (Equipment Database System)

- RE-1CR-3561–ASA .... 62’
- RE-1CR-3561–BSB .... 26’
- RE-1CR-3561–CSA .... 20’
- RE-1CR-3561–DSB .... 16’

3.4 Source Term

3.4.1 Source Term Activity (NUREG-1940 Table 1-1)

	Core Activity (Ci/MWt)
Ba-139	4.74E+04
Ba-140	4.76E+04
Ce-141	4.39E+04
Ce-143	4.00E+04
Ce-144*	3.54E+04
Cm-242	1.12E+03
Cs-134	4.70E+03
Cs-136	1.49E+03
Cs-137*	3.25E+03
I-131	2.67E+04
I-132	3.88E+04
I-133	5.42E+04
I-134	5.98E+04
I-135	5.18E+04
Kr-83m	3.05E+03
Kr-85	2.78E+02
Kr-85m	6.17E+03
Kr-87	1.23E+04
Kr-88	1.70E+04
La-140	4.91E+04

	Core Activity (Ci/MWt)
La-141	4.33E+04
La-142	4.21E+04
Mo-99	5.30E+04
Nb-95	4.50E+04
Nd-147	1.75E+04
Np-239	5.69E+05
Pr-143	3.96E+04
Pu-241	4.26E+03
Rb-86	5.29E+01
Rh-105	2.81E+04
Ru-103	4.34E+04
Ru-105	3.06E+04
Ru-106*	1.55E+04
Sb-127	2.39E+03
Sb-129	8.68E+03
Sr-89	2.41E+04
Sr-90	2.39E+03
Sr-91	3.01E+04
Sr-92	3.24E+04
Tc-99m	4.37E+04

	Core Activity (Ci/MWt)
Te-127	2.36E+03
Te-127m	3.97E+02
Te-129	8.26E+03
Te-129m	1.68E+03
Te-131m	5.41E+03
Te-132	3.81E+04
Xe-131m	3.65E+02
Xe-133	5.43E+04
Xe-133m	1.72E+03
Xe-135	1.42E+04
Xe-135m	1.15E+04
Xe-138	4.56E+04
Y-90	2.45E+03
Y-91	3.17E+04
Y-92	3.26E+04
Y-93	2.52E+04
Zr-95	4.44E+04
Zr-97*	4.23E+04

Notes:

NUREG-1940 radionuclides with an * are assumed to be present in secular equilibrium with short-lived daughters.

Additional source term from activation materials in the reactor vessel, such as vessel walls and internals, is not included.

3.4.2 Decay Time

Note - Source term nuclide half-life and energies for the decay corrected core inventory, including daughter nuclides, are internal to the MicroShield® application.

- HNP URI New Fuel Default Time After Shutdown .... 72 hours

3.4.3 Source Materials Weight

- Fuel Assembly dry weight (FS1-0030999 Table 2-2) .... 1498.9 lb
- # Fuel Assemblies (FSAR Table 4.1.1-1)..157
- Fuel (UO_2) weight (FSAR Table 4.1.1-1)... 181,190 lb
- Zirconium alloy weight (FSAR Table 4.1.1-1)... 41,415 lb
- Other Metals[1] weight (see Section 4.2.2 and Attachment 1) ....1.27E+4 lb

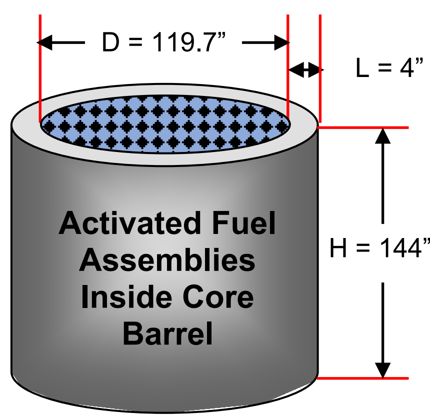
3.4.4 Source Materials Density

- Uranium Dioxide – UO_2 (NIST) .... 10.96 g/cc
- Other Metals – iron (Attachment 1) .... 7.86 g/cc
- Zirconium (Attachment 1)... 6.5 g/cc
- Water (Attachment 1).. 1.0 g/cc
- Air (Attachment 1)... 0.00122 g/cc

3.4.5 Source Geometry

The source is considered a cylinder with core activity evenly distributed throughout the equivalent volume. For the outer liner, core barrel thickness is used, although ID/OD are adjusted for modeling simplification. The use of this geometry is not considered significant to the results as the dose receptor points are located over the source.

- Active fuel height (FSAR Table 4.1.1-1)..144”
- Core equivalent diameter (FSAR Table 4.1.1-1) ....119.7”
- Core Barrel thickness (FSAR Table 4.1.1-1).. 4”



[1] Assembly metal weight (Inconel and other metals) density input as equivalent to iron.

3.4.6 Receptor Geometry

The reflected dose rate at the area radiation monitors is calculated using the methods of Davisson's "Gamma Ray Dose Albedos" (see Attachment 5).

The calculation is based on a concrete reflector at the top of containment, with a diameter equal to the containment dome. The distance from the reflector at θ_o to the radiation monitor is equal to the hypotenuse of the triangle formed by the difference in elevations of the reflector and the monitor and the distance from core center to the monitor.

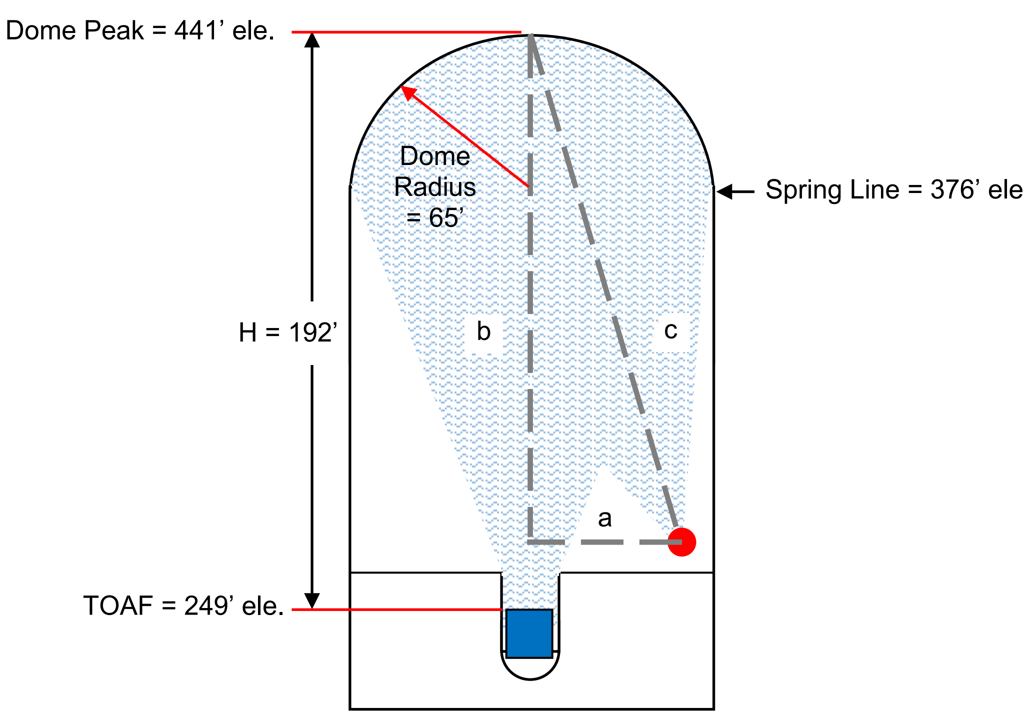
- Concrete is selected as the predominant containment dome reflector material. HNP containment has a carbon steel liner prior to the concrete so both materials contribute to reflection. References for albedo values representing layered materials were not available, thus the greater albedo of the two reflectors is used by itself.

- The reflected dose rate is related to the area of the reflector. Assuming a non-collimated beam originating from the core, the entire containment dome radius would be within line of sight.

RPV Flange Elevation (ft):	260	RPV Inside Diameter (in):	155.5
TOAF Elevation (ft):	249	RPV Inside Diameter (ft):	13.0
Source to Flange (ft):	11	RPV Radius (ft):	6.5

Angle Between Core Center Line & RPV Flange (degrees):	30.5

Distance Where Beam Equals Containment Radius (ft):	122
Elevation Where Beam Equals Containment Radius:	314



4 CALCULATIONS

4.1 Initial Source Term (Reactor Core) Activity

The source term input to MicroShield® is based on NUREG 1940 Table 1-1 isotopes and core activity, adjusted to HNP specific power rating.

$$HNP\ Core\ Activity_i(Ci) = Core\ Activity_i(Ci/MWt) \times HNP(MWt)$$

	NUREG-1940 Table 1-1 Core Activity (Ci/MWt)	HNP Core Activity (Ci)
Ba-139	4.74E+04	1.40E+08
Ba-140	4.76E+04	1.40E+08
Ce-141	4.39E+04	1.29E+08
Ce-143	4.00E+04	1.18E+08
Ce-144*	3.54E+04	1.04E+08
Cm-242	1.12E+03	3.30E+06
Cs-134	4.70E+03	1.39E+07
Cs-136	1.49E+03	4.39E+06
Cs-137*	3.25E+03	9.58E+06
I-131	2.67E+04	7.87E+07
I-132	3.88E+04	1.14E+08
I-133	5.42E+04	1.60E+08
I-134	5.98E+04	1.76E+08
I-135	5.18E+04	1.53E+08
Kr-83m	3.05E+03	8.99E+06
Kr-85	2.78E+02	8.20E+05
Kr-85m	6.17E+03	1.82E+07
Kr-87	1.23E+04	3.63E+07
Kr-88	1.70E+04	5.01E+07
La-140	4.91E+04	1.45E+08

	NUREG-1940 Table 1-1 Core Activity (Ci/MWt)	HNP Core Activity (Ci)
La-141	4.33E+04	1.28E+08
La-142	4.21E+04	1.24E+08
Mo-99	5.30E+04	1.56E+08
Nb-95	4.50E+04	1.33E+08
Nd-147	1.75E+04	5.16E+07
Np-239	5.69E+05	1.68E+09
Pr-143	3.96E+04	1.17E+08
Pu-241	4.26E+03	1.26E+07
Rb-86	5.29E+01	1.56E+05
Rh-105	2.81E+04	8.28E+07
Ru-103	4.34E+04	1.28E+08
Ru-105	3.06E+04	9.02E+07
Ru-106*	1.55E+04	4.57E+07
Sb-127	2.39E+03	7.05E+06
Sb-129	8.68E+03	2.56E+07
Sr-89	2.41E+04	7.10E+07
Sr-90	2.39E+03	7.05E+06
Sr-91	3.01E+04	8.87E+07
Sr-92	3.24E+04	9.55E+07
Tc-99m	4.37E+04	1.29E+08

	NUREG-1940 Table 1-1 Core Activity (Ci/MWt)	HNP Core Activity (Ci)
Te-127	2.36E+03	6.96E+06
Te-127m	3.97E+02	1.17E+06
Te-129	8.26E+03	2.44E+07
Te-129m	1.68E+03	4.95E+06
Te-131m	5.41E+03	1.59E+07
Te-132	3.81E+04	1.12E+08
Xe-131m	3.65E+02	1.08E+06
Xe-133	5.43E+04	1.60E+08
Xe-133m	1.72E+03	5.07E+06
Xe-135	1.42E+04	4.19E+07
Xe-135m	1.15E+04	3.39E+07
Xe-138	4.56E+04	1.34E+08
Y-90	2.45E+03	7.22E+06
Y-91	3.17E+04	9.35E+07
Y-92	3.26E+04	9.61E+07
Y-93	2.52E+04	7.43E+07
Zr-95	4.44E+04	1.31E+08
Zr-97*	4.23E+04	1.25E+08

Rated Power (MWt): 2948

4.2 Source (Reactor Core) Volume

See Attachment 1 for the spreadsheet results for the equations provided below.

4.2.1 Source Cylinder Total Volume

$$Volume_{source}(cc) = \pi r^2(in) \times height(in) \times 16.387\left({}^{cc}/_{in}\right)$$

4.2.2 Weight of Other Metals

$$Other\ Metals(lb) = Fuel\ Assembly(lb) \times \#Assemblies - [UO_2(lb) + Zirc(lb)]$$

4.2.3 Source Materials (Fuel Assemblies) Volume

$$Volume_{material}(cc) = \frac{Mass_{material}(lb) \times 453.592\left({}^{g}/_{lb}\right)}{Density_{material}\left({}^{g}/_{cc}\right)}$$

4.2.4 Source Water Volume

$$Volume_{water}(cc) = Volume_{source}(cc) - \sum Volume_{materials}(cc)$$

4.3 Source Materials Mixed Shielding Density

The source materials mixed shielding density is calculated in accordance with the guidance provided in MicroShield® user's manual as shown in Attachment 2.

See Attachment 1 for the spreadsheet results for the equations provided below.

4.3.1 Source Material Volume Fractions (MVF)

$$MVF\ (\%) = \frac{Volume_{material}(cc)}{Volume_{source}(cc)}$$

4.3.2 Individual Source Material Density Component (IDC)

$$IDC\ \left({}^{g}/_{cc}\right) = MVF\ (\%) \times Density_{material}\left({}^{g}/_{cc}\right)$$

4.3.3 Effective Source Density (ESD) for MicroShield®

$$ESD_{\left({}^{g}/_{cc}\right)} = \sum IDC_{\left({}^{g}/_{cc}\right)}$$

4.4 MicroShield® Results

The relative dose rate at top of active fuel is developed using MicroShield® with the inputs and interim calculated results described in Sections 3 and 4 above.

