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December 16, 2016
L-16-345

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

SUBJECT:
Davis-Besse Nuclear Power Station, Unit No. 1
Docket No. 50-346, License No. NPF-3
Response to Request for Additional Information Regarding License Amendment Request to Adopt National Fire Protection Association (NFPA) Standard 805 (CAC No. MF7190)

By letter dated December 16, 2015 (ADAMS Accession No. ML15350A314), as supplemented by letters dated March 7, 2016 and July 28, 2016 (Accession Nos. ML16067A195 and ML16210A422, respectively), FirstEnergy Nuclear Operating Company (FENOC) submitted a license amendment request (LAR) to change the Davis-Besse Nuclear Power Station, Unit No. 1 fire protection program to one based on the National Fire Protection Association Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition. The Nuclear Regulatory Commission (NRC) requested additional information in a letter dated October 18, 2016 (Accession No. ML16256A066) to complete its review of the LAR.

In accordance with the October 18, 2016 letter, the FENOC response that is due within 60 days is attached.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager - Fleet Licensing, at 330-315-6810.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 16, 2016.

Sincerely,

Brian D. Boles

Brian D. Boles

Attachment: Response to Request for Additional Information

cc: NRC Regional Administrator - Region III
NRC Resident Inspector
NRC Project Manager
Executive Director, Ohio Emergency Management Agency,
State of Ohio (NRC Liaison)
Utility Radiological Safety Board

The NRC staff requested additional information to complete their review of a FENOC LAR for the Davis-Besse Nuclear Power Station (DBNPS). The LAR would change the current fire protection program to one based on the National Fire Protection Association Standard 805 (NFPA 805), "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition. The NRC staff's request is provided below in bold text followed by the corresponding FENOC response.

Fire Modeling (FM) Request for Additional Information (RAI) 01

NFPA 805, Section 2.4.3.3, requires that the probabilistic risk assessment (PRA) approach, methods, and data shall be acceptable to the NRC. LAR, Section 4.5.1.2, "Fire PRA," states that, "Fire modeling was performed as part of the Fire PRA development (NFPA 805, Section 4.2.4.2)." LAR, Attachment J, "Fire Modeling V&V [Verification and Validation]," discusses the acceptability of the fire models used. Address the following questions regarding the information in LAR, Attachment J, Table J-2, "Technical Basis for Fire Modeling Approaches and Methodologies."

a. **On page J-17, the LAR states that to calculate the burning area, the entire width of the cable tray was assumed to ignite and that the length of the tray assumed to initially ignite was determined by the length of the tray exposed to the fire.**

   Explain how the initially ignited cable tray length was determined in the calculations of fire propagation in stacks of horizontal cable trays. Provide technical justification for the approach that was used.

Response:

For cable trays that were ignited due to plume temperatures, the initially-ignited cable tray length was determined based on the fire diameter of the ignition source. For most fire scenarios, a cable tray that is routed over the width of an ignition source will have an initial ignition length equal to the width of the ignition source. For oil fires, the initial tray length ignited was equal to the oil spill diameter. This is consistent with Section R.4.2 of NUREG/CR-6850, "Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology," Appendix R for Chapter 11 cable fires. Cable trays that were ignited due to a high energy arcing fault (HEAF) also use the ignition source width as the initially-ignited cable tray length.

For cable trays that were ignited due to radiant heat, the initially-ignited tray length was assumed to be 24 inches based on the 24-inch cable tray width. This is conservative because the fire diameter is typically smaller than 24 inches at the time cable tray

ignition occurs. Additionally, the trays ignited by radiant heat are usually located slightly above the ignition source, and the flame temperature decreases as a function of distance upwards, which results in a smaller portion of the flame capable of igniting the tray. Assuming the burning area of the initial fire is the square of the tray width is consistent with Section R.1 of NUREG/CR-6850, Volume 2, for modeling cable fires in a fire PRA.

FM RAI 01

b. On page J-17, the LAR states (emphasis added):

> **NUREG/CR-6850, Appendix R[1] provides cable tray properties and guidance on determining the HRRPUA [heat release rate per unit area] and spread rate for both thermoset and thermoplastic cable trays. For most areas, DBNPS fire modeling analyses use the NUREG/CR-6850 spread rates and the most conservative NUREG/CR-6850 Table R-1 bench scale HRRPUAs (adjusted using the Lee correlation, *shown in the table below)* for each cable type in the fire growth analysis for cable trays. For some risk-significant fire scenarios, subsequent fire modeling refinements utilize the refined HRRPUAs recommended by NUREG/CR-7010[2].**

The referenced table was not provided. Provide the referenced table.

Response:

The table referenced on Page J-17 of the LAR is provided below:

	NUREG/CR-6850, Volume 2			NUREG/CR-7010, Volume 1		
Cable Type	**HRRPUA (kW/m^2)**	**Spread Rate (m/h)**	**Basis**	**HRRPUA (kW/m^2)**	**Spread Rate (m/h)**	**Basis**
Thermoset/ Kerite-FR*	215	1.1	Appendix (App) R.3, Table R-1, App. R.4.1.2	150	1.1	Section 10.1, Section 10.3
Thermoplastic /Unknown	269	3.2	App. R.3, Table R-1, App. R.4.1.2	250	3.2	Section 10.1, Section 10.3

*Kerite-FR is treated as a thermoset material per frequently asked question (FAQ) 08-0053.

FM RAI 01

c. On page J-18, the LAR states that if fire propagation to non-cable secondary combustibles was possible, it was included in the ignition source fire growth analysis.

Explain how the time to ignition and subsequent fire propagation and heat release rate of non-cable secondary combustibles was determined.

Response:

All non-cable secondary combustibles were ignited at one minute, which corresponds with the shortest failure time identified in Appendix H of NUREG/CR-6850, Volume 2. The heat release rate (HRR) of the non-cable secondary combustibles was estimated based on the type and quantity. The HRRPUA values were taken from Table 8-1 of NUREG-1805, "Fire Dynamics Tools," and the subsequent burning duration was determined utilizing NUREG-1805 Fire Dynamics Tool (FDT) 08. The total exposed area of the combustibles was determined by walkdowns. The non-cable combustibles were assumed to reach the peak HRR at one minute and remain steady for the entire burning duration.

FM RAI 01

d. On page J-19, the LAR describes the process for placing transient fires in each compartment in the fire PRA. However, the licensee did not describe how it determined the area and elevation of transient fires.

Describe and provide technical justification for the fire area and elevation that were assumed in the transient fire scenarios postulated in the fire PRA.

Response:

Transient fires were modeled with a fire area of 2 feet x 2 feet (4 feet squared, or 4 ft^2) and a fire elevation of 2 feet. The 4 ft^2 area was selected for each transient scenario as a realistic characterization of a transient fuel package.

The fire area is used when calculating the plume zone of influence (ZOI) and flame height. In order to demonstrate that the modeled fire area was acceptable for a given fire, the fire Froude number was calculated to determine whether it was within the validated range of NUREG-1934, "Nuclear Power Plant Fire Modeling Analysis Guidelines (NPP FIRE MAG)." The Froude number is predominately used to validate the use of a model to calculate the plume temperatures and flame heights of a given fire scenario. Technical details demonstrating the fire scenarios are within the validated

range, as well as any justification for fire scenarios outside of the validated range, are outlined in the DBNPS fire modeling verification and validation documentation.

With respect to choosing the fire elevation for transient fires, the use of the height of 2 feet is recommended by IMC 0609. Furthermore, the 2 feet elevation directly adds to the calculated damaging plume height and radial ceiling jet ZOI, which conservatively affects more targets than if the fire elevation was assumed to be at the floor.

FM RAI 01

e. During the onsite audit, the NRC staff observed several electrical cabinets in the back panel area of the control room complex (fire compartment FF-01) that have Plexiglas® doors. Fires involving cabinets with Plexiglas® doors are not explicitly considered in the control room abandonment calculations, although these cabinets may have a higher heat release rate than otherwise identical cabinets with steel doors.

Provide technical justification for not explicitly considering fire scenarios that involve electrical cabinets with Plexiglas® doors in the control room abandonment calculations. Alternatively, confirm that the effect of these scenarios on the probability for control room abandonment and the associated core damage frequency (CDF), change in CDF (ΔCDF), large early release frequency (LERF), and change in LERF (ΔLERF) will be included in the integrated analysis as part of the response to PRA RAI 03.

Response:

The electrical cabinets in the main control room (MCR) that have Plexiglas® doors were modeled in the MCR abandonment analysis using the HRR distribution for Case 2 (vertical cabinets with qualified cable, fire in more than one cable bundle) from Table E-3 in NUREG/CR-6850, Appendix E, which has a 98th percentile of 702 kilowatt (kW) HRR. The following calculation analyzes the combustibles within the cabinets (internal cables and the Plexiglas® door) to demonstrate the modeled HRR for these cabinets bounds the HRR contribution of the doors.

The HRR of the cabinet internal combustible content can be determined utilizing the most recent guidance in NUREG-2178, “Refining And Characterizing Heat Release Rates From Electrical Enclosures During Fire (RACHELLE-FIRE) — Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume,” Table 7-1, which is based on enclosure class or function and volume, internal cable qualification, door condition, and fuel loading. There are three different size electrical cabinets in the MCR that have Plexiglas® doors. These cabinets were assigned to one of the following three groups based on the size of the cabinet and the size of the door, which were determined by walkdowns:

Group 1 – cabinet volume: 28 feet cubed (ft^3), door area: 11.8 ft^2
Group 2 – cabinet volume: 35 ft^3, door area: 8.1 ft^2
Group 3 – cabinet volume: 12 ft^3, door area: 1.8 ft^2

For the HRR determination, each cabinet group utilized the default fuel loading and thermoset cable fuel type. Group 1 and Group 3 cabinets utilized the closed-door condition since there are solid metal panels spanning the entire height of the panel behind the door. Group 2 cabinets utilized the open door condition since there are gaps in the metal panels behind the door which expose the cabinet internals.

The HRR of the doors can be calculated by multiplying the surface area of the Plexiglas® by its HRRPUA. The HRRPUA for Plexiglas® stored vertically is 23.2 kW/ft^2 per the guidance in NUREG/CR-6850, Table G-8. The table below determines the total HRR for each group assuming the complete ignition of the doors.

Group	Cabinet Volume (ft^3)	Plexiglas® Door Area (ft^2)	Cabinet Classification from NUREG-2178	Cabinet HRR (kW)	Plexiglas® HRR (kW)	Total HRR (kW)
1	28	11.8	Medium Enclosure, Closed Door, Thermoplastic (TP) Cables, Default Fuel Loading, 98th Percentile HRR	200	274	474
2	35	8.1	Medium Enclosure, Open Door, TP Cables, Default Fuel Loading, 98th Percentile HRR	325	188	513
3	12	1.8	Small Enclosure, 98th Percentile HRR	45	42	87

Although the MCR abandonment fire scenarios do not explicitly model the Plexiglas® doors, the cabinets with Plexiglas® doors are modeled using the 98th percentile 702 kW HRR from NUREG/CR-6850, which exceeds the total HRR calculated above for the cabinets and is therefore considered bounding. As such, no changes are necessary to the integrated analysis as part of the response to PRA RAI 03.

FM RAI 02

LAR, Section 4.5.1, states, in part, that:

> **In accordance with the guidance in [Regulatory Guide] RG 1.205,[3] a Fire PRA model was developed for DBNPS in compliance with the requirements of Part 4 "Requirements for Fires At Power PRA," of the American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) combined PRA Standard, ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Application,"…**

Part 4 of ASME/ANS RA-Sa-2009 requires damage thresholds be established to support the fire PRA. Thermal impacts must be considered in determining the potential for thermal damage of structures, systems, and components, and appropriate temperature and critical heat flux criteria must be used in the analysis. The LAR does not address the potential failure of sensitive electronics immersed in the lower layer due to the combined effect of the elevated temperature of the lower layer and thermal radiation from the hot gas layer.

Describe the treatment of sensitive electronics throughout the plant where they are in the lower parts of a room when there is a hot gas layer in the upper part of a room. Provide technical justification for such treatment if it is assumed that the electronics do not fail.

Response:

A temperature-sensitive-equipment hot gas layer (HGL) study using the computer program CFAST was performed with varying representative geometries and a range of fire sizes for both fixed and transient sources. The CFAST simulations were used to develop generic categories and are documented in the DBNPS fire modeling verification and validation documentation. For each generic simulation, the upper gas layer and the lower gas layer temperatures were analyzed to determine if the damaging hot gases could descend to equipment level, resulting in equipment failure.