Gamma exposure from buildup is selected as the representative MicroShield® result of interest.

Core Surface (water level TOAF)	**3.232E+09 mR/hr**
Containment Dome Peak	**7.628E+05 mR/hr**

Refer to Attachment 3 for the comprehensive results of the MicroShield® relative dose rates at top of active fuel and containment dome peak.

4.5 3561 Radiation Monitor Dose Rates for RCS Level at TOAF

$$3561s\ (R/hr) = RSDR_o\ (R/hr) \times \cos\theta_o \times \frac{A}{c^2} \times \alpha(E_o, \theta_o, \theta, \emptyset\ )$$

Where:	
3561s	Dose rate at the 3561 radiation monitors (R/hr)
RSDR$_o$	MicroShield® dose rate incident on reflector(dome) surface at θ_o (R/hr)
θ_o	Incident angle with respect to the normal (0°)
A	Reflecting area (ft^2)
c	Distance from center of reflecting area to monitor (ft)
α	Dose albedo – $\alpha(E_o, \theta_o, \theta, \Phi)$
	Note – Since the incident angle (θ_o) is 0 determination and use of the azimuth angle (Φ) is not necessary.

Refer to Attachment 4 for the spreadsheet results for the 3561 monitor back-scatter dose rates due to water level at TOAF.

Refer to Attachment 5 for details and inputs from Davisson, "Gamma Ray Dose Albedos."

4.5.1 Reflecting Area (A)

$$A(ft^2) = 2\pi rh$$

Where:	
r	Containment Dome radius (ft)
h	Containment Dome height from springline (ft)

4.5.2 Distance from Reflector Surface at θ_o to Detector (c)

$$c(ft) = \sqrt{a^2 + b^2}$$

4.5.3 Emerging Polar Angle of the Reflector to the Detector (θ)

$$sinA = \frac{a}{c} = \left(\frac{opposite}{hypotenuse}\right)$$

4.5.4 Dose Albedo Energy Groupings (E_o)

The reference albedos are given for incident gamma energies of 0.2, 0.662, 1.0, 2.5 and 6.13 MeV. The MicroShield® result tables provided the photon energies and activities that are used to develop predominant groupings as follows:

Energy (MeV)	Activity (γ/sec)	Activity (%)	Group Abundance (%)
0.015	1.77E+19	17.01%	56.5%
0.02	6.68E+17	0.64%	
0.03	4.79E+18	4.59%	
0.04	2.52E+18	2.42%	
0.05	3.30E+17	0.32%	
0.06	4.23E+17	0.41%	
0.08	1.86E+18	1.78%	
0.1	1.91E+19	18.34%	
0.15	5.54E+18	5.31%	
0.2	5.97E+18	5.73%	
0.3	7.02E+18	6.73%	22.5%
0.4	2.46E+18	2.36%	
0.5	9.04E+18	8.67%	
0.6	4.96E+18	4.76%	
0.8	1.49E+19	14.33%	15.7%
1	1.47E+18	1.41%	
1.5	5.09E+18	4.88%	5.2%
2	1.55E+17	0.15%	
3	1.74E+17	0.17%	
4	2.81E+03	0.00%	
5	6.93E+07	0.00%	
Totals	1.04E+20	100%	

The first grouping is applied to the 0.2 MeV albedo table column. The second grouping is applied to the 0.662 MeV albedo table column. The third grouping is applied to the 1.0 MeV albedo table column. The fourth grouping is applied to the 2.5 MeV albedo table column.

5 CONCLUSION

5.1 Individual Monitor Results

Monitor	mR/hr
RM-1CR-3561A–SA	1.97E+4
RM-1CR-3561B–SB	2.60E+4
RM-1CR-3561C–SA	2.63E+4
RM-1CR-3561D–SB	2.65E+4

5.2 EAL Threshold Values

Based on monitor accuracy/readability, human factors and the similarity of results between radiation monitors, the CS1.3 and CG1.2 EAL threshold bullets are established as follows:

– Any RM-1CR-3561 (A, B, C, or D) containment radiation monitor > 2.6E+4 mR/hr

6 REFERENCES

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

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

6.3. MicroShield® program, Version 7.02 (08-MSD-7.02-1532)

6.4. National Institute of Standards and Technology (NIST) - http://www.physics.nist.gov/cgi-bin/Star/compos.pl?matno=272 - density of Uranium Dioxide (UO_2)

6.5. ANS/SD-75-14, A Handbook of Radiation Shielding Data, July 1976

6.6. HNP Final Safety Analysis Report Update (FSAR)

- Table 4.1.1-1, Original Core Reactor Design Comparison Table, Amendment 61
- Table 12.3.4-1, Area Radiation Monitors, Amendment 61

6.7. FS1-0030999, Adv. W17 HTP™ Mass & Volume Calculation, Revision 2

6.8. Drawing CAR-2165-G-0013, General Arrangement Containment Building, Sheet 1, Revision 17

6.9. Drawing CAR-2166-G-0453, Containment Building Instrument Location Arrangement, Revision 16

6.10. Drawing CAR 1364-001275, Reactor Vessel General Arrangement, Revision 2

Source Height (in):	144
Source Diameter (in):	119.7
Source Radius (in):	59.85
Source Volume (cm3):	**2.66E+07**
Conversion (gm per lb):	4.54E+02

Fuel Assembly dry weight (lbs):	1.50E+03
# Fuel Assemblies:	157
Weight of all fuel assemblies:	2.35E+05
Weight of UO2 and Zirc:	2.23E+05
Weight of other metals:	**1.27E+04**

	Density (g/cc)	**Mass (lb)**	**Mass (g)**	**Volume (cc)**	**Material Volume Fraction - MVF**	**Individual Density Complnent - IDC (g/cc)**
UO_2	10.96	1.81E+05	8.22E+07	7.50E+06	28.2%	3.09
Zirc	6.5	4.14E+04	1.88E+07	2.89E+06	10.9%	0.71
Metal	7.86	1.27E+04	5.77E+06	7.34E+05	2.8%	0.22
Water	1		1.54E+07	1.54E+07	58.1%	0.58
Effective	**4.60**		**1.22E+08**			

5.2 Case Material Screen

Attenuation and buildup (scatter) of radiation between a source and a dose point are affected by all intervening materials. Shield materials determine the radiation attenuation and buildup characteristics used to calculate the dose rate. A library of material characteristics can be used to formulate custom materials. Published attenuation coefficient data from ANSI/ANS-6.4.3-1991, Gamma-Ray Attenuation Coefficients and Buildup Factors for Engineering Materials, from the American Nuclear Society is provided for 100 atomic elements as well as air, water, and concrete.

MicroShield® provides 12 built-in materials and allows adding as many as eight user-defined materials for each shield region. User-defined materials are called "custom" materials. They must first be created with the custom material selection under the tools menu before they are available for shield input. The 12 built-in materials are chosen because they are used often. These materials and their default densities are:

Material	Density (g/cm^3)	Material	Density (g/cm^3)
Air	0.00122	Tin	7.3
Aluminum	2.702	Titanium	4.5
Concrete	2.35	Tungsten	19.3
Iron	7.86	Uranium	18.75
Lead	11.3	Water	1.0
Nickel	8.9	Zirconium	6.5

The default standard density of air is 0.00122 g/cm^3 to be consistent with ANSI/ANS-6.6.1-1979, American National Standard for Calculation and Measurement of Direct and Scattered Gamma Radiation from LWR Nuclear Power Plants, from the American Nuclear Society.

The Case Material Screen in Figure 5-3 allows the user to enter the materials present in the source and all shields. The column headers show the name of the source or shield and its dimension. Each row represents the type of material present. The user can enter any density for any material by clicking on the appropriate cell and entering the density with the keyboard. As well, hitting the space bar or double-clicking will enter the default density of the material.

When more than one material is specified for a shield, the user has two choices to decide the appropriate density. One choice is to pre-calculate the volume fraction or the effective density for each material in the region and use that as an entry. As an example of pre-calculated volume fractions, a source consists of a cylindrical tank of water with steel internal structures. The source region may be represented as a tank with a homogeneous mixture of steel and water. The volume of the tank is 10,000 liters and it contains 8,000 kg of water and 1560 kg of iron. The material densities to enter are then:

$$\text{Water}: \frac{8\text{x}10^6\text{g}}{10\text{x}10^6\text{cm}^3} = 0.8\,\text{g/cm}^3$$

$$\text{Iron}: \frac{1.56\text{x}10^6\text{g}}{10\text{x}10^6\text{cm}^3} = 0.156\,\text{g/cm}^3$$

Another way to estimate densities is to multiply each individual material volume fraction by its pure density. For example, if a volume contains 95% (by volume) water at a density of 1.0 g/cm^3 and 5% (by volume) steel at a density of 7.85 g/cm^3, then the individual density entries are:

Water: 0.95×1.0 g/cm^3 = 0.95 g/cm^3

Iron: 0.05×7.85 g/cm^3 = 0.3925 g/cm^3

Reinforced concrete can similarly be represented with a bulk material iron density representative of the reinforcing rod and the balance being concrete.

A note of caution is appropriate while using custom and mixed materials. The fundamental coefficient data for attenuation and buildup are based on specific materials for which measurements have been made or which have been derived by more precise computing methods than are used in MicroShield®. The method in MicroShield® for combining materials is an approximation and will vary somewhat from the "true" situation. Therefore, results of the use of custom materials and mixed materials should be viewed with caution until a user has confidence in their validity. If field measurements are available, it is suggested that densities be adjusted to "normalize" the results of calculations to measurements.

The energy-dependent linear attenuation coefficients for any region in a specific case may be read from the case file display option on the home menu. Quite often this display is useful for assessing the cause of differences among various dose rate calculations.

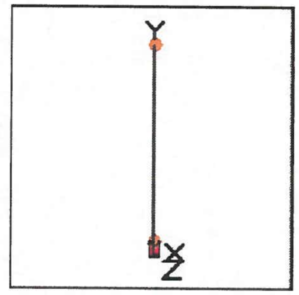
MicroShield 7.02
Duke Energy Corporation (08-MSD-7.02-1532)

Date	By	Checked
06/05/18	William Cerame	Scott McCain

Filename	Run Date	Run Time	Duration
EALCALC 1801 HNP TOAF T72.ms7	June 5, 2018	9:30:25 AM	00:00:00

Project Info	
Case Title	TOAF
Description	72 Hour Decay, TOAF & Top of Dome
Geometry	8 - Cylinder Volume - End Shields

Source Dimensions	
Height	365.76 cm (12 ft)
Radius	152.019 cm (4 ft 11.9 in)



Dose Points			
A	X	Y	Z
#1	0.0 cm (0.0 in)	368.3 cm (12 ft 1.0 in)	0.0 cm (0.0 in)
#2	0.0 cm (0.0 in)	6.2e+3 cm (204 ft)	0.0 cm (0.0 in)

Shields			
Shield N	Dimension	Material	Density
Source	2.66e+07 cm³	Uranium	4.6
Air Gap		Air	0.00122
Wall Clad	10.16 cm	Iron	7.86

Source Input: Grouping Method - Standard Indices
Number of Groups: 25
Lower Energy Cutoff: 0.015
Photons < 0.015: Included
Library: Grove

Nuclide	Ci	Bq	μCi/cm³	Bq/cm³
Ac-225	2.5673e-015	9.4989e-005	9.6679e-017	3.5771e-012
Ac-227				
Am-241	1.6594e+002	6.1397e+012	6.2489e+000	2.3121e+005
At-217				
Ba-137m	9.0610e+006	3.3526e+017	3.4122e+005	1.2625e+010
Ba-139	3.1399e-008	1.1618e+003	1.1824e-009	4.3750e-005
Ba-140	1.1899e+008	4.4027e+018	4.4810e+006	1.6580e+011
Bi-210	3.4441e-015	1.2743e-004	1.2970e-016	4.7989e-012
Bi-211	2.4535e-018	9.0779e-008	9.2393e-020	3.4185e-015
Bi-213				
Bi-214	1.8986e-015	7.0250e-005	7.1499e-017	2.6455e-012
Ce-141	1.2161e+008	4.4997e+018	4.5798e+006	1.6945e+011
Ce-143	2.6007e+007	9.6226e+017	9.7937e+005	3.6237e+010
Ce-144	1.0324e+008	3.8200e+018	3.8879e+006	1.4385e+011

6/5/2018

Note - Microshield measures distance to receptor from the bottom of the source, therefore an additional 12' is added to the "Y" axis to account for core height in the TOAF and containment dome dose rate calculations.

Cm-242	3.2582e+006	1.2055e+017	1.2270e+005	4.5398e+009
Cs-134	1.3862e+007	5.1288e+017	5.2200e+005	1.9314e+010
Cs-135	6.8746e-002	2.5436e+009	2.5888e-003	9.5787e+001
Cs-136	3.7484e+006	1.3869e+017	1.4116e+005	5.2228e+009
Cs-137	9.5782e+006	3.5439e+017	3.6070e+005	1.3346e+010
Cs-138	4.3019e-033	1.5917e-022	1.6200e-034	5.9940e-030
Fr-221				
Fr-223				
I-129	2.6114e-003	9.6623e+007	9.8342e-005	3.6386e+000
I-131	6.2469e+007	2.3114e+018	2.3525e+006	8.7042e+010
I-132	6.0956e+007	2.2554e+018	2.2955e+006	8.4934e+010
I-133	1.4524e+007	5.3740e+017	5.4695e+005	2.0237e+010
I-134	3.3276e-017	1.2312e-006	1.2531e-018	4.6366e-014
I-135	8.0481e+004	2.9778e+015	3.0308e+003	1.1214e+008
Kr-83m	1.2881e-005	4.7660e+005	4.8508e-007	1.7948e-002
Kr-85	8.1975e+005	3.0331e+016	3.0870e+004	1.1422e+009
Kr-85m	2.6430e+002	9.7789e+012	9.9529e+000	3.6826e+005
Kr-87	3.2810e-010	1.2140e+001	1.2356e-011	4.5716e-007
Kr-88	1.1697e+000	4.3281e+010	4.4050e-002	1.6299e+003
La-140	1.3228e+008	4.8942e+018	4.9813e+006	1.8431e+011
La-141	4.0379e+002	1.4940e+013	1.5206e+001	5.6262e+005
La-142	2.8965e-006	1.0717e+005	1.0908e-007	4.0358e-003
Mo-99	7.3253e+007	2.7104e+018	2.7586e+006	1.0207e+011
Nb-93m	1.7431e-005	6.4496e+005	6.5643e-007	2.4288e-002
Nb-95	1.3272e+008	4.9105e+018	4.9979e+006	1.8492e+011
Nb-95m	4.5097e+005	1.6686e+016	1.6983e+004	6.2836e+008
Nb-97	7.0284e+006	2.6005e+017	2.6467e+005	9.7930e+009
Nb-97m	6.1829e+006	2.2877e+017	2.3284e+005	8.6150e+009
Nd-147	4.2697e+007	1.5798e+018	1.6079e+006	5.9492e+010
Np-237	3.3518e-007	1.2402e+004	1.2622e-008	4.6702e-004
Np-239	6.9475e+008	2.5706e+019	2.6163e+007	9.6804e+011
Pa-231	1.8443e-017	6.8240e-007	6.9453e-019	2.5698e-014
Pa-233	8.5135e-009	3.1500e+002	3.2060e-010	1.1862e-005
Pb-209				
Pb-210	3.3174e-015	1.2274e-004	1.2493e-016	4.6223e-012
Pb-211				
Pb-214	1.0151e-015	3.7560e-005	3.8228e-017	1.4144e-012
Pm-147	1.0190e+005	3.7704e+015	3.8374e+003	1.4198e+008
Po-210	2.7135e-015	1.0040e-004	1.0219e-016	3.7809e-012
Po-211	1.2761e-020	4.7216e-010	4.8056e-022	1.7781e-017
Po-213				
Po-214	2.3399e-015	8.6577e-005	8.8117e-017	3.2603e-012
Po-215	2.2776e-020	8.4272e-010	8.5770e-022	3.1735e-017
Po-218	4.9319e-016	1.8248e-005	1.8573e-017	6.8719e-013
Pr-143	1.0885e+008	4.0276e+018	4.0992e+006	1.5167e+011

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Pr-144	1.0325e+008	3.8201e+018	3.8880e+006	1.4386e+011
Pr-144m	1.4764e+006	5.4626e+016	5.5598e+004	2.0571e+009
Pu-238	2.1274e+002	7.8713e+012	8.0113e+000	2.9642e+005
Pu-239	2.6325e+002	9.7403e+012	9.9135e+000	3.6680e+005
Pu-241	1.2595e+007	4.6602e+017	4.7430e+005	1.7549e+010
Ra-223				
Ra-225				
Ra-226	1.0042e-015	3.7156e-005	3.7817e-017	1.3992e-012
Rb-86	1.3955e+005	5.1633e+015	5.2552e+003	1.9444e+008
Rb-87	1.1133e-007	4.1193e+003	4.1925e-009	1.5512e-004
Rb-88	1.3062e+000	4.8329e+010	4.9189e-002	1.8200e+003
Rh-103m	1.2121e+008	4.4849e+018	4.5646e+006	1.6889e+011
Rh-105	2.3345e+007	8.6377e+017	8.7913e+005	3.2528e+010
Rh-105m	2.9109e+002	1.0770e+013	1.0962e+001	4.0559e+005
Rh-106	4.5443e+007	1.6814e+018	1.7113e+006	6.3318e+010
Rn-219				
Rn-222	4.3294e-016	1.6019e-005	1.6304e-017	6.0324e-013
Ru-103	1.2141e+008	4.4922e+018	4.5721e+006	1.6917e+011
Ru-105	1.1848e+003	4.3837e+013	4.4617e+001	1.6508e+006
Ru-106	4.5443e+007	1.6814e+018	1.7113e+006	6.3317e+010
Sb-127	4.1079e+006	1.5199e+017	1.5470e+005	5.7237e+009
Sb-129	3.0359e+002	1.1233e+013	1.1433e+001	4.2301e+005
Sm-147	2.8239e-009	1.0449e+002	1.0634e-010	3.9347e-006
Sr-89	6.8139e+007	2.5211e+018	2.5660e+006	9.4941e+010
Sr-90	7.0486e+006	2.6080e+017	2.6544e+005	9.8212e+009
Sr-91	4.6391e+005	1.7165e+016	1.7470e+004	6.4639e+008
Sr-92	9.5975e-001	3.5511e+010	3.6142e-002	1.3373e+003
Tc-99	3.1115e+000	1.1512e+011	1.1717e-001	4.3354e+003
Tc-99m	7.1409e+007	2.6421e+018	2.6891e+006	9.9497e+010
Te-127	4.9399e+006	1.8278e+017	1.8603e+005	6.8830e+009
Te-127m	1.1653e+006	4.3115e+016	4.3882e+004	1.6236e+009
Te-129	2.9451e+006	1.0897e+017	1.1091e+005	4.1035e+009
Te-129m	4.6749e+006	1.7297e+017	1.7605e+005	6.5137e+009
Te-131	6.7819e+005	2.5093e+016	2.5539e+004	9.4496e+008
Te-131m	3.0125e+006	1.1146e+017	1.1344e+005	4.1974e+009
Te-132	5.9164e+007	2.1891e+018	2.2280e+006	8.2436e+010
Th-227				
Th-229				
Th-230	6.1699e-014	2.2829e-003	2.3235e-015	8.5969e-011
Th-231	5.4638e-010	2.0216e+001	2.0576e-011	7.6130e-007
Tl-207	6.5221e-019	2.4132e-008	2.4561e-020	9.0875e-016
Tl-209				
U-233	2.6642e-017	9.8575e-007	1.0033e-018	3.7122e-014
U-234	2.4821e-006	9.1838e+004	9.3471e-008	3.4584e-003
U-235	1.2194e-009	4.5119e+001	4.5922e-011	1.6991e-006

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U-237	8.1829e+001	3.0277e+012	3.0815e+000	1.1402e+005
Xe-131m	1.0289e+006	3.8071e+016	3.8748e+004	1.4337e+009
Xe-133	1.2684e+008	4.6931e+018	4.7765e+006	1.7673e+011
Xe-133m	2.8555e+006	1.0565e+017	1.0753e+005	3.9787e+009
Xe-135	1.6626e+006	6.1515e+016	6.2609e+004	2.3165e+009
Xe-135m	1.3814e+004	5.1114e+014	5.2023e+002	1.9248e+007
Xe-138	1.2372e-084	4.5778e-074	4.6592e-086	1.7239e-081
Y-90	7.1276e+006	2.6372e+017	2.6841e+005	9.9313e+009
Y-91	9.0815e+007	3.3602e+018	3.4199e+006	1.2654e+011
Y-91m	2.9173e+005	1.0794e+016	1.0986e+004	4.0648e+008
Y-92	3.0442e+002	1.1264e+013	1.1464e+001	4.2417e+005
Y-93	5.3092e+005	1.9644e+016	1.9994e+004	7.3976e+008
Zr-93	5.5552e-002	2.0554e+009	2.0920e-003	7.7404e+001
Zr-95	1.2681e+008	4.6921e+018	4.7755e+006	1.7670e+011
Zr-97	6.5225e+006	2.4133e+017	2.4563e+005	9.0881e+009

Buildup: The material reference is Source

Integration Parameters	
Radial	20
Circumferential	10
Y Direction (axial)	10

Results - Dose Point # 1 - (0,368.3,0) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.015	1.773e+19	8.708e-28	8.572e-15	7.469e-29	7.352e-16
0.02	6.684e+17	5.747e-33	4.308e-16	1.991e-34	1.492e-17
0.03	4.787e+18	1.711e-15	6.393e-15	1.696e-17	6.336e-17
0.04	2.519e+18	1.200e-03	1.236e-03	5.306e-06	5.465e-06
0.05	3.295e+17	1.160e+01	1.205e+01	3.089e-02	3.211e-02
0.06	4.232e+17	3.564e+03	3.744e+03	7.079e+00	7.436e+00
0.08	1.858e+18	3.333e+06	3.597e+06	5.275e+03	5.693e+03
0.1	1.911e+19	5.965e+08	6.597e+08	9.126e+05	1.009e+06
0.15	5.538e+18	5.448e+07	3.407e+08	8.971e+04	5.610e+05
0.2	5.969e+18	1.154e+09	1.750e+09	2.037e+06	3.088e+06
0.3	7.018e+18	1.442e+10	1.707e+10	2.736e+07	3.239e+07
0.4	2.455e+18	1.399e+10	1.625e+10	2.727e+07	3.166e+07
0.5	9.041e+18	9.747e+10	1.145e+11	1.913e+08	2.247e+08
0.6	4.957e+18	8.513e+10	1.013e+11	1.662e+08	1.977e+08
0.8	1.493e+19	4.972e+11	6.165e+11	9.458e+08	1.173e+09
1.0	1.474e+18	7.832e+10	9.997e+10	1.444e+08	1.843e+08
1.5	5.086e+18	5.645e+11	7.392e+11	9.497e+08	1.244e+09
2.0	1.547e+17	2.599e+10	3.472e+10	4.018e+07	5.369e+07
3.0	1.743e+17	4.777e+10	6.396e+10	6.481e+07	8.677e+07
4.0	2.813e+03	1.037e-03	1.367e-03	1.283e-06	1.691e-06

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5.0	6.929e+07	3.136e+01	4.238e+01	3.595e-02	4.859e-02
Totals	**1.042e+20**	**1.427e+12**	**1.806e+12**	**2.560e+09**	**3.232e+09**

Results - Dose Point # 2 - (0,6217.92,0) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.015	1.773e+19	2.541e-33	5.129e-18	2.179e-34	4.399e-19
0.02	6.684e+17	9.344e-36	2.578e-19	3.237e-37	8.930e-21
0.03	4.787e+18	2.994e-18	5.823e-18	2.967e-20	5.771e-20
0.04	2.519e+18	5.567e-07	5.734e-07	2.462e-09	2.536e-09
0.05	3.295e+17	3.114e-03	3.237e-03	8.296e-06	8.624e-06
0.06	4.232e+17	7.171e-01	7.540e-01	1.424e-03	1.498e-03
0.08	1.858e+18	3.174e+02	3.417e+02	5.023e-01	5.407e-01
0.1	1.911e+19	2.313e+04	2.548e+04	3.539e+01	3.897e+01
0.15	5.538e+18	4.388e+03	1.947e+04	7.226e+00	3.206e+01
0.2	5.969e+18	3.936e+04	5.995e+04	6.947e+01	1.058e+02
0.3	7.018e+18	1.309e+06	1.634e+06	2.483e+03	3.100e+03
0.4	2.455e+18	2.126e+06	2.663e+06	4.143e+03	5.189e+03
0.5	9.041e+18	1.748e+07	2.228e+07	3.430e+04	4.374e+04
0.6	4.957e+18	1.636e+07	2.120e+07	3.193e+04	4.139e+04
0.8	1.493e+19	1.030e+08	1.398e+08	1.960e+05	2.658e+05
1.0	1.474e+18	1.708e+07	2.388e+07	3.149e+04	4.401e+04
1.5	5.086e+18	1.339e+08	1.904e+08	2.253e+05	3.203e+05
2.0	1.547e+17	6.494e+06	9.364e+06	1.004e+04	1.448e+04
3.0	1.743e+17	1.270e+07	1.813e+07	1.723e+04	2.459e+04
4.0	2.813e+03	2.858e-07	3.970e-07	3.535e-10	4.912e-10
5.0	6.929e+07	8.844e-03	1.258e-02	1.014e-05	1.442e-05
Totals	**1.042e+20**	**3.106e+08**	**4.294e+08**	**5.530e+05**	**7.628e+05**

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Radiation Monitor	Elevation	Detector to Core CL (a)	Detector to Dome (b)	Dome to Detector (c)	Angle (degrees)	Angle (radians)	Energy Category (MeV)	Energy Group (fraction)	Dose Albedo (α)	Fractional Dose (mR/hr)	Total Dose (mR/hr)
RM-01CR-3561ASA	289.7	62	151.3	163.5	22.28	0.39	0.2	0.57	0.0371	15901.68	**1.97E+04**
							0.6	0.23	0.0145	2475.11	
							1.00	0.16	0.0100	1187.58	
							2.50	0.05	0.0045	176.85	
RM-01CR-3561BSB	289.5	26	151.5	153.7	9.74	0.17	0.2	0.57	0.0432	20949.96	**2.60E+04**
							0.6	0.23	0.0172	3326.36	
							1.00	0.16	0.0114	1544.21	
							2.50	0.05	0.0045	198.68	
RM-01CR-3561CSA	289.5	20	151.5	152.8	7.52	0.13	0.2	0.57	0.0432	21197.56	**2.63E+04**
							0.6	0.23	0.0172	3365.67	
							1.00	0.16	0.0114	1562.46	
							2.50	0.05	0.0045	201.03	
RM-01CR-3561DSB	289.5	16	151.5	152.3	6.03	0.11	0.2	0.57	0.0432	21329.09	**2.65E+04**
							0.6	0.23	0.0172	3386.55	
							1.00	0.16	0.0114	1572.15	
							2.50	0.05	0.0045	202.28	