The conclusions from the temperature-sensitive-equipment HGL study were applied in the fire modeling analysis by correlating each fire-modeled fire compartment to one of the generic fire modeling simulations. The correlation was made by examining the fire compartment parameters (compartment volume and ceiling height), and fire scenario characteristics (heat release rate and fire growth profile). Fire compartments with parameters within the limits of a generic simulation were judged to perform similarly with respect to gas layer formation. Details regarding the application of this study are documented on a compartment basis in the DBNPS fire PRA detailed fire modeling documentation.

The DBNPS treatment of sensitive electronics bounds the combined effect of the elevated temperature of the lower layer and thermal radiation from the hot gas layer

based on the following:

- Many fire compartments either do not contain any temperature-sensitive equipment, or such equipment in the compartment was assumed damaged in every scenario.
- Fire compartments that potentially contain sensitive electronics have been re-evaluated to determine if the upper HGL was capable of creating a 3 kW/m^2 radiant heat flux exposure necessary to damage sensitive electronics.

    - The heat flux emanating from the upper layer was approximated using the Stefan-Boltzmann Equation, $q_r'' = \sigma T_{sl}^4$, which is the same method used to approximate the radiation from the smoke layer for the main control room abandonment criteria in NUREG/CR-6850 Section 11.5.2.11 (a 95 degree Centigrade, or °C, smoke layer will generate a heat flux exceeding 1 kW/m^2 at 6 feet above the floor). The HGL temperature required to generate a heat flux of 3 kW/m^2 is approximately 205 °C.
    - Most fire compartments do not contain any fire scenarios capable of creating a 205 °C HGL temperature. Therefore, the radiant heat flux emanating from the HGL in these compartments will not exceed 3 kW/m^2.
    - Fire scenarios capable of creating a 205 °C HGL, but not modeled as whole room damage, were evaluated to determine if the current sensitive electronic treatment was bounding or if additional equipment failures were required. These scenarios are discussed below:

        - In fire compartment II-01, there are two oil fire scenarios that are capable of creating a HGL temperature exceeding 205 °C but are not modeled as whole room damage. These ignition sources are located on the 585 foot elevation of the turbine building that contains large equipment hatches and areas with grated floor where the hot gases are capable of propagating to the upper elevations. The HGL will be dispersed throughout the compartment and will eventually accumulate at the ceiling of the turbine deck where there are no sensitive electronics. Therefore, the radiant heat flux emanating from the HGL will not exceed 3 kW/m^2 where any potential sensitive electronics are located.
        - In fire compartment R-01, there is a transient fire scenario capable of creating a HGL temperature exceeding 205 °C but is not modeled as whole room damage. The upper gas layer is not expected to accumulate near the potential sensitive electronics since the compartment has a duct chase that is 40 feet high where the majority of the hot gases are assumed to travel up. Additionally, the cabinets containing sensitive electronics in this compartment are non-ventilated; therefore the thermoset screening heat flux (11 kW/m^2), as observed on the outer surface of the cabinet, can be utilized based on the guidance in FAQ 13-0004. The HGL temperature required to generate a heat flux of 11 kW/m^2, calculated using the method stated above, is greater than 330 °C.

There are no transient fire scenarios in this compartment capable of creating a HGL temperature exceeding 330 °C.

Fire Protection Engineering (FPE) RAI 01

The licensee has proposed a license condition that would require implementation of the plant modifications listed in Table S-1, "Plant Modifications Committed," in LAR, Attachment S. In addition, the LAR states that the items listed in Table S-2, "Implementation Items," will be completed prior to implementation of the NFPA 805 fire protection program. The NRC staff identified several inconsistencies between information in other sections of the LAR and Attachment S. Revise Tables S-1 and S-2, as appropriate, or justify the following inconsistencies:

a. Table S-2 does not include the following open items identified in Table B-1: DB-0779, DB-1900, DB-0540, DB-1912, DB-1838, and DB-2041.

Response:

DB-0779, DB-1900, DB-0540, DB-1912, DB-1838, and DB-2041 will be added to Table S-2. A revision to the LAR Attachment S will be provided in a future transmittal.

FPE RAI 01

b. Table S-2, Implementation Item DB-1825, references Attachment Z, which does not exist.

Response:

The Table S-2 implementation item DB-1825 reference to Attachment Z will be deleted. A revision to the LAR Attachment S will be provided in a future transmittal.

FPE RAI 01

c. Table B-3 states that modification DB-2033 will be tracked for implementation under LAR Attachment S. This modification is not listed in Attachment S.

Response:

DB-2033 will be added to Table S-1. If DB-2036 in Table S-2 concludes the instrumentation in its current condition will remain available after an inadvertent containment spray, then no modification for DB-2033 will be necessary. If modifications

are required, then adding DB-2033 to Table S-1 will ensure they are implemented. A revision to the LAR Attachment S will be provided in a future transmittal.

FPE RAI 01

d. Section 4.7.1 of the LAR states that the fire protection design-basis document and supporting documentation will be created as part of the transition to NFPA 805. This action is not listed as an implementation item in Table S-2.

Response:

DB-2049 will be added to Table S-2 as an implementation item to develop the design basis and supporting documentation. The item description is "Develop the DBNPS Fire Safety Analysis (FSA)." A revision to the LAR Attachment S will be provided in a future transmittal.

FPE RAI 01

e. Section 4.7.2 of the LAR discusses the configuration control process as it applies to the requirements in NFPA 805. Changes to the configuration control process are not listed as an implementation item in Table S-2.

Response:

DB-2050 will be added to Table S-2 as an implementation item to update the configuration control process. The item description is "Develop New NFPA 805 Control Procedures and Processes." A revision to the LAR Attachment S will be provided in a future transmittal.

FPE RAI 01

f. In LAR, Attachment L, Approval Request 1, the licensee stated in the discussion of defense-in-depth echelon 1 that procedure changes ensure future cable installations above suspended ceilings will be listed for plenum use or enclosed per NFPA 805, Section 3.3.5.1. The approval request and Table S-2 do not identify these procedure changes as an implementation item.

Response:

The existing implementation item DB-1964 identifies the specification changes to procedures to ensure future cable installation above suspended ceilings will comply with NFPA 805 Section 3.3.3. The implementation item LAR Section/Source reference of

Section 3.3.3 is an error and will be corrected to specify Sections 3.3.5.1 and 3.3.5.3. A revision to the LAR Attachment S will be provided in a future transmittal.

FPE RAI 02

NFPA 805, Section 3.4.1(c), requires that the fire brigade leader and at least two brigade members have sufficient training and knowledge of nuclear safety systems in order to understand the effects of fire and fire suppressants on nuclear safety performance criteria (NSPC). Section 1.6.4.1, "Qualifications," of NRC Regulatory Guide (RG) 1.189, Revision 2, "Fire Protection for Nuclear Power Plants," September 2009 (ADAMS Accession No. ML092580550), states, in part, that:

> **The brigade leader should be competent to assess the potential safety consequences of a fire and advise control room personnel. Such competence by the brigade leader may be evidenced by possession of an operator's license or equivalent knowledge of plant systems.**

In LAR, Attachment A (p. 67), the licensee stated that the fire brigade members are qualified through a training program that is maintained by the training organization. The licensee further stated that qualification requirements detailed in its plant procedures include knowledge of plant systems, layout, and general operation, as well as firefighting skills and attack strategy.

Describe the training that is provided to the fire brigade leader and members that ensures their ability to assess the effects of fire and fire suppressants on the NSPC.

Response:

The Fire Brigade Member Qualification Manual and the Fire Brigade Captain Qualification Manual provide summary qualifications for these positions.

- Member Qualification - Either be an operator, or have completed "Basic Operator Training."
- Captain Qualification (fire brigade leader) - Either hold a Senior Reactor Operator's license or has been a fire brigade member for two years and qualified as Equipment Operator III or greater.

The fire brigade members are equipped for and trained in the methods of fighting fires. From the Fire Brigade Training procedure, continuing training of fire brigade members shall, at a minimum, be provided in several areas including the following:

- Basics of fire propagation and behavior
- Portable extinguishers, hose, hydrants, and related suppression equipment

- Personal protective equipment, communications, and lighting
- Detection and protection systems
- Ventilation systems
- Fire assessment, suppression, and approaches
- Impact of fires on plant safety systems
- Annual self-contained breathing apparatus requalification training
- Fire command and leadership

The Fire Brigade Training and Conduct of Operations procedures require the fire brigade to be staffed by five qualified members (a Fire Captain plus 4 members) who are not relied upon for safe shutdown essential functions in the event of a fire emergency. It is the current practice to allow, at most, two of the fire brigade members per shift to be operators who are fire-brigade-qualified but not yet qualified as Non-Licensed Operator or higher.

A new Attachment S item DB-2062 will be added to require update to the Conduct of Operations procedure and provide clarification required for fire brigade qualifications necessary to meet Section 3.4.1(c). The LAR Attachment A, Table B-1 compliance statement for this record is hereby changed from “Complies” to “Will Comply With the Use of Commitment.” A revision to the LAR Attachment S for this new item will be provided in a future transmittal.

FPE RAI 03

In LAR, Section 4.6.2, “Overview of Post-Transition NFPA 805 Monitoring Program,” Phase 3, the licensee stated that the Electric Power Research Institute (EPRI) Technical Report 1006756, “Fire Protection Surveillance Optimization and Maintenance Guide for Fire Protection Systems and Features,” will be used as input for establishing reliability targets, action levels, and monitoring frequency for fire protection systems and features. The methodology in EPRI Technical Report 1006756 is a performance-based approach to determining acceptable surveillance frequencies for fire protection systems and features that are different than the surveillance frequencies described in the appropriate NFPA code or standard. This performance-based methodology is an alternative method to meeting the NFPA code or standard for the applicable NFPA 805, Chapter 3 requirements for the fire protection system and features. During the onsite audit, the licensee indicated that its current fire protection program revised the frequency of performing the inspection, testing, and maintenance of certain fire protection systems and features using a performance-based method and that some surveillance frequencies are different than the frequencies required by the appropriate NFPA codes and standards. However, the LAR does not appear to include an associated request for approval to use a performance-based method as required by 10 CFR 50.48(c)(2)(vii).

Clarify whether the LAR is requesting NRC approval for the use of a performance-

based method to determine or adjust fire protection surveillance frequencies in accordance with 10 CFR 50.48(c)(2)(vii) or explain why NRC approval is not needed. If NRC approval is requested, provide the information specified in RG 1.205, Section 2.2.2, "Performance-Based Methods for Fire Protection Program Elements and Minimum Design Requirements."

Response:

The compliance strategy for NFPA 805, Section 3.2.3(1), for the use of performance-based methods described in EPRI Technical Report (TR) 1006756 will be included as an approval request in LAR Attachment L. The approval request will be to permit the use of performance-based methods to establish the appropriate inspection, testing, and maintenance frequencies for fire protection systems and features required by NFPA 805. The approval request will discuss how the configuration satisfies the nuclear safety and radiological release performance goals, performance objectives, and performance criteria; maintains safety margins; and maintains fire protection defense-in-depth. A revision to the LAR Attachment L will be provided in a future transmittal.

Also, a LAR Attachment S item will be added to perform a review of the EPRI TR 1006756 program requirements and adjust the site program to address any differences identified, paying particular attention to reliability goals, setting failure criteria, and expectations for addressing a failed test (DB-2063). A revision to the LAR Attachment S will be provided in a future transmittal.

FPE RAI 04

NFPA 805, Section 3.3.5.3, requires that electric cable construction comply with a flame propagation test acceptable to the authority having jurisdiction. In LAR, Attachment A1, "Table B-1 Transition of Fundamental Fire Protection Program and Design Elements Worksheet," the licensee stated (p. 41) that it complies with NFPA 805, Section 3.3.5.3, by previous approval and referenced the NRC safety evaluation dated May 30, 1991 (ADAMS Accession No. ML033490026). The licensee indicated that the fire test used to qualify electric cable initially installed at the plant did not conform with the methodology in the Institute for Electrical and Electronics Engineers (IEEE) Standard No. 383, "IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices and Connections for Nuclear Power Generation Stations," but rather an alternative was used.

a. **In the NRC safety evaluation, the staff accepted the deviation based on the levels of fire protection (e.g., fire detection, fire suppression, and fire barriers) provided for safe shutdown systems and hazardous areas as described in Revision 1 of the licensee's fire area optimization report and in Revision 12 of the licensee's fire hazards analysis report. Describe that the fire protection systems and features credited in the NRC's previous approval of the cable tests.**

Response:

Some electrical cables installed at DBNPS were not tested to the standards referenced in NFPA 805 or the associated FAQ 06-0022. All fire tests for electrical cables installed as part of the original construction of DBNPS were conducted before the issuance of the IEEE 383 flame test; however, the fire tests done were comparable to IEEE 383. Additionally, some of the cables installed for low voltage non-safety related applications such as radiation monitors, GAI-Tronics, and television are not IEEE 383 qualified but are of thermoplastic construction, and are routed separately from cables for safety-related systems.