Top of Active Fuel (ele ft):	249
Spring Line (ele ft):	376
Dome Radius (ft):	65
Dome Peak (ele ft):	441
TOAF to Dome (ft):	192

Dome Hemisphere Reflecting Area (ft2):	2.65E+04

Dome Peak Dose Rate (mR/hr):	7.63E+05
Incident Angle - θo (radians):	0
Cos θo	1

ANS/SD–76/14

A HANDBOOK OF

RADIATION SHIELDING DATA

J. C. COURTNEY, EDITOR

Sponsored by:

Nuclear Science Center
Louisiana State University
Baton Rouge
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Shielding and Dosimetry Division
American Nuclear Society

JULY, 1976

Gamma Ray Dose Albedos

C. M. Davisson
U. S. Naval Research Laboratory

The dose rate reflected from a surface as deduced from Reference 1 through 4 may be represented as:

$$D.R. = D.R._o \cos\theta_o \frac{A}{r^2} \alpha(E_o, \theta_o, \theta, \phi)$$

where

D.R. = Reflected dose rate

$D.R._o$ = Dose rate incident on surface at angle θ_o

A = Reflecting area

r = Distance from center of reflecting area to receptor (A and r^2 must be in the same units)

$\alpha(E_o\ \theta_o, \theta, \phi)$ = Dose albedo

The albedos, $\alpha(E_o, \theta_o, \theta, \phi)$, for gammas incident on water, concrete, iron and lead have been calculated by C. M. Davisson and L. A. Beach[5] using Monte Carlo techniques in an extension of the original work by Theus and Beach[6]. The albedos are given for incident gamma energies of 0.2, 0.662, 1.0, 2.5 and 6.13 MeV and for incident angles with respect to the normal of 0^o, 22^o, 44^o, 66^o and 88^o, as well as for point sources on the surface of the materials. The emerging polar angles, θ_i, as well as the emerging sectors or directions into which the emerging gammas were divided are shown in Fig. 5.13. The values of the polar angles, θ_i, and of the azimuthal angles ϕ, defining the emerging directions, Ω_k, are given on each page of Table 5.8.

Note: The dose albedo values have statistical errors that range from 40% or 50% at very small albedo values to 5% or 10% at large albedo values.

References

[1] Reactor Shielding Design Manual, T. Rockwell III, editor, TID-7004 (March 1956) p. 334.

[2] D. J. Raso, "Monte Carlo Calculations on the Reflection and Transmission of Scattered Gamma Rays," Nucl. Sci. and Eng. 17, 411 (1963). This report has a good discussion of the meaning of various terms and derived quantities. The dose albedos given here are those which he described in quotes, as "dose" albedos.

[3] W. E. Selph, "Neutrons and Gamma-Ray Albedos," DASA-1892-2 (May 1967), ORNL-RSIC-21 (February 1968), or Chapter 4 of Weapons Radiation Shielding Handbook (NTIS No. AD-816 092). The dose albedos given here are those defined as α_{D2} in this report.

[4] R. L. French and M. B. Wells, "An-Angle-Dependent Albedo for Fast-Neutron Reflection Calculations," Nucl. Sci. and Eng. 19, 441 (1964).

[5] C. M. Davisson and L. A. Beach, "Gamma-Ray Albedos of Iron," NRL Quarterly on Nucl. Sci. and Tech. (January 1, 1960), p. 43; and private communication.

[6] R. B. Theus and L. A. Beach, "Gamma-Ray Albedo," NRL Quarterly on Nucl. Sci. and Tech. (July-September 1955).

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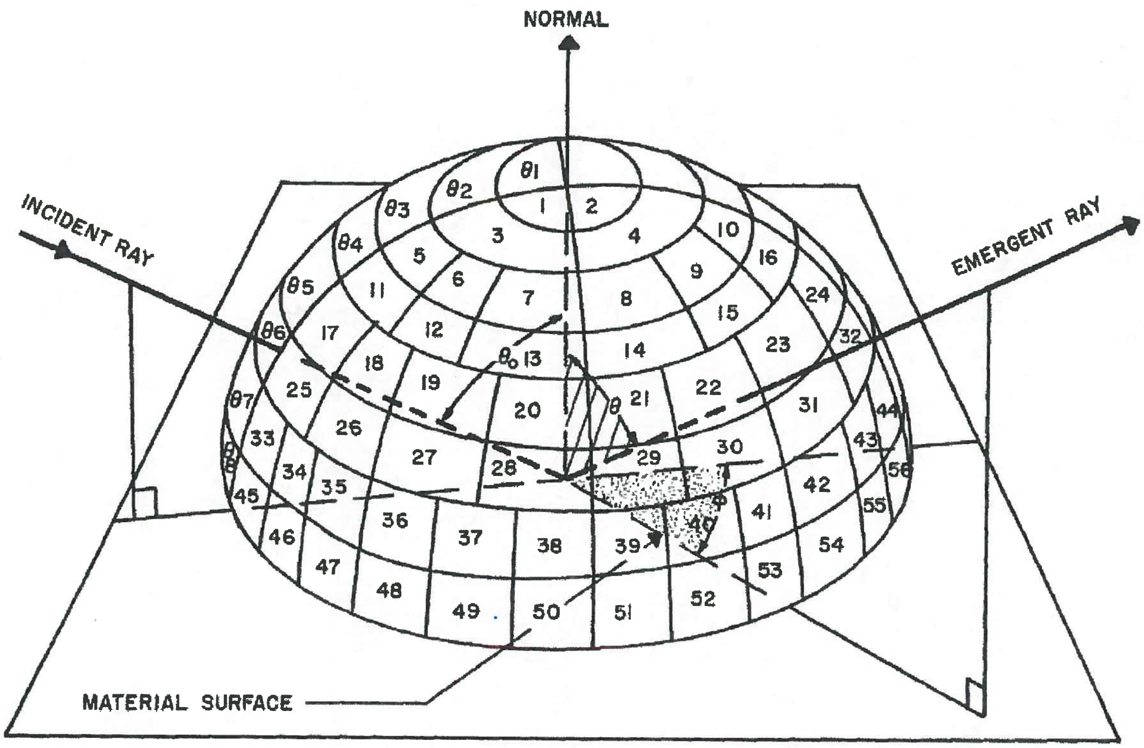
Figure 5.13



Geometry and Solid Angle Divisions

Table 5.8 continued

GAMMA RAY DOSE ALBEDOS
(in percent)

Emerging Polar Angle θ_i	Emerging Direction Ω_k	φ	Water 6.13 MeV incident at 0° *	22°	44°	66°	88°	Point * Source	Concrete + 0.2 MeV incident at 0° *	22°	44°	66°	88°	Point * Source
θ_1 0.0-15.4	1	90.0-180.0	.2655	.3096	.3638	.6899	1.0608	.6976	4.5232	4.4243	4.7637	5.3214	5.9071	5.1588
	2	0.0- 90.0	±.0341	.3354	.4794	.7897	1.3721	±.0161	±.1273	3.8358	4.5791	6.1261	6.0659	±.2651
θ_2 15.4-21.8	3	90.0-180.0	.2578	.2447	.3884	.6006	.9558	.6227	4.1008	4.3431	4.3651	5.0983	5.5753	5.2882
	4	0.0- 90.0	±.0025	.2420	.3649	.6566	1.4205	±.0108	±.0541	4.1439	4.8685	5.3798	6.7652	±.0430
θ_3 21.8-34.8	5	150.0-180.0		.2879	.2932	.5174	.9825			3.9924	4.6116	4.7018	5.8179	
	6	120.0-150.0		.2227	.3198	.5674	.9718			3.8218	4.5287	4.9943	5.1744	
	7	90.0-120.0	.8308	.3155	.3846	.5157	1.0534	.6527	3.7130	4.0026	4.2369	5.4383	6.0965	5.4233
	8	60.0- 90.0	±.0078	.2423	.3401	.5861	1.2751	±.0194	±.0882	3.6716	4.1899	5.0787	6.1243	±.1175
	9	30.0- 60.0		.3316	.4760	.7639	1.6719			4.2174	4.5325	6.1074	9.1190	
	10	0.0- 30.0		.3057	.5149	.8429	2.0850			4.0847	4.3484	5.9747	8.1129	
θ_4 34.8-44.4	11	150.0-180.0		.2141	.2669	.4442	.6686			3.5770	4.1608	4.6135	4.6975	
	12	120.0-150.0		.2173	.2873	.4324	.7498			3.5108	3.7679	5.2446	5.6307	
	13	90.0-120.0	.2286	.2320	.3494	.6564	.9466	.6696	3.4657	3.4567	4.1646	4.9897	6.1642	5.0246
	14	60.0- 90.0	±.0119	.2820	.3575	.7095	1.2473	±.0176	±.0412	3.4292	3.9545	5.2921	6.0036	±.0490
	15	30.0- 60.0		.2583	.3974	.8255	1.9624			3.2757	4.3277	5.7964	8.9277	
	16	0.0- 30.0		.2636	.4196	1.0195	3.0115			4.0912	4.3774	6.1438	10.6385	
θ_5 44.4-55.2	17	157.5-180.0		.1610	.2355	.3074	.8453			3.2055	3.4274	4.6624	5.4600	
	18	135.0-157.5		.1759	.1988	.3244	.6713			3.1530	3.1680	4.3647	5.2342	
	19	112.5-135.0		.2036	.2619	.5201	.8712			2.8823	3.6708	4.3318	4.7637	
	20	90.0-112.5	.2080	.2101	.3028	.4606	.8563	.6972	2.9762	3.2066	3.6614	4.3557	5.5118	4.7362
	21	67.5- 90.0	±.0059	.2270	.2649	.4800	1.3248	±.0308	±.1024	3.0541	3.4667	4.6505	5.7264	±.1002
	22	45.0- 67.5		.2187	.3081	.7226	1.8876			3.3882	3.9259	5.5386	7.6715	
	23	22.5- 45.0		.2429	.3885	1.1044	3.2042			3.3180	4.1492	6.0254	10.5245	
	24	0.0- 22.5		.2118	.4728	1.0819	5.1885			2.9231	4.5731	6.9974	12.4259	
θ_6 55.2-64.6	25	157.5-180.0		.1590	.1479	.3316	.6101			2.2100	3.1600	3.9595	4.4952	
	26	135.0-157.5		.1393	.1902	.2893	.6823			2.5306	2.9034	3.9702	4.9446	
	27	112.5-135.0		.1558	.2492	.3812	.7071			2.2988	2.7043	3.4321	5.3944	
	28	90.0-112.5	.1770	.1314	.1754	.4116	.8769	.7172	2.4186	2.6234	3.1339	3.9496	5.4647	4.7027
	29	67.5- 90.0	±.0071	.1785	.2529	.5295	1.1431	±.0321	±.0393	2.4971	3.2553	4.1030	6.5517	±.0994
	30	45.0- 67.5		.1433	.2435	.6470	2.0865			2.7271	2.8219	4.6986	7.4996	
	31	22.5- 45.0		.1769	.4152	1.0312	4.2078			2.4802	3.1664	6.2828	12.8378	
	32	0.0- 22.5		.1957	.3689	1.3812	8.2644			2.6657	3.9543	7.0802	15.7200	
θ_7 64.6-77.6	33	165.0-180.0		.1096	.0873	.1655	.5455			1.6948	2.0742	2.8414	4.3220	
	34	150.0-165.0		.0813	.1116	.1816	.4830			1.5881	1.9194	2.5071	4.4477	
	35	135.0-150.0		.1187	.1472	.2569	.4355			1.9763	1.9715	2.6430	4.8948	
	36	120.0-135.0		.1194	.1622	.2040	.5520			1.5987	1.9533	2.7197	4.1239	
	37	105.0-120.0		.1036	.1193	.2179	.6518			1.8348	2.2759	2.9081	4.8545	
	38	90.0-105.0	.0982	.1021	.1617	.2629	.7543	.8353	1.6023	1.7625	1.8482	2.7899	4.7097	2.8096
	39	75.0- 90.0	±.0037	.1226	.1940	.3141	1.4250	±.0357	±.0253	1.5841	2.2123	3.0426	5.7263	±.0696
	40	60.0- 75.0		.1098	.1489	.4164	1.4403			1.6523	2.4310	3.4018	6.6971	
	41	45.0- 60.0		.0867	.2014	.5064	2.4667			1.7458	2.5793	3.5082	8.7597	
	42	30.0- 45.0		.1103	.2846	.7999	4.9109			1.7666	2.7971	4.0234	13.1884	
	43	15.0- 30.0		.1021	.2704	1.2737	10.3622			1.8128	3.0706	5.8535	14.9223	
	44	0.0- 15.0		.1545	.3172	1.8039	20.2675			2.1231	3.9404	6.4699	16.8259	
θ_8 77.6-90.0	45	165.0-180.0		.0530	.0260	.0508	.2528			.3761	.6508	.9831	2.6020	
	46	150.0-165.0		.0180	.0438	.0415	.2679			.5185	.6061	.9309	2.6709	
	47	135.0-150.0		.0226	.0314	.0606	.2907			.5987	.7263	.8010	3.0183	
	48	120.0-135.0		.0157	.0282	.0654	.2663			.5704	.5103	.8820	2.8103	
	49	105.0-120.0		.0315	.0434	.0784	.3243			.5234	.7247	.9878	2.5319	
	50	90.0-105.0	.0327	.0229	.0361	.0813	.3970	.8457	.5243	.5707	.8062	1.3871	2.8380	1.9840
	51	75.0- 90.0	±.0031	.0484	.0960	.0652	.5842	±.0458	±.0326	.6815	.9439	1.3997	3.3985	±.0946
	52	60.0- 75.0		.0308	.0870	.1249	1.0010			.7262	.9293	1.1153	4.2514	
	53	45.0- 60.0		.0603	.0630	.2678	1.4694			.5681	1.0523	2.0029	6.5419	
	54	30.0- 45.0		.0605	.1041	.3706	3.9511			.7038	.9577	2.0887	9.7498	
	55	15.0- 30.0		.0580	.1441	.8033	11.3771			.6632	1.3774	3.1693	12.9003	
	56	0.0- 15.0		.0346	.2239	1.5243	38.5024			.6344	1.3625	3.1294	13.1635	
Sum over Ω			8.4613	9.0476	13.6134	30.7243	158.8846	42.0609	128.6133	135.3857	164.2352	227.3906	386.2157	227.1319
Total Dose Albedos			.9494	1.0151	1.5274	3.4360	17.8268	4.7192	14.4304	15.1002	18.4271	25.5138	43.4118	25.4847

*Symmetrical sources, so Ω values averaged.
+For composition see following page

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Table 5.8 continued

GAMMA RAY DOSE ALBEDOS
(in percent)

Emerging Polar Angle θ_i	Emerging Direction Ω_k	φ	Concrete † 0.662 MeV incident at 0° *	22°	44°	66°	88°	Point * Source	Concrete † 1.00 MeV incident at 0° *	22°	44°	66°	88°	Point * Source
θ_1 0.0-15.4	1	90.0-180.0	1.7235	1.7597	2.2777	2.6810	3.3634	2.6836	1.1448	1.1035	1.4476	2.1105	2.5946	1.9543
	2	0.0- 90.0	±.0667	1.7050	2.0660	2.8851	3.8974	±.1314	±.0808	1.1460	1.5772	2.2313	3.1876	±.0188
θ_2 15.4-21.8	3	90.0-180.0	1.6116	1.7130	1.9468	2.5441	2.9454	2.5246	1.0172	1.1160	1.4448	2.1496	2.3893	2.0806
	4	0.0- 90.0	±.0286	1.9284	2.2575	3.1203	4.4469	±.1440	±.0418	1.1929	1.5012	2.4402	3.6558	±.0657
θ_3 21.8-34.8	5	150.0-180.0	1.4511	1.6153	1.6792	2.4838	2.7653	2.6652	.9962	1.0610	1.1343	1.5255	1.8918	1.9928
	6	120.0-150.0	±.0231	1.3958	1.8732	2.4549	2.6187	±.0580	±.0205	.8699	1.1947	1.7591	1.9574	±.0685
	7	90.0-120.0		1.6171	1.7695	2.4321	3.1994			.9827	1.3857	1.9154	2.6979	
	8	60.0- 90.0		1.7327	2.1598	2.9783	4.0218			1.1299	1.3562	2.1449	3.2901	
	9	30.0- 60.0		1.7565	2.2596	3.7876	5.5501			1.2984	1.6173	2.5944	4.5625	
	10	0.0- 30.0		1.6593	2.6210	3.9158	6.2732			1.1968	1.5970	3.0886	5.1807	
θ_4 34.8-44.4	11	150.0-180.0	1.4056	1.3301	1.5202	2.0564	2.4478	2.7485	.5650	.9734	1.0381	1.4437	1.7974	2.0672
	12	120.0-150.0	±.0588	1.3777	1.5698	2.1031	2.4006	±.0430	±.0273	.9362	1.0746	1.4839	1.8761	±.0390
	13	90.0-120.0		1.5141	1.8149	2.4931	2.8279			1.0161	1.3523	1.9556	2.1826	
	14	60.0- 90.0		1.5568	1.8687	2.6667	3.8192			.9334	1.5487	2.2760	2.9461	
	15	30.0- 60.0		1.5502	2.1703	3.6205	6.8377			1.1098	1.6111	2.8121	5.3828	
	16	0.0- 30.0		1.6353	2.4504	4.3106	8.7624			1.1294	1.8046	3.5768	7.5430	
θ_5 44.4-55.2	17	157.5-180.0	1.1750	1.1835	1.3731	1.5480	2.1925	2.7133	.7662	.8224	.8356	1.1753	1.5603	2.1904
	18	135.0-157.5	±.0370	1.3368	1.3392	1.7286	2.1881	±.1031	±.0304	.7883	.8735	1.2353	1.6847	±.0728
	19	112.5-135.0		1.1268	1.5658	2.1548	2.3631			.7727	1.0432	1.4835	1.8523	
	20	90.0-112.5		1.2916	1.5131	2.1287	2.8125			.7836	1.1264	1.4884	2.3154	
	21	67.5- 90.0		1.4411	1.7658	2.5523	3.9315			.8495	1.2386	2.0937	3.3244	
	22	45.0- 67.5		1.2726	2.1503	2.8475	5.8450			.9540	1.5370	2.3985	4.9460	
	23	22.5- 45.0		1.4413	2.1872	4.4856	8.5907			1.0820	1.7867	3.4487	7.8503	
	24	0.0- 22.5		1.4266	2.5701	4.6773	11.8869			1.0938	1.8370	4.1599	10.6785	
θ_6 55.2-64.6	25	157.5-180.0	.3512	.8503	1.0790	1.5766	1.8279	2.7574	.6573	.6010	.6568	1.0369	1.3170	2.2644
	26	135.0-157.5	±.0321	1.0381	.9941	1.4919	2.1053	±.0526	±.0153	.6918	.7157	1.1205	1.5499	±.0577
	27	112.5-135.0		.9003	1.3028	1.5000	2.5272			.6449	.6550	1.1177	1.8813	
	28	90.0-112.5		1.0842	1.3038	1.9003	2.8284			.7079	.8658	1.2360	2.2479	
	29	67.5- 90.0		.9888	1.4045	2.2988	4.2218			.7860	.8235	1.7611	3.2118	
	30	45.0- 67.5		1.1252	1.7101	3.1901	5.7558			.8350	1.3643	2.5075	4.8773	
	31	22.5- 45.0		1.1565	2.0752	4.8711	10.1097			.8927	1.7029	3.6096	8.3203	
	32	0.0- 22.5		1.2092	2.3217	5.4436	16.1340			.7734	1.9016	4.1876	16.4409	
θ_7 64.6-77.6	33	165.0-180.0	.6804	.5877	.6933	.8220	1.6082	2.4288	.4664	.4065	.4165	.6229	1.0858	2.0862
	34	150.0-165.0	±.0179	.5334	.7990	.9752	1.6332	±.0668	±.0200	.3368	.5243	.7396	1.2279	±.0646
	35	135.0-150.0		.7098	.6596	1.0530	1.8298			.3941	.4823	.6368	1.2729	
	36	120.0-135.0		.5552	.6505	1.1433	1.9709			.4061	.5116	.7718	1.5392	
	37	105.0-120.0		.7537	.7829	1.0284	2.2125			.4571	.5713	.8122	1.7477	
	38	90.0-105.0		.7957	.8236	1.3800	2.7667			.4517	.6219	1.1266	2.2470	
	39	75.0- 90.0		.8331	1.1684	1.7906	3.7156			.6184	.8151	1.4093	2.9293	
	40	60.0- 75.0		.9290	1.1384	2.0584	4.7747			.6194	.8823	1.8400	4.5087	
	41	45.0- 60.0		.8947	1.2641	2.8866	6.7246			.6021	.9909	2.5313	6.3592	
	42	30.0- 45.0		.9338	1.6641	3.9088	11.9356			.8603	1.1384	3.1438	10.3826	
	43	15.0- 30.0		.8473	1.8589	4.9260	17.0101			.5516	1.4438	4.1275	17.9330	
	44	0.0- 15.0		1.1750	1.8808	5.8095	21.3304			.8188	1.8096	5.2776	22.8816	
θ_8 77.6-90.0	45	165.0-180.0	.2413	.2448	.2186	.4014	.9673	1.4411	.1836	.1193	.1388	.2121	.6108	1.3139
	46	150.0-165.0	±.0135	.2263	.2957	.3154	.9256	±.0442	±.0187	.1400	.2011	.2152	.5935	±.0702
	47	135.0-150.0		.2449	.2043	.4096	1.0372			.1661	.1409	.3004	.7449	
	48	120.0-135.0		.2414	.2622	.4143	1.4216			.1372	.1357	.2395	.9284	
	49	105.0-120.0		.1778	.3527	.4708	1.3059			.1283	.1785	.3337	.9934	
	50	90.0-105.0		.2961	.3281	.5010	1.2886			.1650	.3153	.4278	1.1835	
	51	75.0- 90.0		.2145	.3910	.7054	2.1334			.1201	.2810	.4867	1.6142	
	52	60.0- 75.0		.3894	.4801	.8190	3.5707			.2264	.3777	.6724	2.8628	
	53	45.0- 60.0		.2618	.5859	1.3892	5.4957			.2041	.3833	1.3637	4.4470	
	54	30.0- 45.0		.3328	.6877	1.7761	8.5488			.2315	.6056	1.6301	8.3207	
	55	15.0- 30.0		.4218	.9078	2.9412	15.0340			.3005	.7342	2.4865	17.0845	
	56	0.0- 15.0		.4069	1.1469	3.2682	19.3449			.3290	.8140	3.1314	22.1170	
Sum over Ω			52.0000	97.7278	78.3301	132.3333	290.2237	133.1027	35.2958	39.0749	56.3430	103.9138	253.5002	108.8888
Total Dose Albedos			5.8344	6.4770	8.7886	14.8478	32.5630	14.9341	3.8602	4.3842	6.3217	11.6591	29.5647	12.2173

*Symmetrical sources, so Ω values averaged
†Composition in percent by weight: O 52.9, Si 33.7, Ca 4.4, Al 3.4, Na 1.6, Fe 1.4, K 1.3, H 1.0, Mg 0.2, C 0.1

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Table 5.8 continued

GAMMA RAY DOSE ALBEDOS
(in percent)