Electrical cables have been subsequently reviewed to determine the materials of construction and to classify the materials as either thermoset or thermoplastic. This cable type classification was used as input to the fire modeling and fire risk evaluations performed for the NFPA 805 project, and confirms that the amount of cable either known to be thermoplastic or modeled as thermoplastic due to unknown material type is less than 10 percent of the total cable in the cable trays. Therefore, based on the results of the fire modeling, subsequent PRA analysis, and fire compartment fire risk evaluation (FRE) reviews, only the fire protection detection and suppression systems listed in LAR Table 4-3 are required for NFPA 805 to reduce risk or for defense-in-depth due to electrical cables materials of construction.

The compliance statement for LAR Attachment A, Table B-1, Section 3.3.5.3 hereby deletes “Complies by Previous NRC Approval.”

Furthermore, LAR Attachment L, Approval Request 2 will be revised to state that the electrical cable material properties have been evaluated in fire modeling, PRA, and in the FREs to determine the locations where fire suppression or detection is required for risk reduction or defense-in-depth. A revision to the LAR Attachment L will be provided in a future transmittal.

FPE RAI 04

b. In LAR, Attachment A1 (p. 41), the licensee stated that there are limited amounts of thermoplastic cables and they account for less than 10 percent of the cables in trays. The licensee requested approval of a performance-based method in LAR Attachment L, Approval Request 2, for the installed thermoplastic cables. In its basis for maintaining the safety margin and defense-in-depth, the licensee stated that flame spread to adjacent cable trays in high density safety-related areas is reduced by the use of solid-bottom trays with a layer of ceramic fiber on top. The licensee further stated that the detailed fire models, as well as the PRA, identified and accounted for thermoplastic cable impacts. Identify the fire areas where this passive feature is credited in the fire risk evaluation to meet the NSPC.

Response:

Fire compartments that credit the passive feature are identified in the “Other” column of LAR Table 4-3 with an “R*.” The “R*” abbreviation at the end of LAR Table 4-3 is hereby updated to note that fire compartment II-01 (turbine building) does not include the ceramic fiber on top.

FPE RAI 05

Section 2.3.2 of RG 1.205 provides guidance on using previously approved alternatives to meet NFPA 805 requirements. The guidance indicates that licensees can use existing exemptions or deviations to demonstrate compliance with NFPA 805, provided the licensee acceptably addresses the continued validity of any exemption or deviation in effect at the time of the NFPA 805 licensee amendment application. In LAR, Attachments A1 and A2, the licensee used previous NRC approval as the basis for complying with certain NFPA 805, Chapter 3, design elements. However, the LAR did not specify if the previous NRC approval remains valid for NFPA 805, Sections 3.3.5.1, 3.3.5.3, and 3.8.1.

Discuss why the previous NRC approval remains valid for NFPA 805, Sections 3.3.5.1, 3.3.5.3, and 3.8.1, and for any other NFPA 805 sections where the LAR does not clearly specify that the previous NRC approval remains valid.

Response:

<u>NFPA 805 Section 3.3.5.1</u>:

The compliance statement for LAR Attachment A, Table B-1 for NFPA 805 Section 3.3.5.1 is hereby changed to “Complies” instead of “Complies by Previous NRC Approval.” The “Complies by Previous NRC Approval” was due to lack of a fixed fire suppression system above the suspended ceiling of the main control room, as required by Appendix A of Branch Technical Position (BTP) 9.5-1. The BTP acceptance criteria do not apply for transition to NFPA 805, so the “Complies by Previous NRC Approval” is not necessary for transition.

<u>NFPA 805 Section 3.3.5.3</u>:

Previous NRC approval for this section is addressed as part of the response to FPE RAI 04(a). “Complies by Previous NRC Approval” is deleted.

<u>NFPA 805 Section 3.8.1</u>:

The code of record for the fire alarm system is NFPA 72-1990 “National Fire Alarm Code.” The fire alarm system was updated with a new fire alarm panel installed to NFPA 72-1990 requirements by a previous modification, and compliance to NFPA

72-1990 is documented in an existing engineering equivalency evaluation (EEEE). This review supersedes the NFPA 72D-1975 "Standard for the Installation Maintenance and Use of Proprietary Protective Signaling Systems for Watchman Fire Alarm and Supervisory Service" compliance report discussed in an NRC safety evaluation dated May 30, 1991 for the previous fire alarm system. The EEEE determined that the fire alarm system complies with the code; however, some of the code sections were resolved by "complies with intent" with an explanation which demonstrates the functional equivalency to these requirements.

The NFPA 805 section 3.8.1 record was previously classified with both "Complies by Previous NRC Approval" and "Complies With Use of EEEE." The installed fire alarm panel system is not the same one previously evaluated by the NRC, and the EEEE review supersedes the previous code review. Therefore, the LAR Attachment A, Table B-1 compliance statement for this record is hereby revised to delete "Complies by Previous NRC Approval."

Extent of Condition Review:

A review of Chapter 3 Records where "Complies by Previous NRC Approval" had been credited identified that a total of 15 other sections were not included as part of Attachment K. These 15 sections were all related to NFPA code reviews and BTP 9.5-1 Appendix A positions, including configuration deviations, which were approved by an NRC safety evaluation dated May 30, 1991 (ML033490026). These 15 sections did not have a corresponding statement confirming that the basis for the previous NRC approval remains valid. Validation of the bases for these sections is as follows.

- An Attachment K Licensing Action 16 has been developed for the 11 sections listed below.
    - 3.3.1.2 (5): Flammable and Combustible Liquids
    - 3.3.4: Insulation Materials
    - 3.3.7.1: Storage of Flammable Gas
    - 3.3.8: Bulk Storage of Flammable & Combustible Liquids
    - 3.5.1: Water Supply; Adequate Reliability, Quantity & Duration
    - 3.5.2: Water Tanks
    - 3.5.10: Underground Yard Fire Main Loop
    - 3.5.13: Headers
    - 3.6.1: Standpipe and Hose Systems
    - 3.6.2: Water Flow and Nozzle Pressure
    - 3.6.4: Manual Fire Suppression

- The following two LAR Attachment A, Table B-1 sections are hereby changed from "Complies by Previous Approval" to "Will Comply with Use of Commitment." LAR Attachment S implementation items will be added. The NFPA code of record compliance reviews will be EEEEs. Compliance statements that were defined using the terminology of "complies with intent" are to be validated to demonstrate equivalency to NFPA code requirements. If a requirement cannot be demonstrated as functionally equivalent to the NFPA code requirements, a plant

modification will be initiated to ensure conformance or a NRC deviation request will be required.

  - 3.5.3: Fire Pumps (DB-2053)
  - 3.7: Fire Extinguishers (DB-2054)

- The following two LAR Attachment A, Table B-1 sections are hereby changed from "Complies by Previous NRC Approval" to "Will Comply with Use of Commitment." LAR Attachment S implementation items will be added. The NFPA code of record compliance reviews will be EEEEs. Prior applicable compliances that were defined using the terminology of "complies with intent" are to be validated to demonstrate that the fire detection or suppression system is equivalent or adequate for the hazard. If a fire detection or suppression requirement cannot be demonstrated as equivalent or adequate for the hazard (the system does not meet the performance requirements established by the Chapter 4 nuclear safety performance analyses), a plant modification will be initiated to ensure conformance or a NRC deviation request will be required.

  - 3.8.2: Fire Detection (DB-2055)
  - 3.9.1: Automatic and Manual Water-Based Fire Suppression Systems (DB-2056)

- An additional LAR Attachment S implementation item will be added for the following LAR Attachment A, Table B-1 section. The records are classified as "Complies With Use of EEEE;" however there are additional NFPA code of record compliance reviews that will be EEEEs. Prior applicable compliances that were defined using the terminology of "complies with intent" are to be validated to demonstrate that the fire door or damper is equivalent or adequate for the hazard. If a requirement cannot be demonstrated as equivalent or adequate for the hazard (passive fire protection does not meet the performance requirements established by the Chapter 4 nuclear safety performance analyses), then other corrective actions will be required.

  - 3.11.3: Passive Protection (DB-2057)

Licensing Action 16 (as noted above), Licensing Action 3 (for compartments E-01 & F-01), Licensing Action 11 (for record 3.11.1), and Licensing Action 12 (records for 3.3.12, 3.3.12(1), 3.3.12(2), 3.3.12(4), and 3.3.12(5)) has confirmed that the configuration and conditions still exist and that the basis for the previous NRC approval remains valid for Attachment K sections. A revision to the LAR Attachments K and S will be provided in a future transmittal.

FPE RAI 07

NFPA 805, Section 3.11.2, requires that fire barriers required by Chapter 4 include a specific fire resistance rating and be designed and installed to meet the specific

fire resistance rating using assemblies qualified by fire tests. In LAR Attachment A2, for various compartments and rooms (e.g., fire compartment V-01 room 405, fire compartment DG-01 room 402), the licensee stated that it will comply with previous approval for structural steel that could not be adequately fire proofed. The licensee stated that overhead sprinklers have been provided rather than applying the fireproof coating. In LAR, Table 4-3, the licensee identified fire protection systems and features that are credited to meet the NSPC required by NFPA 805, Chapter 4. The automatic sprinkler system in several rooms (e.g., rooms 402 and 405) were not identified as a system required for the previously approved licensing action. Clarify whether the sprinkler system in room 402, room 405, and other rooms identified in the previous approval are required systems credited to meet the NSPC.

Response:

The sprinkler systems in room 402, room 405, and other rooms identified below are required transitioning systems and credited to meet the NSPC. A revision to LAR Attachment K will be provided in a future transmittal to include new Licensing Action 15, which is based on the NRC acceptance letter dated July 17, 1980. This NRC letter acknowledged that the application of adequate fireproof coatings to the structural steel could not be applied in several rooms (rooms 208, 236, 303, 402, 405, and 427); therefore, the proposed alternative approach was to install overhead sprinkler systems. This alternative approach was accepted by the NRC.

The approval dated July 17, 1980, states:

> By letter dated May 15, 1980, you submitted a revision to the Davis-Besse Unit No. 1 Fire Hazards Analysis Report. The revision describes proposed changes to the fire protection methods for several rooms in which the fireproof coatings to be applied to structural steel, as called for in the original plan, could not be effectively installed. The affected areas are rooms 208, 236, 303, 402, 405, and 427. The alternative you have proposed is to install overhead sprinkler systems in these rooms rather than to apply the fireproof coatings.
>
> We find that the alternative method which you have proposed for the above rooms will provide the protection that we intended in the Davis-Besse Fire Protection Safety Evaluation Report dated July 26, 1979. The changes are therefore acceptable.

LAR Table 4-3 information for rooms 208, 236, 303, 402, and 405 is hereby revised by adding an “L” in the column for auto suppression to associate the sprinkler system with the new Licensing Action 15. For Room 427 in Fire Compartment DF-01, the “L” is already included in the auto suppression column for both Licensing Action 11 and the new Licensing Action 15. The LAR Attachment A2, Table B-1 Compliance Statement for these rooms hereby adds “Complies by Previous Approval” associated with Licensing Action 15.

FPE RAI 08

NFPA 805, Section 3.3.7.2, requires outdoor high-pressure flammable gas storage containers be located so that the long axis is not pointed at buildings. In LAR, Attachment A1 (p. 46), the licensee stated that the hydrogen and propane storage tanks are oriented with the long axis toward buildings. The licensee stated that it complies with NFPA 805, Section 3.3.7.2, using an existing engineering equivalency evaluation. The licensee stated that outdoor flammable gas storage orientation was evaluated for compliance with NFPA 50A-1973, "Standard for Gaseous Hydrogen Systems at Consumer Sites," and NFPA 58-2004, "Liquid Petroleum Gas Code," and concluded that the orientation of the tanks is acceptable. However, NFPA 805, Section 3.3.7.2, is not associated with compliance with other codes.

The information provided in the LAR is not sufficient to support the licensee's statement that it complies with NFPA 805, Section 3.3.7.2, by use of an existing engineering equivalency evaluation. The NRC staff reviewed the evaluation as part of its audit of licensee documents (see ADAMS Accession No. ML16075A111), and it appears that the evaluation included use of an unapproved risk-informed, performance-based method.

a. Provide a summary of the methods used in the evaluation, including any risk-informed, performance-based methods. For each risk-informed, performance-based method, clarify whether the LAR is requesting NRC approval in accordance with 10 CFR 50.48(c)(2)(vii) or explain why NRC approval is not needed. If NRC approval is requested, provide the information in RG 1.205, Section 2.2.2.

Response:

The EEEE for NFPA 805 Section 3.3.7.2, outdoor bulk high pressure flammable storage container locations and orientation, has been revised to clarify that the existing plant configuration is functionally equivalent to the code requirement. Therefore, the evaluation does not use a risk-informed or performance-based approach. The response to Section 3.3.7.2 remains unchanged as "Complies with use of EEEE."

The orientation technical basis for considering end failure of horizontal pressure tanks potentially causing property damage originated with NFPA 30. Safe distances from structures are applied for flammable gases in the specific codes, such as NFPA 50A and NFPA 58. The safe distance requirements address potential damage whether the horizontal high pressure container axis is parallel to the nearest structure or directed towards it; consequently, there is a reasonable safe distance based on the size of the container. The methodology used for the EEEE determined that spatial separation provides many multiples of the code-required minimum safe distance. Based on this separation and other factors such as substantial securement and installed relief devices prescribed by the installation code of record (NFPA 50A-1973 for hydrogen and NFPA 58-2004 for propane), the configuration provides "functional equivalency" to preclude

any potential damage to nearby structures. Since the evaluation determined that the location of the bulk storage flammable gas tanks was functionally equivalent to the NFPA 805 requirement, NRC approval is not required, and therefore, is not requested.

FPE RAI 08

b. Verify that the outdoor high-pressure flammable gas storage configurations meet the distance requirements in the code of record. Demonstrate that structures, systems, and components important to nuclear safety will not be adversely impacted by a failure of the outdoor high-pressure flammable gas storage containers.

Response:

The stand-off distance requirements for the location of the flammable gas bulk storage containers exceed their code-of-record required distances as follows:

- Hydrogen trailer: NFPA 50A requires a safe distance to the nearest structure of 25 feet. The closest end of the hydrogen trailer to the auxiliary and turbine buildings is approximately 239 feet. This exceeds the minimum stand-off distance by a multiple of approximately 9 times.

- Propane tanks: NFPA 58 requires a safe distance to the nearest structure of 10 feet. The closest end of the propane tank to the turbine building is 43 feet. This exceeds the minimum stand-off distance by a multiple of approximately 4 times.

In addition, the propane tanks are secured to concrete foundations and the hydrogen cylinders are secured between two heavy steel bulkheads that are mounted to the frame of the trailer. Also, the exposed surfaces of the structures are of rugged concrete and steel construction. Therefore, since the locations are multiple distances from other structures, are securely mounted, and have relief devices to preclude a container rupture, there is reasonable assurance that structures, systems, and components important to nuclear safety will not be adversely impacted.

FPE RAI 09

NFPA 805, Section 3.3.7.1, requires that storage of flammable gas be located outdoors, or in separate detached buildings, so that a fire or explosion will not adversely impact systems, equipment, or components important to nuclear safety. This section also requires NFPA 50A be followed for hydrogen storage. In LAR, Attachment A1 (p. 45), the licensee stated it complies in that flammable gas storage is "located outdoors, or in separate detached buildings." The licensee also stated that it complies by previous NRC approval and included an excerpt from the NRC safety evaluation dated May 30, 1991.

The LAR does not clearly distinguish between the parts of NFPA 805, Section 3.3.7.1, that DBNPS complies with and the parts where previous NRC approval is credited. The LAR also does not sufficiently describe the specific NFPA code conformance analysis.

Identify the parts of NFPA 805, Section 3.3.7.1, that DBNPS complies with and the parts where previous NRC approval is credited. Describe the applicability of the previous NRC approval to compliance with NFPA 805, Section 3.3.7.1, that clearly connects the approval with the criteria in this element. Where previous NRC approval is credited to demonstrate compliance, address why the approval remains valid. In addition, clarify whether the flammable-gas storage is located outdoors or is in one or more separate attached buildings.

Response:

The bulk storage of hydrogen is located outside in the north yard area, and propane tanks are located outside in the west yard. Hydrogen cylinder storage is also located outside in the yard, west of the low level radwaste storage facility building.

The hydrogen storage installation was evaluated to the requirements of NFPA 50A-1973, as reported to the NRC in Attachment 16 of the DBNPS letter dated July 31, 1989. NFPA 50A provides the installation requirements for hydrogen systems and storage, which are to prevent a hydrogen fire and explosion by requiring minimum safe distances, design criteria, and maintenance practices; therefore ensuring that systems, equipment, or components important to nuclear safety are not impacted. The NRC review of the installation resulted in a request for additional clarification of maintenance-related deviations. The response to those deviations was provided in a DBNPS letter dated October 11, 1989. The NRC acceptance of the NFPA 50A installation code compliance review was documented in the NRC safety evaluation dated May 30, 1991. Excerpts of the letters are as follows:

<u>July 31, 1989 DBNPS letter</u>

> 1.0 INTRODUCTION
> The National Fire Protection Association Standard for Gaseous Hydrogen Systems at Consumer Sites (NFPA 50A) was reviewed for compliance. This review was performed to document the adequacy of the Davis-Besse fire protection features.
>
> 2.0 SUMMARY
> One area of the Standard with which Davis-Besse does not comply involves the testing of systems after installation of components. The problem is with procedures used to change out portable cylinders or mobile units do not require leak testing of the threaded connections that are used at the cylinder or unit connections. A procedure change request was initiated to resolve this deviation.

October 11, 1989 DBNPS letter

NFPA 50A- Request [NRC]:
Standard section 8.81 of the NFPA 50A review discussed the hydrogen tanks and mobile units system maintenance and stated the other system components are maintained in accordance with plant procedures. A request was made to more clearly explain the other system components and their procedures.

Response [Davis-Besse]:
The system components include various devices which have individual preventive maintenance procedures to maintain the components. The interval of calibration/checking varies but the longest interval for the installed components on both the turbine and makeup tank hydrogen systems is 60 months…

May 30, 1991 NRC Safety Evaluation

Section I, Introduction:
Consequently, the licensee provided supplemental information which explicitly identified deviations from staff fire protection guidelines and the relevant National Fire Protection Association (NFPA) Standards and provided justification as to why these deviations were not safety significant. The staff considers these deviations to fall within two categories. The first are those deviations which represent minor variances. These minor variances and those features of the Davis-Besse Fire Protection Program which conform with NRC and NFPA criteria are described comprehensively in the documents cited above and are not discussed in detail in this safety evaluation since the staff finds that the minor deviations are acceptable. The second category are those deviations which are not considered by the staff to be minor variances and for which there was, initially, some concern on the part of the staff regarding the licensees's justification of its technical approach. The staff's basis for accepting these latter deviations is contained in the following evaluation.

Section IV:
During its review of the NFPA conformance analyses, the staff requested clarification on a number of issues including the lack of automatic shutoff valves for some of the flammable liquid tanks, maintenance of hydrogen system features…. The licensee provided satisfactory responses to these requests in its letter dated October 11, 1989. On this basis, the staff concludes that this issue is resolved.

The new Attachment K Licensing Action 16 (discussed in the response to FPE RAI 05) will provide validation that the existing plant conditions are consistent with the previous NRC approval.

The statement in LAR Attachment A, Table B-1 section 3.3.7.1 that flammable gas cylinders may be stored in separate detached buildings under the "Complies" compliance basis is hereby deleted. The following wording under the "Complies by Previous Approval" compliance basis is hereby deleted: "… the lack of automatic shutoff valves for some of the flammable liquid tanks…;" and "… ventilation in Seal Oil Room No. 333." This wording is irrelevant to the record.

Safe Shutdown Analysis (SSA) RAI 01

NFPA 805, Section 4.2.1, requires that the effects of fire suppression activities on the ability to achieve the NSPC be evaluated. In LAR, Attachment C (p. 308), the licensee stated that for fire compartment OS (outside areas), impacts to equipment due to activation of the automatic suppression located in OS-02, rooms 001 and 002, are beyond the scope of the detailed fire model analysis and are to be treated as an uncertainty. Clarify the basis for the statement and describe if the fire suppression activities in these rooms will adversely affect the ability to achieve the NSPC.

Response:

LAR Attachment C is hereby revised by removing the referenced sentence on page 308: "Impacts to equipment due to activation of the automatic suppression located in OS-02, Rooms 001 and 002, are beyond the scope of the Detailed Fire Model analysis and are to be treated as an uncertainty."

Fire Compartment OS-02 consists of the station blackout (SBO) diesel generator (DG) building, which includes Rooms 001 (fuel oil storage room) and 002 (DG room). Should there be a fire in the SBO DG building, the emergency diesel generators are still available to safely shut down the plant. The equipment in these rooms are not required to achieve the NSPC. There are no variances from deterministic requirements (VFDRs) associated with these rooms.

SSA RAI 02

As described in NFPA 805, Chapter 1, the standard applies to plants during all phases of plant operations, including shutdown. In LAR, Section 4.3.2, the licensee summarized the results of the evaluation process for non-power operations (NPO). Provide the following information pertaining to NPO discussions provided in the results discussion in LAR, Section 4.3, and LAR, Attachment D.

a. LAR, Section 4.3.2, states that for those components which had not been previously analyzed in support of the at-power analysis or whose functional requirements may have been different for the NPO analysis, cable selection was performed in accordance with the

project procedures. Provide a general description of components that are different for the NPO analysis.

Response:

During the development of the NPO modes transition report, the plant components identified as required to support the success paths for the required key safety functions (KSF) for NPO were determined. These components (intersystem connections between NPO systems and safe shutdown systems and instrumentation) associated with NPO KSFs were compared to the safe shutdown (SSD) required components that had been previously analyzed.

The identified NPO components that were not previously analyzed as SSD components, or for which their NPO position is different from SSD positions, were selected for cable identification and cable analysis. These components include, but are not limited to, the following:

- System interface valves between the decay heat (DH) removal (DHR) system and the reactor coolant system (RCS).
- Cooling system and support components for the DH removal system.
- Associated instrumentation for these systems.
- RCS inventory – Borated water storage tank (BWST) level instrumentation and flow paths from the BWST to the RCS.
- Reactivity control - Borated water sources and flowpaths (boric acid addition system (BAAS) train 1 or 2, high pressure injection (HPI) train 1 or 2, makeup train 1 or 2, and clean waste receiver tank (CWRT) train 1 or 2), and nuclear instrumentation.

Below is a listing of components identified as requiring cable selection for NPO.

Component ID	Component Description	NPO Position	SSD Position
DH13B	DH13B DH Cooler 1 Bypass Flow Control Valve	Open	Closed
LI10577A	Reactor Coolant Level Indicator	On	NA
LI10577B	Reactor Coolant Level Indicator	On	NA
TEDH8B	TEDH8B DH Pump1 Suction Temperature	On	NA
TEDH8A	TEDH8A DH Pump1 Suction Temperature	On	NA
TEIM07E	Incore Thermocouple E7	On	NA
TEIM07M	Incore Thermocouple M7	On	NA
DH13A	DH13A DH Cooler 2 Bypass Flow Control Valve	Open	Closed
DH830	DH830 DH Cooler 1/2 Cross-Connect	Open/Closed	NA
RC239B	PZR liquid sample	Open	Closed
CS1530	[Containment] CTMT Spray Discharge	Open	Closed
CS1531	CTMT Spray Discharge	Open	Closed

Component ID	Component Description	NPO Position	SSD Position
P56-1	CS Pump-1	On	Off
P56-2	CS Pump -2	On	Off
DH7A	BWST Isolation Valve Line 2	Open	Closed
DH7B	BWST Isolation Valve Line 1	Open	Closed
DH9A	DH Pump Suction From Emergency Sump	Open	Closed
DH9B	DH Pump 1 Suction From Emergency Sump	Open	Closed
LI1403	Surge Tank Level Indicator	On	NA
TS4688	Temperature Switch	On	NA
TS4698	Temperature Switch	On	NA
LI1525A	BWST Level Indication	On	NA
LI1525B	BWST Level Indication	On	NA
LI1525C	BWST Level Indication	On	NA
LI1525D	BWST Level Indication	On	NA
MU40	Batch Feed Isolation Valve	Open	NA
P49-1	CWRT Pump-1	On	NA
P49-2	CWRT Pump-2	On	NA
WC1751	CWRT 1 Out Flow Control Valve (RWCP)	On	NA
WC1752	CWRT 2 Out Flow Control Valve (RWCP)	On	NA
WC1761	BA EVAP From CWRT Inlet Flow Control Valve	On	NA
WC3526	Clean Waste Booster Pumps Bypass Valve	Open	NA
WC3527	Clean Waste Booster Pump Suction Flow Control Valve	Closed	NA
MU12A	Makeup Filter 1 Inlet Isolation	Open/Closed	NA
MU12B	Makeup Filter 2 Inlet Isolation	Open/Closed	NA
P38-1	BA Pump 1	On	NA
P38-2	BA Pump 2	On	NA

The identified NPO components and their relationship to SSD, and the need to perform cable identification and cable analysis are listed in Attachment 2, "Component Identification and Support System Dependencies," in the Non-Power Operational Modes Transition Report.