Emerging Polar Angle θ_i	Ω_k	φ	Concrete † 2.50 MeV incident at 0° *	22°	44°	66°	88°	Point * Source	Concrete † 6.13 MeV incident at 0° *	22°	44°	66°	88°	Point * Source
θ_1 0.0-15.4	1	90.0-180.0	.4462	.5148	.6759	1.0003	1.3063	1.0800	.3499	.3667	.3845	.6516	1.0531	.7128
	2	0.0- 90.0	±.0070	.4859	.7357	1.2620	1.9701	±.0052	±.0226	.4085	.4996	.7087	1.5861	±.0297
θ_2 15.4-21.8	3	90.0-180.0	.4780	.4676	.6778	1.0381	1.3095	1.0300	.3163	.3511	.4096	.6215	1.0793	.6370
	4	0.0- 90.0	±.0226	.5108	.7309	1.2172	2.4493	±.0620	±.0110	.3345	.4902	.8081	1.3683	±.0066
	5	150.0-180.0		.4205	.5640	.8719	1.0862			.3565	.4497	.6084	1.0506	
	6	120.0-150.0		.4827	.5944	.8105	1.3255			.2761	.3845	.5172	.9892	
θ_3 21.8-34.8	7	90.0-120.0	.4494	.5328	.5699	.8734	1.5293	1.0703	.2808	.3001	.3807	.6222	1.3224	.7203
	8	60.0- 90.0	±.0183	.4903	.6540	1.2638	2.1777	±.0472	±.0180	.3842	.4199	.7134	1.4991	±.[illegible]
	9	30.0- 60.0		.5720	.6978	1.4007	2.6867			.2721	.4565	.9157	1.7594	
	10	0.0- 30.0		.4765	.8064	1.7671	3.1494			.3153	.5229	.9915	2.[illegible]91	
	11	150.0-180.0		.3825	.4647	.6678	1.0237			.2553	.3136	.5023	.7071	
	12	120.0-150.0		.4319	.5244	.6590	1.1345			.3280	.3068	.4117	.8657	
θ_4 34.8-44.4	13	90.0-120.0	.4220	.4354	.5830	1.0458	1.4102	1.1824	.2682	.3077	.3912	.6519	1.0627	.8172
	14	60.0- 90.0	±.0160	.5155	.6061	1.2208	1.6337	±.0566	±.0117	.3108	.4562	.7[illegible]	1.4262	±.0464
	15	30.0- 60.0		.5359	.8370	1.5657	3.6244			.2991	.4638	.8917	2.1442	
	16	0.0- 30.0		.5490	.8328	1.9276	5.2219			.3680	.5292	1.[illegible]	3.0392	
	17	157.5-180.0		.3172	.4169	.5710	1.1315			.2482	.3620	.4[illegible]5	.9944	
	18	135.0-157.5		.3449	.4000	.6718	.3498			.2318	.2974	[illegible]	.8879	
	19	112.5-135.0		.3051	.4446	.7313	1.2498			.2704	.2759	.4492	.[illegible]5	
θ_5 44.4-55.2	20	90.0-112.5	.3518	.4068	.4990	.8106	1.3755	1.3021	.2495	.3587	.3319	.5546	1.1210	.7944
	21	67.5- 90.0	±.0134	.4457	.5978	1.0904	2.0103	±.0242	±.[illegible]	.3457	.3975	.[illegible]95	1.2483	±.[illegible]
	22	45.0- 67.5		.3681	.7345	1.3173	2.9517			.2201	.3919	.7891	2.1832	
	23	22.5- 45.0		.4605	.7405	1.8964	5.7301			.2648	.4431	1.[illegible]59	3.2827	
	24	0.0- 22.5		.4361	.9473	2.4680	7.9027			.2473	.4927	1.1354	4.9014	
	25	157.5-180.0		.2541	.2768	.4512	.8019			.1475	.1391	.2491	.4234	
	26	135.0-157.5		.3102	.3117	.5048	.9731			.1597	.1929	.2[illegible]	.8459	
	27	112.5-135.0		.3246	.4048	.5804	.9963			.2287	.2575	.2246	.8509	
θ_6 55.2-64.6	28	90.0-112.5	.2830	.3088	.4149	.6[illegible]2	1.2556	1.3295	.1853	.1837	.23[illegible]	.4177	.9229	.7738
	29	67.5- 90.0	±.0105	.3342	.4816	1.0276	1.8954	±.0321	±.[illegible]123	.2124	.2207	.5121	1.9477	±.0229
	30	45.0- 67.5		.3500	.5327	1.3451	3.6204			.2126	.3030	.7530	2.3641	
	31	22.5- 45.0		.3927	.7827	2.3745	6.6717			.2161	.4135	1.2[illegible]	4.[illegible]725	
	32	0.0- 22.5		.3445	1.0602	2.6[illegible]7	12.8330			.2423	.4291	1.2[illegible]	8.1150	
	33	165.0-180.0		.1873	.1971	.3579	.5688			.1119	.1647	.172[illegible]	[illegible]	
	34	150.0-165.0		.1895	.1929	.3305	.5867			.1916	.1382	.2225	[illegible]	
	35	135.0-150.0		.1930	.2251	.3646	.7197			.1648	.1425	.22[illegible]	[illegible]	
	36	120.0-135.0		.1694	.1836	.3420	.8[illegible]2			.1522	.1780	.2[illegible]	[illegible]	
	37	105.0-120.0		.2286	.3029	.3551	.8635			.1904	.1430	[illegible]	.7953	
θ_7 64.6-77.6	38	90.0-105.0	.2050	.2058	.2422	.5331	1.2612	1.4115	.1211	.1272	.2119	.3323	.87[illegible]1	.7502
	39	75.0- 90.0	±.0103	.1967	.3224	.6087	1.7791	±.0642	±.[illegible]	.1391	.1974	.4402	1.4293	±.0[illegible]51
	40	60.0- 75.0		.2854	.4326	.8843	2.4472			.1262	.229[illegible]	.43[illegible]	1.8023	
	41	45.0- 60.0		.2814	.4961	1.2482	4.4222			.1592	.23[illegible]	[illegible]	2.44[illegible]2	
	42	30.0- 45.0		.2892	.6667	1.3969	7.[illegible]			.1576	.3336	1.[illegible]	[illegible]	
	43	15.0- 30.0		.2920	.6501	2.2916	14.7134			.1647	.3355	1.42[illegible]	10.2224	
	44	0.0- 15.0		.3426	.9045	3.27[illegible]	24.[illegible]173			.22[illegible]	.3332	1.[illegible]	1[illegible]	
	45	165.0-180.0		.3473	.0447	.0592	.2895			.0215	.0242	[illegible]	[illegible]	
	46	150.0-165.0		.[illegible]718	.1[illegible]53	.0872	.3[illegible]			.0519	.[illegible]95	[illegible]	[illegible]	
	47	135.0-150.0		.[illegible]711	.[illegible]197	.095[illegible]	.4287			.[illegible]208	.0741		[illegible]	
	48	120.0-135.0		.[illegible]354	.0858	.1122	.3969			.[illegible]28[illegible]	[illegible]		[illegible]	
	49	105.0-120.0		.[illegible]950	[illegible]	.1[illegible]	[illegible]			[illegible]	.[illegible]712			
θ_8 77.6-90.0	50	90.0-105.0	.0767	.2727	.1327	.1[illegible]	.7222	1.0792	.0466	.[illegible]511	[illegible]			[illegible]
	51	75.0- 90.0	±.[illegible]	.0741	.1[illegible]5	[illegible]	1.[illegible]	±.0780	±.0144	.[illegible]227	.[illegible]472			±.[illegible]
	52	60.0- 75.0		.1879	.1095	.3504	1.[illegible]81			.0799	.17[illegible]			
	53	45.0- 60.0		.0823	.1582	.[illegible]2	2.[illegible]			.[illegible]313	.14[illegible]			
	54	30.0- 45.0		.1145	.[illegible]11	1.1369	[illegible]			[illegible]	[illegible]			
	55	15.0- 30.0		.1[illegible]	.3383	1.7[illegible]3	1[illegible]			[illegible]	.1[illegible]			
	56	[illegible]- 15.0		.1164	.4592	2.2[illegible]	[illegible].1132			.[illegible]28	[illegible]			
Sum over Ω_k			15.4759	17.4022	27.3109	[illegible]	[illegible]	[illegible]	1.[illegible]	11.4141	[illegible]	[illegible]	[illegible]	[illegible]
Total Dose Albedos			1.[illegible]	1.2524	2.3527	[illegible]	[illegible]	7.7211	1.1371	1.29[illegible]	1.79[illegible]	[illegible]	[illegible]	[illegible]

*Symmetrical sources, so 2 values averaged
†For composition see previous page

SERIAL HNP-18-004

ENCLOSURE 6

PROPOSED HARRIS NUCLEAR PLANT EMERGENCY ACTION LEVEL WALLCHART CHANGES (MARKUP)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

R – Abnorm. Rad Levels / Rad Effluent

1 – Rad Effluent

General Emergency – Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

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

RG1.2
Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

RG1.3
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation.

(Notes 1, 2)

Site Area Emergency – Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RS1.1
Reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4)

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

RS1.3
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation.

(Notes 1, 2)

Alert – Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RA1.1
Reading on **any** Table R-1 effluent radiation monitor > column "Alert" for ≥ 15 min. (Notes 1, 2, 4)

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

RA1.3
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)

RA1.4
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)

Unusual Event – Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

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

RU1.2
Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2)

2 – Irradiated Fuel Event

General Emergency – Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RG2.1
Spent fuel pool level cannot be restored to at least 260.7 ft. (Level 3) for ≥ 60 min. (Note 1)

Site Area Emergency – Spent fuel pool level at the top of the fuel racks

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RS2.1
Lowering of spent fuel pool level ≤ 260.7 ft. (Level 3)

Alert – Significant lowering of water level above, or damage to, irradiated fuel

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RA2.1
Uncovery of irradiated fuel in the REFUELING PATHWAY

RA2.2
Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by high alarm on **any** of the following:
- Table R-2 refueling pathway area radiation monitors
- 1REM-*1FL-3508A-SA, FHB Emergency Exhaust
- 1REM-*1FL-3508B-SB, FHB Emergency Exhaust

RA2.3
Lowering of spent fuel pool level ≤ 270.7 ft. (Level 2)

Unusual Event – Unplanned loss of water level above irradiated fuel

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RU2.1
UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication (LI-01SF-5101A/LI-01SF-5102A/LI-01SF-5103A, LI-403 or RCS standpipe)
AND
UNPLANNED rise in corresponding area radiation levels as indicated by **any** Table R-2 area radiation monitors

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Plant Vent	RM-21AV-3509-1SA	1.05E+8 µCi/sec	1.05E+7 µCi/sec	1.05E+6 µCi/sec	1.14E+4 µCi/sec
Gaseous	Turbine Building	RM-1TV-3536-1	4.60E+8 µCi/sec	4.60E+7 µCi/sec	4.60E+6 µCi/sec	1.38E+4 µCi/sec
Gaseous	Waste Process Building Vent 5	RM-1WV-3546-1	7.74E+9 µCi/sec	7.74E+8 µCi/sec	7.75E+7 µCi/sec	2.49E+5 µCi/sec
Gaseous	Waste Process Building Vent 5A	RM-1WV-3547-1	7.76E+9 µCi/sec	7.76E+8 µCi/sec	7.76E+7 µCi/sec	1.45E+4 µCi/sec
Liquid	Treated Laundry & Hot Shower Tank Discharge	REM-1WL-3540	-----	-----	-----	7.02E-04 µCi/ml
Liquid	Waste Monitor/Waste Evaporator Condensate Tank Discharge	REM-21WL-3541	-----	-----	-----	1.97E-03 µCi/ml
Liquid	Secondary Waste Sample Tank Discharge	REM-21WS-3542	-----	-----	-----	7.02E-04 µCi/ml

Table R-2 Refueling Pathway AreaRadiation Monitors

Containment
- RM-1CR-3561A-SA Containment Ventilation Isolation
- RM-1CR-3561B-SB Containment Ventilation Isolation
- RM-1CR-3561C-SA Containment Ventilation Isolation
- RM-1CR-3561D-SB Containment Ventilation Isolation

Fuel Handling Building
- RM-1FR-3564A-SA Spent Fuel Pool SW, SE, SW
- RM-1FR-3564B-SB Spent Fuel Pool SW, SE, SE
- RM-1FR-3565A-SA Spent Fuel Pool SW, SE, SW
- RM-1FR-3565B-SB Spent Fuel Pool SW, SE, SE
- RM-1FR-3566A-SA Spent Fuel Pool NW, NE, NE
- RM-1FR-3566B-SB Spent Fuel Pool NW, NE, NW
- RM-1FR-3567A-SA Spent Fuel Pool NW, NE, NW
- RM-1FR-3567B-SB Spent Fuel Pool NW, NE, NE

3 – Area Rad Levels

General Emergency – None

Site Area Emergency – None

Alert – Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RA3.1
Dose rates > 15 mR/hr in **EITHER** of the following areas:
Control Room (RM-21RR-3560-SA)
OR
Central Alarm Station (by survey)

RA3.2
An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-3/H-2 rooms or areas (Note 5)

Unusual Event – None

H – Hazards

1 – Security

General Emergency – HOSTILE ACTION resulting in loss of physical control of the facility

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HG1.1
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision
AND EITHER of the following has occurred:
Any of the following safety functions cannot be controlled or maintained
- Reactivity
- Core Cooling
- RCS heat removal

OR
Damage to spent fuel has occurred or is IMMINENT

Site Area Emergency – HOSTILE ACTION within the PROTECTED AREA

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HS1.1
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision

Alert – HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA1.1
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervision

HA1.2
A validated notification from NRC of an aircraft attack threat within 30 min. of the site

Unusual Event – Confirmed SECURITY CONDITION or threat

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU1.1
A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervision

HU1.2
Notification of a credible security threat directed at the site

HU1.3
A validated notification from the NRC providing information of an aircraft threat to the site

2 – Seismic Event

General Emergency – None

Site Area Emergency – (Table R-3/H-2)

Alert – None

[Refer to EAL CA6.1 OR SA9.1 for escalation due to seismic event]

Unusual Event – Seismic event greater than OBE levels

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU2.1
Seismic event > OBE as indicated by **any** of the following:
- ALB-10/4-4, SEISMIC MON SYS OBE EXCEEDED is ALARMED
- ALARM light on Seismic Switch Power Supply is LIT
- Any red alarm light is LIT on the Response Spectrum Annunciator

Table R-3/H-2 Safe Operation & Shutdown Rooms/Areas

Room / Area	Mode(s)
- RAB 190 (RHR pumps)	4
- RAB 216 (BIT)	1, 2, 3, 4, 5
- RAB 236 (CSIP, Primary Sample Sink, AFW pumps, CCW pumps and HX, Boric Acid Transfer Pumps, Mezzanine Area)	1, 2, 3, 4, 5
- RAB 261 (RHR Heat Exchangers, Demin. Valve Gallery, VCT Valve Gallery)	1, 2, 3, 4, 5
- RAB 286 (Switchgear)	1, 2, 3, 4, 5
- Steam Tunnel	1, 2, 3, 4
- ESW intakes	1, 2, 3, 4, 5

3 – Natural or Tech. Hazard

General Emergency – None

Site Area Emergency – None

Alert – None

[Refer to EAL CA6.1 OR SA9.1 for escalation due to natural or technological hazard]

Unusual Event – Hazardous event

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU3.1
A tornado strike within the PROTECTED AREA

HU3.2
Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode

HU3.3
Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

HU3.4
A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)

4 – Fire

Alert – [Refer to EAL CA6.1 OR SA9.1 for escalation due to FIRE]

Unusual Event – FIRE potentially degrading the level of safety of the plant

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU4.1
A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):
- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND
The FIRE is located within **any** Table H-1 area

HU4.2
Receipt of a single fire alarm (i.e., **no** other indications of a FIRE)
AND
The fire alarm is indicating a FIRE within **any** Table H-1 area
AND
The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1)

HU4.3
A FIRE within the plant PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)

HU4.4
A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

Table H-1 Fire Areas
- Containment
- Reactor Auxiliary Building
- Emergency Diesel Generator Building
- Diesel Fuel Oil Storage Building (DFOST)
- ESW Intake Structure and Auxiliary Reservoir Intake Structure

NOTES

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

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

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

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

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

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

Note 7: This EAL does **not** apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 8: A manual trip action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

5 – Hazardous Gases

General Emergency – None

Site Area Emergency – None

Alert – Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA5.1
Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table R-3/H-2 rooms or areas
AND
Entry into the room or area is prohibited or IMPEDED (Note 5)

Unusual Event – None

6 – Control Room Evacuation

General Emergency – None

Site Area Emergency – Inability to control a key safety function from outside the Control Room

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HS6.1
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Control Panel
AND
Control of **any** of the following key safety functions is **not** reestablished within 15 min. (Note 1):
- Reactivity
- Core Cooling
- RCS heat removal

Alert – Control Room evacuation resulting in transfer of plant control to alternate locations

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA6.1
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Control Panel

Unusual Event – None

7 – EC Judgment

General Emergency – Other conditions exist which in the judgment of the Emergency Coordinator warrant declaration of a General Emergency

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HG7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area

Site Area Emergency – Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a Site Area Emergency

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HS7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the SITE BOUNDARY.