SSA RAI 02

b. LAR, Attachment D, states the licensee followed the guidance of Frequently Asked Question (FAQ) 07-0040, "Non-Power Operations Clarifications," dated August 11, 2008 (ADAMS Accession No. ML082200528). In LAR, Attachment D, the licensee stated that 46 fire compartments were found to have pinch points resulting in the potential loss of one or more key safety function success paths. For fire compartments with pinch points that result in a complete loss of any key safety function, describe the methods that will be used to ensure that the NSPC are achieved and maintained.

Response:

Attachment 7 of the DBNPS Non-Power Operational Modes Transition Report contains the "Key Shutdown Defense in Depth Functions Affected by Fire Compartment" table. Analysis results for each fire compartment indicates with a “Y” where the fire has a potential to cause loss of all success paths for one or more KSF or performance goal. The NPO Modes Transition Report identified 86 pinch points for a potential loss of a KSF. Of these pinch points, 25 percent were for a potential loss of the decay heat removal KSF, 50 percent were for a potential loss of the inventory and pressure control KSF, 15 percent were for a potential loss of the reactivity KSF, and the remaining 10 percent were for a potential loss of the support (electrical power availability) KSF.

In order to preclude the potential loss of KSF, the analysis information, as recommended by the DBNPS NPO Modes Transition Report, will be incorporated into plant procedures as stated in Attachment S implementation item DB-1908, “Revise Procedures and Conduct Training to Implement NPO Requirements for NFPA 805.” This item will add the necessary steps to ensure that the KSFs are achieved and maintained as appropriate administrative procedures.

SSA RAI 02

c. During NPO modes, spurious actuation of valves can have a significant impact on the ability to maintain a key safety function. Provide a description of any actions (e.g., pre-fire rack-out, actuation of pinning valves, and isolation of air supplies) being credited to minimize the impact of fire-induced spurious actuations on power-operated valves (e.g., air-operated valves and motor-operated valves) during NPO.

Response:

Revisions to applicable plant procedures will incorporate the insights and strategies documented in the NPO Modes Transition Report to mitigate potential effects of a fire event. Potential actions for defense-in-depth strategies include pre-positioning of valves or removing power from valves. Procedure updates to implement NPO requirements are a part of implementation item DB-1908 in LAR Attachment S, Table S-2.

SSA RAI 02

d. The description of the NPO review in the LAR does not identify locations where key safety functions are achieved by recovery actions (RAs) or for which instrumentation not already included in the at-power analysis is needed to support RAs required to maintain safe and stable conditions. Identify those RAs and instrumentation relied upon in the NPO, and describe how RA feasibility is evaluated.

Include in the description whether these variables have been or will be factored into operator procedures supporting these actions.

Response:

At this time there are no credited recovery actions to achieve a KSF, therefore no KSF recovery action feasibility review has been performed. Procedure updates to implement NPO requirements are a part of implementation item DB-1908 in LAR Attachment S, Table S-2.

The following are examples of instruments that could be credited for potential NPO recovery actions, including, but not limited to:

Decay Heat Removal
- LI10577A and LI10577B (reactor coolant system level)
- TEDH8B (decay heat pump 1 suction temperature)
- TEIMO7E and TEIM07M (incore thermocouples)
- LI1402 and LI1403 (surge tank level)
- TS4688 and TS4698 (temperature switches)

Reactivity Control and Reactor Coolant System Inventory
- NINI1 and NINI2 (source range nuclear instrumentation)
- NI5874A and NI5875A (gamma metrics subcritical monitor)
- LI1525A, B, C, and D (borated water storage tank level)
- LI1894 and LI1898 (local level indicator, clean waste receiver tank)
- FIMU22 (local flow indicator for discharge of boric acid pump)

Process Monitoring (in addition to instrumentation listed above)
- FI 6425 and FI6435 (makeup pump injection flow)

SSA RAI 03

In LAR, Section 4.2.1.1, the licensee indicated that the method used to perform the NSCA either meets the NRC-endorsed guidance in Chapter 3 of Nuclear Energy Institute (NEI) 00-01, Revision 2, "Guidance for Post-Fire Safe Shutdown Circuit Analysis" (ADAMS Accession No. ML091770265), or meets the intent of the endorsed guidance with adequate justification. NEI 00-01, Section 3.5.1.1, states that for ungrounded direct-current circuits, multiple shorts-to-ground are to be evaluated for their impact and a single hot short from the same power source is assumed to occur unless it can be demonstrated that the occurrence of a same source short is not possible in the affected fire area. In LAR, Attachment B, "Table B-2 Nuclear Safety Capability Assessment, Methodology Review," the licensee stated that it aligns with the intent of the NEI 00-01 guidance. However, in its discussion relative to multiple high-impedance faults, the licensee stated that existing circuit analyses relied on ungrounded direct-current circuits of the proper polarity not faulting in some cases. Provide the technical basis for the

assumption that ungrounded direct-current circuits of the proper polarity will not fault and discuss how this methodology meets the intent of the guidance in NEI 00-01 with respect to circuit failures for ungrounded direct-current circuits.

Response:

The statement that "…existing circuit analyses relied on ungrounded direct-current circuits of the proper polarity will not fault in some cases…" is in reference to certain cases of fire damage documented in the Fire Hazards Analysis Report (FHAR) for compliance with 10 CFR 50 Appendix R regulations. For these cases, the FHAR documented the following resolutions as the technical basis for compliance with 10 CFR 50 Appendix R:

1. RCS high point vent valves: "In order for spurious actuation of valves RC4608A, B or RC4610A, B to occur, 2 concurrent hot shorts of the proper polarity without grounding to the C cable would be required. Since these valves are aligned in series, an inadvertent breach of a High/Low Pressure Interface could only occur in the event of 4 specific concurrent hot shorts of the proper polarity." Therefore, it was not considered credible for the Appendix R analysis that a flow path could spuriously open.
2. RCS high point vent valves (containment penetration areas): "Spurious operations of both valves requires multiple hot shorts of the right polarity. This is incredible because there is not available power source for RC4610B." Therefore, it was not considered credible for the Appendix R analysis that a flow path could spuriously open.
3. Various components: "The solenoid/relay and isolating contact(s) is (are) outside the fire area and spurious operation due to a hot short on one side of the circuit and a simultaneous ground on the other is considered incredible." Therefore, it was not considered credible for the Appendix R analysis that a spurious operation could occur.

Therefore, this methodology aligns with the intent of the guidance in NEI 00-01 with respect to circuit failures for ungrounded direct-current circuits. These resolution categories will not be transitioned to the NFPA 805 licensing basis. When the component is required to maintain long term safe and stable conditions at hot standby, but had cable damage, a VFDR was assigned and a fire risk evaluation was performed.

SSA RAI 04

In LAR, Section 4.2.1.1, the licensee indicated that the method used to perform the NSCA either meets the NRC-endorsed guidance in Chapter 3 of NEI 00-01, Revision 2, or meets the intent of the endorsed guidance with adequate justification.

NEI 00-01, Section 3.5.1.2, describes the spurious operation criteria to address the effect of multiple spurious operations. In LAR, Attachment B (p. 92), the

licensee stated that it is, "Not in Alignment, but no Adverse Consequences," with the NEI 00-01 guidance. However, in the alignment basis, the licensee stated that emergent industry issues related to multiple spurious operations are being addressed during the transition to NFPA 805, and that the issues were reviewed and analyzed per the guidance provided in NEI 00-01 and NEI 04-02, Revision 2, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)" (ADAMS Accession No. ML081130188). In LAR, Attachment F, the licensee stated it conducted the review of multiple spurious operations in accordance with NEI 04-02, Revision 1, and RG 1.205, Revision 0, as supplemented by FAQ 07-0038, "Lessons Learned on Multiple Spurious Operations" (ADAMS Accession No. ML110140242).

Describe the specific methodology that is not in alignment with NEI 00-01, and either provide justification for deviating from the NEI guidance or discuss how the NSCA methodology will align with the guidance in NEI 00-01, Revision 2.

Response:

The LAR Attachment B, "Alignment Statement" for Section 3.5.1.2, "Spurious Operation Criteria," that reads "Not in Alignment, but No Adverse Consequences," is incorrect and is hereby changed to "Aligns."

The LAR Attachment B, Section 3.0, "Deterministic Methodology," contains the following statements:

> The Davis-Besse specific analysis approved by the NRC is contained in the Fire Hazard Analysis Report (FHAR). The FHAR methodology in general is consistent with the guidance provided in NEI 00-01 with some exceptions:

and

> Emergent industry issues related to operator manual actions (OMAs) and multiple spurious operations (MSOs) are being addressed during the transition to NFPA 805.

These statements are referring to the plant-specific methodology used at the time the FHAR was created. The specific analysis methodology and results for MSOs are documented in Attachment F of the LAR, "Fire-Induced Multiple Spurious Operations Resolution." This methodology fully aligns with all requirements of NEI 04-02, Revision 1, and Regulatory Guide 1.205, Revision 0, as supplemented by FAQ 07-0038, Revision 3.

SSA RAI 05

NFPA 805, Section 4.1, requires that once a determination has been made that a fire protection system or feature is required to achieve the performance criteria of Section 1.5, its design and qualification shall meet the applicable requirement of Chapter 3. In LAR, Table 4-3, the licensee identified that 1-hour fire rated electrical raceway fire barrier systems (ERFBS) are credited in fire compartment A-08, for example, to protect cables that are required to achieve and maintain the NSPC. In LAR, Table 4-3, under the heading "Other," the licensee also identified that 1-hour fire rated ERFBS and cable tray systems that are protected on three sides by metal covers and on top by ceramic fiber (e.g., Kaowool) are credited to protect selected cables for risk reduction.

Clarify the basis for differentiating these "other" types of fire barrier features from the fire barriers credited as an ERFBS, and clarify whether these "other" types of fire barriers systems will meet NFPA 805, Section 3.11.5, or the appropriate Chapter 3 requirement. For those types of fire barrier systems that do not meet NFPA 805, Section 3.11.5, describe and justify how the risk reduction credit will be determined.

Response:

NFPA 805 Section 3.11.5 ERFBS are identified in LAR Table 4-3 under the column labeled "ERFBS."

In Fire Compartment A-08, along with other fire compartments, electrical cable raceways are protected from fire by the following three configurations:

- Selected electrical raceways are configured and credited for NFPA 805 transition with ERFBSs;
- Other electrical raceways in Fire Compartment A-08 (along with other fire compartments) are protected by a cable tray configuration consisting of three solid metal sides (two solid sides and solid bottom) with a ceramic fiber blanket over the top; except
- Fire Compartment II-01 has electrical raceways consisting of three solid metal sides (two solid sides and solid bottom) and no ceramic fiber over the top of the electrical cables.

The fire barriers identified in the "ERFBS" column of Table 4-3 are tested configurations, and the insulation and fire properties provide protection from fire damage to the electrical cables for a time duration greater than or equal to its configuration acceptance test duration.

The electrical raceway configuration of three-sided metal (sides and bottom) and topped with ceramic fiber material is not a tested configuration, but provides reasonable assurance to limit the electrical cable jacketing from becoming exposed to and ignited during a fire. The three metal sides provide reasonable assurance to limit the electrical

cable jacketing from becoming exposed to fires initiating at lower elevations by acting as radiant heat shielding to delay the propagation of cable damage in the electrical raceways above the fire. Therefore, it is used within the analysis to delay subsequent electrical tray ignition. This configuration is not identified in the "ERFBS" column of Table 4-3, but is identified in the "Other" column, along with an appropriate note.

These configurations were evaluated for reducing cable damage during the compartment fire modeling, PRA, and fire risk evaluations. Per Appendix Q.2.2 of NUREG/CR-6850, cable trays with bottom covers have been credited to prevent damage and ignition by 20-minutes for thermoset cables. In accordance with the detailed fire modeling procedure, a 4-minute delay has been credited to prevent damage and ignition for cables trays containing thermoplastic or Kerite cables. Cable tray covers have only been credited if they have been evaluated and confirmed to be outside the ZOI of any high hazard event (HEAF, hydrogen or transformer explosion), which may cause mechanical damage to the cover.

LAR Table 4-3 for Fire Compartment A-08 mistakenly accounted for the ERFBS in both the "ERFBS" and "Other" columns. It should only have been listed in the "ERFBS" column. LAR Table 4-3 is hereby changed to remove "R3" in the "Other" and "Comments" columns for Fire Compartment A-08.

SSA RAI 06

NFPA 805, Section 4.3.2, allows the use of RAs to demonstrate availability of a success path for the NSPC. Feasibility of the RA can be demonstrated by applying the feasibility criteria and guidance in FAQ 07-0030, "Establishing Recovery Actions" (see ADAMS Accession Nos. ML103090602 and ML110070485). One of the criteria in FAQ 07-0030 is to perform periodic drills that simulate the conditions to the extent practical. In LAR, Attachment G, "Recovery Actions Transition," the licensee stated that it will update the fire brigade drills after completion of the procedures and training, and identified this as an Implementation Item DB-1941 in LAR, Attachment S, Table S-2. The activities performed by the fire brigade are generally focused on fire fighting and suppression of a fire in the plant. Discuss the proposed changes to the fire brigade drills and address whether any fire brigade members are credited to perform any RAs required to meet the NSPC.

Response:

Recovery actions credited to meet NSPCs will not be performed by any fire brigade member actively engaged in firefighting or post-fire cleanup. All credited recovery actions will be performed by assigned operations department personnel, and they will be directed by a licensed operations unit supervisor using approved safe shutdown procedures.

Implementation item DB-1941 includes non-fire brigade changes to safe shutdown

operations procedures and incorporate all new recovery actions required for transition to NFPA 805. In addition, it identifies updates to fire brigade training materials and drills. The following implementation items also track changes to fire brigade materials and are documented in LAR Attachment S, Table S-2:

- DB-0341
- DB-0538
- DB-0557
- DB-1074
- DB-1093
- DB-1095

These implementation items are collectively summarized as follows:

The fire brigade pre-fire plans and training materials will be updated to include instruction for the containment and monitoring of potentially-contaminated fire suppression water and products of combustion. Updates of the pre-fire plans and training lesson plans will include guidance for the judicious use of fire hose spray and an awareness item for the potential of flooding within the fire compartment and water run-off to adjacent areas. There will also be updates of pre-fire plan information to include safe shutdown components and power supplies identified in the safe shutdown analysis. This may be used by operators as an aid to identify fire-affected equipment.

SSA RAI 07

The nuclear safety goal described in NFPA 805, Section 1.3.1, requires that reasonable assurance be provided such that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition. In LAR, Section 4.2.1.2, the licensee described the methods used to maintain safe and stable hot standby conditions. The licensee stated that inventory makeup to the reactor coolant system (RCS) will be required to account for nominal RCS leakage and RCS shrinkage due to cooldown, as well as reactor coolant pump (RCP) seal injection. In LAR, Attachment G, the licensee identified a number of RAs associated with variances from deterministic requirements (VFDRs) that involve a loss of RCP seal cooling via the seal injection flow path (e.g., VFDRs DB-1029, DB-1381, DB-1383). The licensee stated that within 8 hours, the RA will involve manually aligning seal injection flow to all the RCP seals, manually align component cooling water to the RCP thermal barrier, or cooling down the RCS to place the plant between 280 degrees Fahrenheit (°F) and 350 °F. The first two options reestablish RCP seal cooling to prevent a loss-of-coolant accident through the seal.

Discuss how maintaining the RCS temperature between 280 °F and 350 °F is sufficient to prevent RCP seal failure. Clarify if the operator activities to cool down the plant to an RCS temperature between 280 °F and 350 °F are required to maintain safe and stable conditions. Describe the specific RAs required to be

performed, if any, to achieve and maintain this condition.

Response:

Maintaining safe and stable hot standby conditions requires the average reactor coolant temperature to remain greater than or equal to 280 ºF per Technical Specifications Table 1.1-1. The 350 ºF upper temperature limit is from the ethylene propylene elastomer Flowserve® N-9000 RCP seals that were installed during previous refueling outages. This seal design has been tested to withstand RCS temperatures for greater than eight hours without seal cooling, and the elastomers are rated to approximately 350 °F before breakdown. Therefore, an RCS temperature range between 280 °F and 350 °F maintains the plant in safe and stable hot standby conditions and prevents RCP seal failure.

Based on manufacturer testing, the elastomer components of the N-9000 seals should not experience any significant failure for short periods of time (24 hours or less), even if RCS temperature remains high without seal cooling. Therefore, operator activities to cool down the plant to an RCS temperature between 280 °F and 350 °F are not required to maintain safe and stable conditions. This is considered a defense-in-depth recovery action.

No special plant alignment is needed to cool down below 350 °F, and the decay heat removal system is not needed. Any recovery actions associated with systems or components required for this cooldown would also be required in order to maintain safe and stable conditions and are documented in LAR Attachment G. Therefore, no specific recovery actions are required to be performed to achieve and maintain this condition.

The recovery actions for loss of RCP seal cooling were marked as “Risk Reduction.” However, the only action credited for risk reduction is tripping the RCPs. When the trip can be accomplished from the control room, this action is not included in LAR Attachment G. When RCP trip circuitry is fire-affected, associated recovery actions are documented elsewhere in LAR Attachment G. Therefore, the three options to cool the RCP seals are considered defense-in-depth recovery actions.

SSA RAI 08

NFPA 805, Section 4.2.4, requires that when the use of RAs has resulted in the use of a performance-based approach, the additional risk presented by their use is to be evaluated. In LAR, Attachment G (p. G-51), the licensee identified an RA required for risk reduction in fire compartment G-02, which is associated with tripping the auxiliary feed pump 1 and its solenoid control valve, and the licensee indicated that this RA is associated with VFDR DB-1184. However, this VFDR is not identified in the NSCA summary for fire compartment G-02 in LAR, Attachment C.

Clarify whether VFDR DB-1184 is applicable to fire compartment G-02. If so,

confirm that the RA identified for fire compartment G-02 is required and that the additional risk presented by its use was evaluated in this fire compartment.

Response:

A review of the LAR Attachment C, fire compartment G-02, and the associated fire risk evaluation indicate DB-1184 is required and evaluated for fire compartment G-02.

The recovery action required by DB-1184 is contained in an attachment to the procedure for a serious station fire. A FAQ 07-0030 feasibility analysis was performed by an expert panel, and no issues were identified that would prevent performance of these component manipulations for this fire compartment. The risk of this VFDR was evaluated in the fire PRA, and therefore the additional risk of this recovery action is included in the values previously reported for fire compartment G-02 in LAR, Table W-3.

SSA RAI 09

NFPA 805, Section 4.2.4, requires that when the use of RAs results in the use of a performance-based approach, the additional risk presented by their use be evaluated. In LAR, Attachment C, the licensee identified RAs as required to meet the risk, safety margin, and defense-in-depth acceptance criteria in the resolution of VFDR DB-1923 in fire compartment II-01 and VFDRs DB-1227 and DB-1268 in fire compartment V-01. However, the licensee did not identify the RAs in LAR, Attachment G. Provide any additions to LAR Attachment G, as appropriate, and address the following:

a. Clarify whether the RAs are required for risk or defense-in-depth to resolve VFDRs DB-1923, DB-1227, and DB-1268. If so, verify that the RA is feasible, and verify that the additional risk of their use has been included in the fire area risk reported in LAR, Attachment W.

Response:

The usual recovery action for DB-1923 is to trip the turbine locally, because the VFDR renders turbine trip from the control room unavailable. However, for fire compartment II-01, the local manual turbine trip on the turbine front standard is located within the fire compartment. Therefore, the analyzed action for this VFDR in II-01 is to close the main steam isolation valves (MSIVs) and MSIV bypass valves from the main control room. Because this is a control room action, it is not a recovery action, and thus would not be included in Attachment G. This action is a defense-in-depth action in the fire risk evaluation.

After review of the LAR Attachment C and the cable analysis for V-01, it was determined DB-1227 and DB-1268 are not required for fire compartment V-01, because they were duplicates of the V-01 associations to DB-1217 and DB-1258. DB-1217 and DB-1258 are Train 1 VFDRS. DB-1227 and DB-1268 are Train 2 VFDRs. The recovery

action for these VFDRs is to “Trip both Containment Spray Pumps.” There is no need for two VFDRs performing the same recovery action in the same compartment for fire induced-failures. The associations for V-01 to DB-1227 and DB-1268 had been removed from Attachment G, and are hereby removed from the “Comments” section for fire compartment V-01 in LAR Attachment C.

SSA RAI 09

b. Confirm that there are no other VFDRs in the NSCA that credit RAs to meet the NSPC that are not identified in LAR, Attachment G. Identify any discrepancies, clarify whether the RA is required to meet the NSPC, and verify that the RA is feasible.

Response:

A review of the LAR Attachment C and Attachment G has confirmed that there are no discrepancies with the VFDRs identified with recovery actions credited to meet the risk, safety margin, and defense-in-depth acceptance criteria between Attachment C and those identified in Attachment G, other than the previously identified VFDRs in the response to SSA RAI 09.a (DB-1923, DB-1227, and DB-1268).

SSA RAI 10

NFPA 805, Section 4.2.1, requires that one success path necessary to achieve and maintain the NSPC be maintained free of fire damage by a single fire. In LAR, Attachment C, for all fire compartments, the licensee referenced VFDR DB-2012, which states:

> **Fire damage to installed makeup pumps could result in loss of ability to maintain RCS Inventory and Pressure. This could challenge the NSPC for Inventory and Pressure. This is a separation issue.**

The LAR states that this VFDR will be corrected by plant modification ECP 13-0463, which installed additional RCS charging pumps, connections, and associated auxiliaries. The LAR further indicates that these modifications are associated with the development of diverse and flexible coping strategies (FLEX). However, the LAR does not provided sufficient information on how the disposition of this VFDR meets the NSPC. Provide the following information related to the disposition of VFDR DB-2012:

a. Discuss the extent to which fire damage to installed makeup pumps and other systems and components would affect RCS inventory and pressure control (e.g., spurious operations that involve RCS inventory loss, shrinkage due to fire-induced secondary system cooldown).

Response:

Fire damage to the installed makeup pumps and associated components could cause RCS overfill or overpressure due to spurious operation, and, if unmitigated, could challenge the power-operated relief valves (PORVs) or primary safety valves. If the makeup pumps fail to operate, loss of RCS makeup capabilities could result.

With the installed makeup pumps unavailable due to loss of offsite power (LOOP) or fire damage, failure to isolate letdown and any other RCS diversion flow path, or a RCP seal failure could cause a loss of RCS inventory, and eventually result in fuel damage.

Any concurrent secondary overcooling event, caused by fire damage (such as a failed open MSIV, failed open main steam atmospheric vent valve, or spurious overfeed of steam generators) could cause a RCS shrinkage (loss of level control) due to excessive or uncontrolled RCS cooldown.

SSA RAI 10

b. Describe the modifications associated with ECP 13-0463. Describe the RAs that will be required to use the additional RCS charging pumps. Explain how the modifications and RAs will adequately meet the NSPC associated with RCS inventory and pressure control.

Response:

FLEX RCS Pump 1 is located in the auxiliary building at the 545 foot elevation in the equipment and pipe chase. A suction line to the pump is connected via hose to the existing borated water storage tank (BWST) suction line. The pump's discharge is connected via high pressure hose to an existing high pressure injection (HPI) discharge line. Hard piping is routed such that the hose connection points are readily accessible after a beyond-design-basis external event (BDBEE).

The alternate FLEX RCS makeup, utilizing clean waste receiver tank (CWRT) 1-1, also consists of a pump, valves, piping, hose, and high pressure connections. FLEX RCS pump 2 is permanently staged in the auxiliary building at the 545 foot elevation in the CWRT No. 1 room. A penetration containing hard piping is installed to support the FLEX mitigation strategy. A hose connects the hard piped tie-in to the penetration piping and from the penetration piping to the pump inlet. The pump outlet discharges via high pressure hose connected to a point on the HPI discharge line. A penetration containing hard piping is installed for the pump discharge connection. A hose connects the pump outlet to the penetration piping and separately from the penetration piping to the tie-in point. Hard piping is routed such that the hose connection points are readily accessible after a BDBEE.

In every deterministic case, the function for make-up or for high pressure injection (HPI) remains available, and the NSPC are met without VFDR DB-2012. As a result, VFDR DB-2012 has been closed. Therefore, no recovery actions are credited to resolve this concern. Implementation item DB-1983 addresses the modification for the FLEX RCS makeup modification, which installed FLEX RCS makeup pumps (credited by PRA) as an overall plant design modification that is not credited to meet any NSPC. The modification is tracked by DB-1983 within LAR Attachment S (Table S-1). Procedures will be created as part of the design modification process to provide portable 480 volt power supply to the FLEX RCS makeup pumps and perform a manual valve alignment to supply inventory to the RCS as credited in the PRA.

SSA RAI 10

c. Confirm that the RAs associated with the RCS charging pumps, connections, and associated auxiliaries are determined feasible in accordance with FAQ 07-0030, "Establishing Recovery Actions," in every fire compartment where these modifications are credited. Describe the thermal-hydraulic analysis used to establish the feasibility of the RAs, including assumptions.

Response:

As explained in the response to SSA RAI 10.b, VFDR DB-2012 has been closed; the NSPC are met without this VFDR. Therefore, no recovery actions are credited to resolve this concern.