Alert – Other conditions exist that in the judgment of the Emergency Coordinator warrant declaration of an Alert

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Unusual Event – Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a UE

Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

C – Cold SD/ Refuel System Malfunct.

1 – RCS Level

General Emergency – Loss of RCS inventory affecting fuel clad integrity with containment challenged

Modes: 5 | 6

CG1.1
RCS level < 63% RVLIS Full Range for ≥ 30 min. (Note 1)
AND
Any Containment Challenge indication, Table C-2

CG1.2
RCS water level **cannot** be monitored for ≥ 30 min. (Note 1)
AND
Core uncovery is indicated by **any** of the following:
- UNPLANNED increase in **any** Table C-1 sump or tank of sufficient magnitude to indicate core uncovery
- ~~Containment radiation > 10,000 R/hr (RM-1CR-3589-SA or RM-1CR-3590-SB)~~
- Erratic source range monitor indication

AND
Any Containment Challenge indication, Table C-2

Replace with: "A Containment Ventilation Isolation Radiation Monitor > 2.6E+04 mR/hr (RM-1CR-3561A-SA, RM-1CR-3561B-SB, RM-1CR-3561C-SA, or RM-1CR-3561D-SB)"

Site Area Emergency – Loss of RCS inventory affecting core decay heat removal capability

Modes: 5 | 6

CS1.1
With CONTAINMENT CLOSURE **not** established, RCS level < 70% RVLIS Full Range

CS1.2
With CONTAINMENT CLOSURE established, RCS level < 63% RVLIS Full Range

CS1.3
RCS water level **cannot** be monitored for ≥ 30 min. (Note 1)
AND
Core uncovery is indicated by **any** of the following:
- UNPLANNED increase in **any** Table C-1 sump or tank of sufficient magnitude to indicate core uncovery
- ~~Containment radiation > 10,000 R/hr (RM-1CR-3589-SA or RM-1CR-3590-SB)~~
- Erratic source range monitor indication

Alert – Loss of RCS inventory

Modes: 5 | 6

CA1.1
Loss of RCS inventory as indicated by LI-403 or RCS standpipe level < - 82 in.

CA1.2
RCS water level **cannot** be monitored for ≥ 15 min. (Note 1)
AND EITHER
- UNPLANNED increase in **any** Table C-1 sump or tank due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

Unusual Event – UNPLANNED loss of RCS inventory for 15 minutes or longer

Modes: 5 | 6

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

CU1.2
RCS water level **cannot** be monitored
AND EITHER
- UNPLANNED increase in **any** Table C-1 sump or tank due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

Table C-1 Sumps / Tanks
- Containment sumps
- PRT
- RCDT
- CCW surge tank
- RAB sumps
- RWST
- RMWST
- Recycle Holdup Tank

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

2 – Loss of Emergency AC Power

Alert – Loss of **all** offsite and **all** onsite AC power to emergency buses for 15 minutes or longer

Modes: 5 | 6 | DEF

CA2.1
Loss of **all** offsite and **all** onsite AC power capability to 6.9 KV emergency buses 1A-SA and 1B-SB for ≥ 15 min. (Note 1)

Unusual Event – Loss of **all** but one AC power source to emergency buses for 15 minutes or longer

Modes: 5 | 6 | DEF

CU2.1
AC power capability, Table C-6, to emergency 6.9 KV buses 1A-SA and 1B-SB 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

Table C-6 AC Power Sources

Offsite
- SUT 1A
- SUT 1B
- UAT 1A/1B backfed via Main Transformer (**only** if already aligned)

Onsite
- EDG 1A-SA
- EDG 1B-SB

3 – RCS Temp.

Alert – Inability to maintain plant in cold shutdown

Modes: 5 | 6

CA3.1
UNPLANNED increase in RCS temperature to > 200°F for > Table C-3 duration (Note 1)
OR
UNPLANNED RCS pressure increase > 10 psig (this does **not** apply during water-solid plant conditions)

Unusual Event – UNPLANNED increase in RCS temperature

Modes: 5 | 6

CU3.1
UNPLANNED increase in RCS temperature to > 200°F

CU3.2
Loss of **all** RCS temperature and RCS level indication for ≥ 15 min. (Note 1)

Table C-3 RCS Heat-up Duration Thresholds

RCS Status	Containment Closure Status	Heat-up Duration
Intact (but **not** REDUCED INVENTORY)	N/A	60 min.*
Not intact **OR** At REDUCED INVENTORY	established	20 min.*
	not established	0 min.

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable

4 – Loss of Vital DC Power

General Emergency – None

Site Area Emergency – None

Unusual Event – Loss of Vital DC power for 15 minutes or longer

Modes: 5 | 6

CU4.1
< 105 VDC bus voltage indications on Technical Specification **required** 125 VDC buses (DP-1A-SA, DP-1B-SB) for ≥ 15 min. (Note 1)

5 – Loss of Comm.

General Emergency – None

Site Area Emergency – None

Unusual Event – Loss of **all** onsite or offsite communications capabilities

Modes: 5 | 6 | DEF

CU5.1
Loss of **all** Table C-4 onsite communication methods
OR
Loss of **all** Table C-4 offsite communication methods
OR
Loss of **all** Table C-4 NRC communication methods

Table C-4 Communication Methods

System	Onsite	Offsite	NRC
PABX telephone (desk phones)	X	X	X
HE&EC PABX telephone		X	X
Site paging system	X		
Satellite phone		X	X
DEMNET		X	
Radio communications networks	X		
NRC ETS phone			X
NRC HPN phone			X

6 – Hazardous Event Affecting Safety Systems

General Emergency – None

Site Area Emergency – None

Alert – Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

Modes: 5 | 6

CA6.1
The occurrence of **any** Table C-5 hazardous event
AND EITHER:
- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Unusual Event – None

Table C-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

NOTES

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

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

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

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

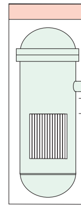
Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then **no** emergency classification is warranted.

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

Note 7: This EAL does **not** apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 8: A manual trip action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

RCS Levels



	Plant El.	Standpipe	RVLIS Full Range
Reactor Vessel Flange	260.62'	0"	89%
RHR Pump Operations	253.75'	- 82"	
6 in. < Bottom of Hotleg	252.04'	-----	70%
Top of Active Fuel	249.01'	-----	63%

Modes: 1 Power Operations | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refuel | DEF Defueled

DUKE ENERGY

Harris Nuclear Plant
Classification of Emergency
EP-EAL Matrix Revision 1

EAL - COLD MODES 5, 6 & Defueled

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

R — Abnorm. Rad Levels / Rad Effluent

1 Rad Effluent

General Emergency — Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE (Modes: 1 2 3 4 5 6 DEF)

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

RG1.2
Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

RG1.3
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation.

(Notes 1, 2)

Site Area Emergency — Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE (Modes: 1 2 3 4 5 6 DEF)

RS1.1
Reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4)

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

RS1.3
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation.

(Notes 1, 2)

Alert — Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE (Modes: 1 2 3 4 5 6 DEF)

RA1.1
Reading on **any** Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4)

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

RA1.3
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)

RA1.4
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)

Unusual Event — Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer (Modes: 1 2 3 4 5 6 DEF)

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

RU1.2
Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2)

2 Irradiated Fuel Event

General Emergency — Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer (Modes: 1 2 3 4 5 6 DEF)

RG2.1
Spent fuel pool level cannot be restored to at least 260.7 ft. (Level 3) for ≥ 60 min. (Note 1)

Site Area Emergency — Spent fuel pool level at the top of the fuel racks (Modes: 1 2 3 4 5 6 DEF)

RS2.1
Lowering of spent fuel pool level ≤ 260.7 ft. (Level 3)

Alert — Significant lowering of water level above, or damage to, irradiated fuel (Modes: 1 2 3 4 5 6 DEF)

RA2.1
Uncovery of irradiated fuel in the REFUELING PATHWAY

RA2.2
Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by high alarm on **any** of the following:
- Table R-2 refueling pathway area radiation monitors
- 1REM-*1FL-3508A-SA, FHB Emergency Exhaust
- 1REM-*1FL-3508B-SB, FHB Emergency Exhaust

RA2.3
Lowering of spent fuel pool level ≤ 270.7 ft. (Level 2)

Unusual Event — Unplanned loss of water level above irradiated fuel (Modes: 1 2 3 4 5 6 DEF)

RU2.1
UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication (LI-01SF-5101A/LI-01SF-5102A/LI-01SF-5103A, LI-403 or RCS standpipe)
AND
UNPLANNED rise in corresponding area radiation levels as indicated by **any** Table R-2 area radiation monitors

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Plant Vent	RM-21AV-3509-1SA	1.05E+8 μCi/sec	1.05E+7 μCi/sec	1.05E+6 μCi/sec	1.14E+4 μCi/sec
Gaseous	Turbine Building	RM-1TV-3536-1	4.60E+8 μCi/sec	4.60E+7 μCi/sec	4.60E+6 μCi/sec	1.38E+4 μCi/sec
Gaseous	Waste Process Building Vent 5	RM-1WV-3546-1	7.74E+9 μCi/sec	7.74E+8 μCi/sec	7.75E+7 μCi/sec	2.49E+5 μCi/sec
Gaseous	Waste Process Building Vent 5A	RM-1WV-3547-1	7.76E+9 μCi/sec	7.76E+8 μCi/sec	7.76E+7 μCi/sec	1.45E+4 μCi/sec
Liquid	Treated Laundry & Hot Shower Tank Discharge	REM-1WL-3540	-----	-----	-----	7.02E-04 μCi/ml
Liquid	Waste Monitor/Waste Evaporator Condensate Tank Discharge	REM-21WL-3541	-----	-----	-----	1.97E-03 μCi/ml
Liquid	Secondary Waste Sample Tank Discharge	REM-21WS-3542	-----	-----	-----	7.02E-04 μCi/ml

Table R-2 Refueling Pathway AreaRadiation Monitors

Containment
- RM-1CR-3561A-SA Containment Ventilation Isolation
- RM-1CR-3561B-SB Containment Ventilation Isolation
- RM-1CR-3561C-SA Containment Ventilation Isolation
- RM-1CR-3561D-SB Containment Ventilation Isolation

Fuel Handling Building
- RM-1FR-3564A-SA Spent Fuel Pool SW, SE, SW
- RM-1FR-3564B-SB Spent Fuel Pool SW, SE, SE
- RM-1FR-3565A-SA Spent Fuel Pool SW, SE, SW
- RM-1FR-3565B-SB Spent Fuel Pool SW, SE, SE
- RM-1FR-3566A-SA Spent Fuel Pool NE, NW, NE
- RM-1FR-3566B-SB Spent Fuel Pool NW, NE, NW
- RM-1FR-3567A-SA Spent Fuel Pool NW, NE, NW
- RM-1FR-3567B-SB Spent Fuel Pool NE, NW, NE

3 Area Rad Levels

General Emergency: None

Site Area Emergency: None

Alert — Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown (Modes: 1 2 3 4 5 6 DEF)

RA3.1
Dose rates > 15 mR/hr in **EITHER** of the following areas:
Control Room (RM-21RR-3560-SA)
OR
Central Alarm Station (by survey)

RA3.2
An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-3/H-2 rooms or areas (Note 5)

H — Hazards

1 Security

General Emergency — HOSTILE ACTION resulting in loss of physical control of the facility (Modes: 1 2 3 4 5 6 DEF)

HG1.1
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision
AND EITHER of the following has occurred:
Any of the following safety functions cannot be controlled or maintained
- Reactivity
- Core Cooling
- RCS heat removal

OR
Damage to spent fuel has occurred or is IMMINENT

Site Area Emergency — HOSTILE ACTION within the PROTECTED AREA (Modes: 1 2 3 4 5 6 DEF)

HS1.1
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision

Alert — HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes (Modes: 1 2 3 4 5 6 DEF)

HA1.1
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervision

HA1.2
A validated notification from NRC of an aircraft attack threat within 30 min. of the site

Unusual Event — Confirmed SECURITY CONDITION or threat (Modes: 1 2 3 4 5 6 DEF)

HU1.1
A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervision

HU1.2
Notification of a credible security threat directed at the site

HU1.3
A validated notification from the NRC providing information of an aircraft threat to the site

2 Seismic Event

General Emergency: None

Alert: None

[Refer to EAL CA6.1 OR SA9.1 for escalation due to seismic event]

Unusual Event — Seismic event greater than OBE levels (Modes: 1 2 3 4 5 6 DEF)

HU2.1
Seismic event > OBE as indicated by **any** of the following:
- ALB-10/4-4, SEISMIC MON SYS OBE EXCEEDED is ALARMED
- ALARM light on Seismic Switch Power Supply is LIT
- Any red alarm light is LIT on the Response Spectrum Annunciator