SSA RAI 11

NFPA 805, Section 4.1, requires that once a determination has been made, a fire protection system or feature is required to achieve the performance criteria of Section 1.5, and its design and qualification shall meet the applicable requirements of Chapter 3. In LAR, Table 4-3, the licensee identified that "other" fire protection features are credited in a licensing action and identified Licensing Action 8, "Manhole MH3001 Cable Separation," in LAR, Attachment C, as required to meet the NSPC. However, LAR, Attachment K, does not identify any fire protection system or feature required for the basis of the previous approval of Licensing Action 8.

Clarify whether fire protection systems and features are required in Licensing Action 8 to meet the NFPA 805, Chapter 4, requirements, and verify that it meets the applicable NFPA 805, Chapter 3, requirements.

Response:

No fire protection systems are required under LAR Attachment K, Licensing Action 8.

The Licensing Action 8 exemption from the NRC for manhole MH3001 was for cable separation measuring less than 20 feet while there was no fire suppression and detection present. Specifically, the request from DBNPS letter dated July 31, 1989 is with respect to redundant circuits in the subject manhole associated with the service water system, including pumps (P3-1, P3-2, and P3-3), the backup pump (P-180), valves (SW 1395 and SW 1399), and motor control centers (MCCs E12C and F12C), which are less than six feet from one another.

This manhole has neither active nor passive fire protection features. The NRC's prior approval in letter dated April 18, 1990 evaluated and accepted that the fire compartment did not have any installed redundant electrical cable fire protective separation, nor either a fire suppression or fire detection system. The basis for the approval is due to the cable construction being equivalent to IEEE Standard 383-1974, the cables installed per Reg. Guide 1.75, the cables being protected from a fault condition by overcurrent devices, and the enclosure being made of concrete with the only access through a normally-closed steel manway. The cables continue to meet these cable construction requirements as described in the response to section 3.3.5.3, and are protected by overcurrent devices for fault protection as required by section 2.4.2.2. The limited access and construction materials of the enclosure and manhole also remain the same.

The LAR Table 4-3 "Comment" column is hereby revised by: 1) adding a comment clarifying the "L" under the "Other" column for fire compartment MA-01; and 2) adding the following note: "The enclosure is made of concrete with the only access through a normally-closed steel manway."

SSA RAI 12

The regulations in 10 CFR 50.48(c)(2)(vii) permit licensees to request NRC approval to use performance-based methods for the fire protection program elements and minimum design requirements in NFPA 805, Chapter 3.

Section 2.3.2 of RG 1.205 provides guidance on using previously approved alternatives to meet NFPA 805 requirements. The guidance indicates that licensees can use existing exemptions or deviations to demonstrate compliance with NFPA 805, provided the licensee acceptably addresses the continued validity of any exemption or deviation in effect at the time of the NFPA 805 licensee amendment application.

Licensing Action 2 of LAR, Attachment K, states that an existing exemption to the separation requirements for the component cooling water pumps located in fire compartment T-01 will be transitioned to NFPA 805. The LAR states that transition of this exemption requires NRC approval and is included in LAR, Attachment L, as Approval Request 8. The LAR indicates that the approval requests in Attachment L are being requested in accordance with 10 CFR 50.48(c)(2)(vii). However, Approval Request 8 is associated with a deviation from deterministic separation requirements in NFPA 805, Chapter 4. Thus, Approval Request 8

cannot be approved under 10 CFR 0.48(c)(2)(vii) since this regulation applies only to NFPA 805, Chapter 3, requirements.

Revise the LAR to withdraw Approval Request 8. Describe how the component cooling water pump separation issue in fire compartment T-01 will meet the requirements of NFPA 805, Chapter 4. Alternatively, describe any associated plant modifications that will be used meet the NFPA 805, Chapter 4, requirements, and provide the necessary implementation items for LAR, Attachment S.

Response:

LAR Attachment L, Approval Request 8 is hereby withdrawn. LAR Attachment K, Licensing Action 2, "CCW [component cooling water] Pump Separation," is hereby modified from "To be transitioned? Yes" to "To be transitioned? No."

A variance from the deterministic requirements of NFPA 805 has been developed (VFDR DB-2061) to address the potential for a fire in compartment T-01 to affect all CCW pumps.

A new LAR Attachment S, Table S-1 item DB-2061 will be added. Should FRE of this VFDR warrant an additional modification for acceptable risk, defense-in-depth, and safety margin results, then a modification will be developed and implemented.

A revision to the LAR Attachments K, L, and S will be provided in a future transmittal.

PRA RAI 01 – Internal Events PRA Gap Findings and Facts and Observations

Attachment U, "Internal Events PRA Quality," of the LAR provides information regarding the peer review of the DBNPS internal events PRA. The following questions relate to disposition of the internal events gap assessment findings and facts and observations (F&Os) associated with this peer review.

a) Gap finding for Supporting Requirement HR-A1 – Pre-Initiators

The gap assessment finding associated with supporting requirement HR-A1 states that the licensee's modeling of pre-initiator human failure events (HFEs) was limited to events identified as "potentially important." The gap assessment finding observed that the risk significance of a pre-initiator can be sensitive to the configuration modeled in the PRA. The LAR does not define the quantitative criteria used to screen pre-initiators, and does not explain how risk significant pre-initiators, which were previously screened, will be considered in future updates to the PRA that reflect changes in plant configuration.

i. **Describe the approach of screening potential pre-initiator HFEs based on risk significance. Include a discussion of the criteria used to justify that the excluded events are insignificant to the NFPA 805 application. Alternatively, confirm that these events will be included in the integrated analysis provided in response to PRA RAI 03.**

Response:

The Human Reliability Analysis notebook prior to the peer review did include the option of screening potential pre-initiator HFEs based on risk significance. After re-reviewing the criteria applied, screening based on risk-significance was not performed. The screening criteria applied included:

- Not in PRA model
- No Impact on component success criteria
- Compelling indicator such as annunciator or monitor recognizable by an operator before, during or immediately after restoration back to service
- Component can be actuated or repositioned successfully upon an actuation signal
- Operability test and maintenance or calibration and verified on a periodic checklist (daily or more frequent)
- Operability test after maintenance or calibration and independent verification
- Independent verification and the component is sealed
- System or component is re-aligned with a startup procedure
- Manufacturing defect
- Error caused by instrument drift

Since screening based on risk-significance was not performed, no risk criteria were established for screening. Therefore, there is no impact to the LAR.

ii. **Provide an implementation item for LAR, Table S-2, to update the risk significance determination of pre-initiators as part of the PRA maintenance and upgrade process, or justify why it is not necessary. The justification should explain how DBNPS will ensure that risk-significant pre-initiator HFEs will not be excluded in post-transition updates to the PRA.**

Response:

Screening pre-initiators based on “risk-significance” is not performed. All screening is performed on a qualitative basis. The Human Reliability

Analysis Notebook has been updated to remove the “risk-significance” criteria and in addition states:

> The criteria used is a comprehensive list and includes:
>
> - Not in PRA model
> - No impact on (or not relevant to) PRA Top Event (CDF or LERF)
> - No impact on component success criteria
> - Compelling indications such as an annunciator or monitor recognizable by an operator before, during, or immediately after restoration back to service
> - Component can be actuated or repositioned successfully upon an actuation signal
> - Operability Test after Maintenance or Calibration AND verified on a periodic checklist (daily or more frequent)
> - Operability Test after Maintenance or Calibration AND Independent Verification
> - Independent Verification AND the component is sealed
> - System or Component is re-aligned with a start-up procedure
> - Manufacturing Defect
> - Error Caused by Instrument Drift
> - Resulted from Equipment Damage Due to Material Defect.

In order to ensure screening based on risk-significance in the future does not occur, the following statement has been made in the human reliability analysis (HRA) notebook:

> Any future additions to these criteria [listed above] must be evaluated for potential impact on PRA applications (including but not limited to, the NFPA 805 fire protection program) prior to use of the criterion. Note: Criterion which was related to screening based on risk significance, was never used and has been deleted as an option due to potential adverse impacts on applications.

Current procedures for the PRA maintenance and upgrade process do not specify this level of detail. Instead, the technical adequacy is controlled to the standard within each notebook as they are the record that is maintained.

Screening based on risk-significance is not and will not be used; therefore, a LAR Attachment S, Table S-2 implementation item is not required to update the determination of risk significance.

PRA RAI 01

b) F&O LE-E4-01 – Circular Logic

F&O LE-E4-01 states that it was difficult to determine if circular logic was appropriately modeled, and the LAR does not provide additional details on how circular logic was modeled. Describe how circular logic loops were modeled. Provide the bases (e.g., timing associated with the particular equipment) for breaking the circular logic and justify that the treatment does not exclude important risk contributors, such as dependent failures.

Response:

Athough F&O LE-E4-01 was written during the Level 2 peer review, there are no circular logic loops that are specific to the Level 2 analysis. Internal events peer reviews did not determine this to be an issue as no F&O against circular logic was identified during other reviews. This was a documentation issue only; therefore little detail was included into the short F&O resolution. However, it was included in the modeling guideline notebook.

To answer the larger question on circular logic, the following provides a better discussion as to how circular logic was addressed. Circular logic or a logic loop is defined as the infinite circulation of supporting relations due to their mutual system dependencies in the fault tree analysis. An example is the CCW cooling to the emergency diesel generators (EDGs), and the circular dependence of the CCW pumps on power (from the EDGs following a LOOP). The computer program CAFTA has a tool to identify logical loops that may exist in a fault tree, and loops must be broken prior to quantification. The approach used to break these circular logic loops was to develop special gates for the last level of support, without the circular dependence, which then feed through the systems causing the loop. Conceptually, if system A and system B form a logical loop, a special gate containing all failures of system A except for its system B dependencies was created. The special system A gate then serves as the dependency to system B. System B is then linked into the system A logic that serves the rest of the fault tree. In this way, all dependencies are maintained and no risk contributors are lost in the process. All logic loops were broken in this manner. For further clarity, some examples from the model are described below:

For the EDG cooling logical loop, a special EDG gate was developed that does not include CCW cooling dependencies but does include all other EDG failure modes. This gate supports a special gate that represents the loss of its associated essential bus (C1_41). The special essential bus gate is then used for the power dependencies for service water (SW) and CCW. The SW and CCW trees then provide the cooling support for the EDG trees that support the power dependencies for all trees other than SW and CCW. Timing is not used as the basis for this process. Provided that offsite power is not available, any failure of an EDG would fail its associated CCW and SW trains and their loads, and any failure of CCW and SW equipment that supports an EDG will fail the

associated EDG and its loads. Figure 1 illustrates how the EDG logical loops were broken.

Logical loops for station and instrument air (IA) and turbine plant cooling water (TPCW) were broken in the same manner, and can help illustrate the circular logic breaking methodology, even though both systems are considered failed in the fire PRA model. Special logic for TPCW that does not include SW or electrical power (EP) dependencies was developed to support logic for IA that also does not include SW or electrical power dependencies. The special IA logic is then used to support IA dependencies in the TPCW system. Also, the special IA logic supports SW valves that, when failed, could cause a loss of SW or SW's support of TPCW. This logic then supports the normal TPCW logic, which supports the normal IA logic that serves as dependencies for other plant loads. In addition, the IA logic that does not contain SW or electrical power dependencies also supports the auxiliary building non-radiological ventilation, which supports ventilation to the low voltage switchgear rooms. The low voltage switchgear ventilation logic supports power to the EDG ventilation as well as cooling for the startup transformers. The EDG ventilation logic supports successful EDG operation to supply power to SW, which then supports TPCW. The startup transformer cooling logic supports logic for power to the A and B busses, which ultimately supports the logic for the normal TPCW and IA logic. The IA support system initiating event logic also uses the TPCW logic that excludes SW and electrical power, as well as power logic that excludes failures of the startup transformers. This is because, for an IA initiating event, there would be no plant trip, or need to transfer to the startup transformers until after all IA is lost. Figure 2 illustrates the IA and TPCW logical loop break method.