Table R-3/H-2 Safe Operation & Shutdown Rooms/Areas

Room / Area	Mode(s)
- RAB 190 (RHR pumps)	4
- RAB 216 (BIT)	1, 2, 3, 4, 5
- RAB 236 (CSIP, Primary Sample Sink, AFW pumps, CCW pumps and HX, Boric Acid Transfer Pumps, Mezzanine Area)	1, 2, 3, 4, 5
- RAB 261 (RHR Heat Exchangers, Demin. Valve Gallery, VCT Valve Gallery)	1, 2, 3, 4, 5
- RAB 286 (Switchgear)	1, 2, 3, 4, 5
- Steam Tunnel	1, 2, 3, 4
- ESW intakes	1, 2, 3, 4, 5

3 Natural or Tech. Hazard

General Emergency: None

Site Area Emergency: None

Alert: None

[Refer to EAL CA6.1 OR SA9.1 for escalation due to natural or technological hazard]

Unusual Event — Hazardous event (Modes: 1 2 3 4 5 6 DEF)

HU3.1
A tornado strike within the PROTECTED AREA

HU3.2
Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode

HU3.3
Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

HU3.4
A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)

4 Fire

NOTES

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

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

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

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

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

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

Note 7: This EAL does **not** apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 8: A manual trip action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

Table H-1 Fire Areas
- Containment
- Reactor Auxiliary Building
- Emergency Diesel Generator Building
- Diesel Fuel Oil Storage Building (DFOST)
- ESW Intake Structure and Auxiliary Reservoir Intake Structure

[Refer to EAL CA6.1 OR SA9.1 for escalation due to FIRE]

Unusual Event — FIRE potentially degrading the level of safety of the plant (Modes: 1 2 3 4 5 6 DEF)

HU4.1
A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):
- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND
The FIRE is located within **any** Table H-1 area

HU4.2
Receipt of a single fire alarm (i.e., **no** other indications of a FIRE)
AND
The fire alarm is indicating a FIRE within **any** Table H-1 area
AND
The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1)

HU4.3
A FIRE within the plant PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)

HU4.4
A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

5 Hazardous Gases

General Emergency: None

Site Area Emergency: None

Alert — Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown (Modes: 1 2 3 4 5 6 DEF)

HA5.1
Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table R-3/H-2 rooms or areas
AND
Entry into the room or area is prohibited or IMPEDED (Note 5)

Unusual Event: None

6 Control Room Evacuation

General Emergency: None

Site Area Emergency — Inability to control a key safety function from outside the Control Room (Modes: 1 2 3 4 5 6 DEF)

HS6.1
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Control Panel
AND
Control of **any** of the following key safety functions is not reestablished within 15 min. (Note 1):
- Reactivity
- Core Cooling
- RCS heat removal

Alert — Control Room evacuation resulting in transfer of plant control to alternate locations (Modes: 1 2 3 4 5 6 DEF)

HA6.1
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Control Panel

Unusual Event: None

7 EC Judgment

General Emergency — Other conditions exist which in the judgment of the Emergency Coordinator warrant declaration of a General Emergency (Modes: 1 2 3 4 5 6 DEF)

HG7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area

Site Area Emergency — Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a Site Area Emergency (Modes: 1 2 3 4 5 6 DEF)

HS7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the SITE BOUNDARY.

Alert — Other conditions exist that in the judgment of the Emergency Coordinator warrant declaration of an Alert (Modes: 1 2 3 4 5 6 DEF)

HA7.1
Other conditions exist which, in the judgment of the Emergency Coordinator, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security threat 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.

Unusual Event — Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a UE (Modes: 1 2 3 4 5 6 DEF)

HU7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

Modes: 1 Power Operations | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refuel | DEF Defueled

DUKE ENERGY

Harris Nuclear Plant Classification of Emergency EP-EAL Matrix Revision 1

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

S — System Malfunct.

1 Loss of Emergency AC Power

General Emergency — Prolonged loss of all offsite and all onsite AC power to emergency buses (Modes: 1 2 3 4)

SG1.1
Loss of **all** offsite and **all** onsite AC power capability to 6.9 KV emergency buses 1A-SA and 1B-SB
AND EITHER:
- Restoration of at least one emergency bus in < 4 hours is **not** likely
- Core Cooling RED Path entry conditions met

General Emergency — Loss of all emergency AC and vital DC power sources for 15 minutes or longer (Modes: 1 2 3 4)

SG1.2
Loss of **all** offsite and **all** onsite AC power capability to 6.9 KV emergency buses 1A-SA and 1B-SB for ≥ 15 min.
AND
Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **both** emergency DC buses (DP-1A-SA, DP-1B-SB) for ≥ 15 min.
(Note 1)

Site Area Emergency — Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer (Modes: 1 2 3 4)

SS1.1
Loss of **all** offsite and **all** onsite AC power capability to 6.9 KV emergency buses 1A-SA and 1B-SB for ≥ 15 min. (Note 1)

Alert — Loss of all but one AC power source to emergency buses for 15 minutes or longer (Modes: 1 2 3 4)

SA1.1
AC power capability, Table S-5, to 6.9 KV emergency buses 1A-SA and 1B-SB 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

Unusual Event — Loss of all offsite AC power capability to emergency buses for 15 minutes or longer (Modes: 1 2 3 4)

SU1.1
Loss of **all** offsite AC power capability, Table S-5, to 6.9 KV emergency buses 1A-SA and 1B-SB for ≥ 15 min. (Note 1)

Table S-5 AC Power Sources

Offsite
- SUT 1A
- SUT 1B
- UAT 1A/1B backfed via Main Transformer (**only** if already aligned)

Onsite
- UAT 1A/1B via Main Generator
- EDG 1A-SA
- EDG 1B-SB

2 Loss of Vital DC Power

Site Area Emergency — Loss of all vital DC power for 15 minutes or longer (Modes: 1 2 3 4)

SS2.1
Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **both** emergency DC buses (DP-1A-SA, DP-1B-SB) for ≥ 15 min. (Note 1)

Alert: None

3 Loss of CR Indications

General Emergency: None

Table S-1 Safety System Parameters
- Reactor power
- RCS level
- RCS pressure
- Core exit T/C temperature
- Level in at least one S/G
- Auxiliary feed flow in at least one S/G

Alert — UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress (Modes: 1 2 3 4)

SA3.1
An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1)
AND
Any significant transient is in progress, Table S-2

Unusual Event — UNPLANNED loss of Control Room indications for 15 minutes or longer (Modes: 1 2 3 4)

SU3.1
An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1)

4 RCS Activity

General Emergency: None

Table S-2 Significant Transients
- Reactor trip
- Runback > 25% thermal power
- Electrical load rejection > 25% electrical load
- Safety injection actuation

Alert: None

[Refer to fission product barrier EALs for escalation due to fuel clad failures]

Unusual Event — RCS activity greater than Technical Specification allowable limits (Modes: 1 2 3 4)

SU4.1
RCS activity > Technical Specification Section 3.4.8 limits

SU4.2
Valid Gross Failed Fuel Detector (RS-7411A) high alarm (> 1E+04 cpm)

5 RCS Leakage

General Emergency: None

Site Area Emergency: None

Alert: None

[Refer to fission product barrier EALs for escalation due to RCS leakage]

Unusual Event — RCS leakage for 15 minutes or longer (Modes: 1 2 3 4)

SU5.1
RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min.
OR
RCS identified leakage > 25 gpm for ≥ 15 min.
OR
Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min.
(Note 1)

6 RPS Failure

General Emergency: None

Site Area Emergency — Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal (Modes: 1)

SS6.1
An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%
AND
All actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5%
AND EITHER:
- Core Cooling RED Path entry conditions met
- Heat Sink RED Path entry conditions met

Alert — Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor (Modes: 1)

SA6.1
An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%
AND
Manual trip actions taken at the reactor control console (actuation of MCB Reactor Trip Switch #1, #2 or MCB Turbine Trip switch) are **not** successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8)

Unusual Event — Automatic or manual trip fails to shut down the reactor (Modes: 1)

SU6.1
An automatic trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** RPS setpoint is exceeded
AND
A subsequent automatic trip or manual trip action taken at the reactor control console (actuation of MCB Reactor Trip Switch #1, #2 or MCB Turbine Trip switch) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

SU6.2
A manual trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** manual trip action was initiated
AND
A subsequent automatic trip or manual trip action taken at the reactor control console (actuation of MCB Reactor Trip Switch #1, #2 or MCB Turbine Trip switch) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

NOTES

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

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

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

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

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

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

Note 7: This EAL does **not** apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 8: A manual trip action is **any** operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies.

7 Loss of Comm.

Site Area Emergency: None

Alert: None

Table S-3 Communication Methods

System	Onsite	Offsite	NRC
PABX telephone (desk phones)	X	X	X
HE&EC PABX telephone		X	X
Site paging system	X		
Satellite phone		X	X
DEMNET		X	
Radio communications networks	X		
NRC ETS phone			X
NRC HPN phone			X

Unusual Event — Loss of all onsite or offsite communications capabilities (Modes: 1 2 3 4)

SU7.1
Loss of **all** Table S-3 onsite communication methods
OR
Loss of **all** Table S-3 offsite communication methods
OR
Loss of **all** Table S-3 NRC communication methods

8 CNMT Failure

Site Area Emergency: None

Alert: None

Unusual Event — Failure to isolate containment or loss of containment pressure control (Modes: 1 2 3 4)

SU8.1
EITHER:
Any penetration is **not** isolated within 15 min. of a VALID containment isolation signal
OR
Containment pressure > 10 psig with < one full train of depressurization equipment operating (one CNMT spray pump and two CNMT fan coolers) per design for ≥ 15 min. (Note 1)

9 Hazardous Event Affecting Safety Systems

General Emergency: None

Table S-4 Hazardous Events
- Seismic event (earthquake)
- Internal or external FLOODING event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Other events with similar hazard characteristics as determined by the Shift Manager

Alert — Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode (Modes: 1 2 3 4)

SA9.1
The occurrence of **any** Table S-4 hazardous event
AND EITHER:
- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Unusual Event: None

F — Fission Product Barriers

FG1.1 (Modes: 1 2 3 4)
Loss of **any** two barriers
AND
Loss or potential loss of third barrier (Table F-1)

FS1.1 (Modes: 1 2 3 4)
Loss or potential loss of **any** two barriers (Table F-1)

FA1.1 (Modes: 1 2 3 4)
Any loss or **any** potential loss of either Fuel Clad or RCS (Table F-1)

Unusual Event: None

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad (FC) Barrier: Loss	Fuel Clad (FC) Barrier: Potential Loss	Reactor Coolant System (RCS) Barrier: Loss	Reactor Coolant System (RCS) Barrier: Potential Loss	Containment (CNMT) Barrier: Loss	Containment (CNMT) Barrier: Potential Loss
A. RCS or SG Tube Leakage	None	None	1. An automatic or manual ECCS (SI) actuation required by **EITHER**: • UNISOLABLE RCS leakage • SG tube RUPTURE	1. Operation of a standby charging pump is required due to **EITHER**: • UNISOLABLE RCS leakage • SG tube leakage 2. CSFST Integrity-RED Path entry conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B. Inadequate Heat Removal	1. CSFST Core Cooling-RED Path entry conditions met	1. CSFST Core Cooling-ORANGE Path entry conditions met 2. CSFST Heat Sink-RED Path entry conditions met **AND** Heat Sink is required	None	1. CSFST Heat Sink-RED Path entry conditions met **AND** Heat Sink is required	None	1. CSFST Core Cooling-RED Path entry conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C. CNMT Radiation / RCS Activity	1. ~~Containment radiation >150 R/hr (RM-1CR-3589-SA or RM-1CR-3590-SB)~~ 2. Dose equivalent I-131 coolant activity > 300 μCi/gm	None	1. ~~Containment Leak Detection Monitor Noble Gas (REM-1LT-3502A-SA) > 8.3E-3 μCi/ml~~	None	None	1. ~~Containment radiation > 600 R/hr (RM-1CR-3589-SA or RM-1CR-3590-SB)~~
D. CNMT Integrity or Bypass	None	None	None	None	1. Containment isolation is required **AND EITHER** • Containment integrity has been lost based on Emergency Coordinator judgment • UNISOLABLE pathway from Containment to the environment exists 2. Indications of RCS leakage outside of containment	1. CSFST Containment-RED Path entry conditions met 2. Containment hydrogen concentration > 4% 3. Containment pressure > 10 psig with < one full train of depressurization equipment operating (one CNMT spray pump and two CNMT fan coolers) per design for ≥ 15 min.(Note 1)
E. EC Judgment	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier

Revise to "1. (RM-1CR-3589SA or RM-1CR-3590SB) > Table F-2 Column FC Barrier Loss"

Revise to 1. " (RM-1CR-3561A-SA, RM-1CR-3561B-SB, RM-1CR-3561C-SA, or RM-1CR-3561D-SB) > Table F-2 Column RCS Barrier Loss"

Revise to "1. (RM-1CR-3589SA or RM-1CR-3590SB) > Table F-2 Column CNMT Potential Loss"

Add INSERT A →

EAL- HOT MODES 1, 2, 3 & 4

INSERT A:

Table F-2 Containment Radiation			
Time After S/D (Hours)	FC Barrier Loss R/hr	RCS Barrier Loss mR/hr	CNMT Potential Loss R/hr
0 - 1	130	1.37E+03	2360
1 - 2	110	1.12E+03	2000
2 - 8	70	6.35E+02	1300
> 8	21	1.37E+02	390