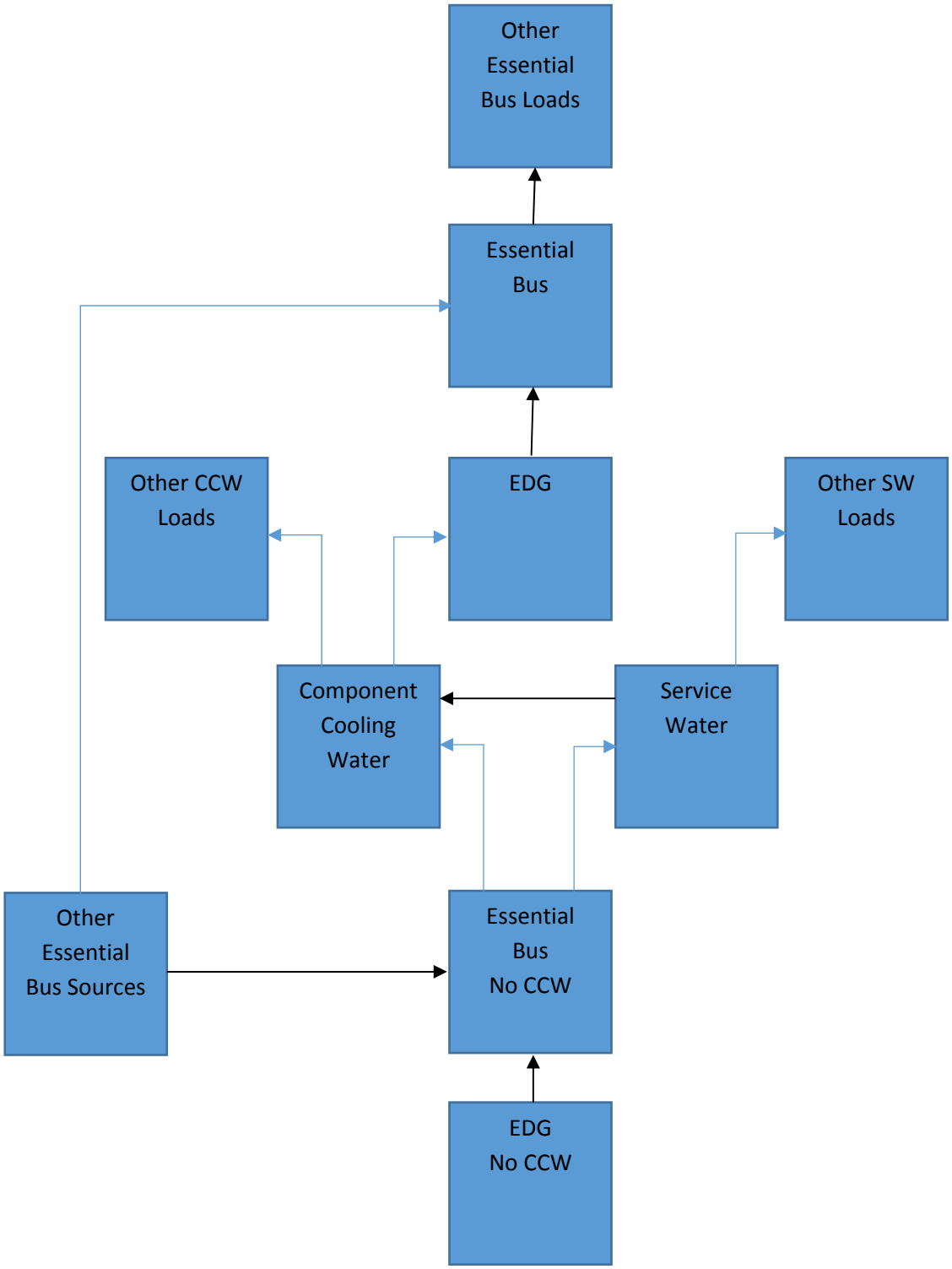

Figure 1. Emergency Diesel Generator Circular Logic Breaks

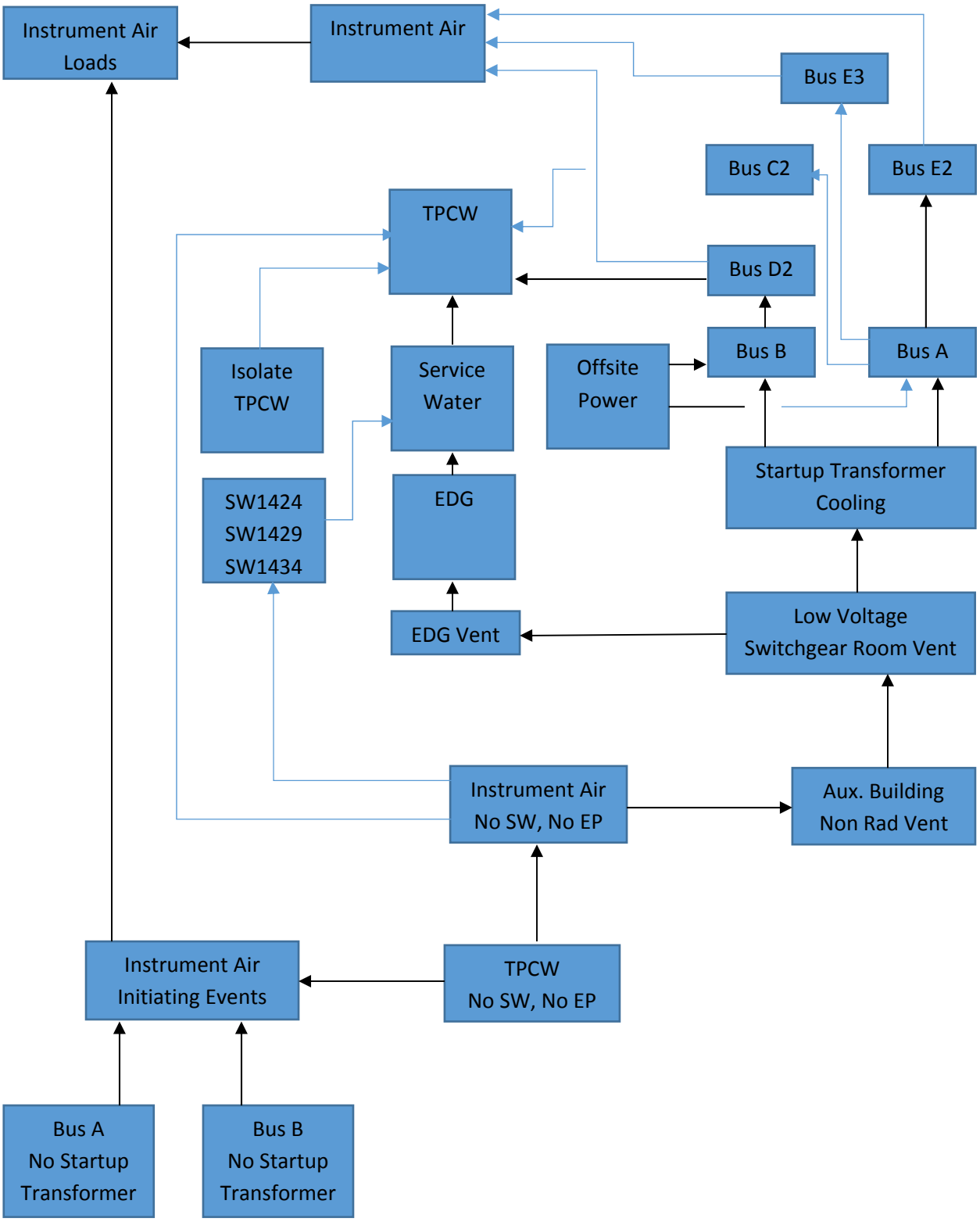

Figure 2. Turbine Plant Cooling Water and Instrument Air Circular Logic Breaks

PRA RAI 04 – Internal Events PRA Peer Review

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. RG 1.200, Revision 2 "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (ADAMS Accession No. ML090410014), describes a peer review process using an associated ASME/ANS standard (currently ASME/ANS-RA-Sa-2009) as one acceptable approach for determining the technical adequacy of the PRA once acceptable consensus approaches or models have been established.

Attachment U of the LAR explains that an assessment was performed in April 2008 by a contractor on the internal events PRA against supporting requirements in PRA Standard ASME/ANS RA-Sb-2005, "Addenda to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," dated December 30, 2005, as qualified by RG 1.200, Revision 1. Differences exist between the supporting requirements in PRA Standard ASME/ANS RA-Sb-2005, as qualified by RG 1.200, Revision 1, and the supporting requirements in ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2. In accordance with RG 1.200, Revision 2, it is expected that the differences between the current version of the PRA standard and the earlier version of the standard used in the internal events PRA assessment be identified and addressed (i.e., a gap assessment be performed). While LAR, Table U-1, provides a cross reference between the two versions of the standard, the LAR does not address the additional changes between the standards that would require re-evaluation of the PRA against the current ASME/ANS PRA standard. Provide a gap assessment of the internal events PRA against ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2.

Response:

A comparison was made between the ASME/ANS RA-Sb-2005 standard and the ASME/ANS RA-Sa-2009 standard. For the majority of the supporting requirements the wording was unchanged, or the intent remained the same. There were no changes to the supporting requirements that were found to have an impact on the internal events PRA model. As discussed in response to PRA RAI 05, the large early release frequency (LE) element and the internal flooding elements were reviewed against the ASME/ANS RA-Sa-2009 standard, following NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," guidance. Therefore, the elements for LE and internal flooding were not required. However, they are included for completeness.

The reviews and subsequent findings or suggestions were identified in order to meet Capability Category II (CCII) for all supporting requirements (SRs). By addressing the findings or suggestions all internal event and internal flooding SRs meet CCII or better.

See below for a SR-by-SR and high-level requirement (HLR)-by-HLR comparison.

Keyword in "Change From 05 to 09" Column	Description of Keyword
Exact	no change, not even to wording
Same	same intent with slightly different wording (ex. Writing out acronyms, rearranging a sentence…)
Change	Change
N/A	SR deleted
New	new to standard

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
Section 2 (internal events)				
HLR-IE-A	HLR-IE-A	Exact		
HLR-IE-B	HLR-IE-B	Exact		
HLR-IE-C	HLR-IE-C	Exact		
HLR-IE-D	HLR-IE-D	Same	Wording	Change in the order of the wording, no change to intent
IE-A1	IE-A1	Exact		
IE-A2	IE-A2	Change	Removes internal flooding initiators	Meet CC I/II/III assessed in Internal Flooding (IF)
IE-A3	IE-A3	Exact		
IE-A3a	IE-A4	Exact		
IE-A4	IE-A5	Exact		
IE-A4a	IE-A6	Exact		
IE-A5	IE-A7	Exact		
IE-A6	IE-A8	Exact		
IE-A7	IE-A9	Exact		
IE-A8-deleted	n/a	n/a		
IE-A9-deleted	n/a	n/a		
IE-A10	IE-A10	Exact		
IE-B1	IE-B1	Same	Changed reference from Paragraph 4.5.2 to 2-2.2 and from 4.5.8 to 2-2.7	Text of the referenced sections is essentially the same
IE-B2	IE-B2	Exact		

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
IE-B3	IE-B3	Same	CC II avoid subsuming event into a group unless is changed to DO NOT SUBSUME scenarios into a group	Intent of CC II unchanged. Initiating events are not subsumed unless the impacts are comparable or less than those of the remaining events in that group
IE-B4	IE-B4	Exact		
IE-B5	IE-B5	Exact		
IE-C1	IE-C1	Exact		
IE-C1a	IE-C2	Exact		
IE-C1b	IE-C3	Same	Reference to other supporting requirements changed in numbers	Text of the referenced SRs is essentially the Same
IE-C2	IE-C4	Same	Note 2 changed to Reference 2-2	No change to intent of SR
IE-C3	IE-C5	Exact		
IE-C4	IE-C6	Same	Reference to Paragraphs 4.5.6 changed to Section 2-2.7, reference to Paragraph 4.5.8 changed to Section 2-2.7	Text of the referenced Sections is essentially the Same
IE-C5	IE-C7	Same	Added an acceptable method (NUREG/CR-6928)	No change to intent of SR
IE-C6	IE-C8	Same	Changed reference from Paragraph 4.5.4 to Section 2-2.4, and Paragraph 4.5.6 to Section 2-2.6	Text of the referenced Sections is essentially the Same
IE-C7	IE-C9	Same	Changed reference from Paragraph 4.5.4 to Section 2-2.4	Text of the referenced Sections is essentially the Same
IE-C8	IE-C10	Exact		

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
IE-C9	IE-C11	Same	Changed reference from Paragraph 4.5.5 to Section 2-2.5	Text of the referenced Sections is essentially the Same
IE-C10	IE-C12	Exact		
IE-C11	IE-C13	Same	Changed "Use generic data and INCLUDE plant-specific functions" to "Use generic data and INCLUDE plant-specific features to decide which generic data are most applicable"	IE frequencies, including LOCA frequencies, based on generic-PWR numbers from NRC Parameter Estimate 2007. IE frequencies were compared with Three Mile Island and Arkansas Nuclear One – Unit 1 to verify reasonableness.
IE-C12	IE-C14	Exact		
IE-C13	IE-C15	Exact		
IE-D1	IE-D1	Exact		
IE-D2	IE-D2	Exact		
IE-D3	IE-D3	Same	Added reference to QU-E1 & QU-E2	Added reference to SRs on uncertainty and assumptions, no change to intent
HLR-AS-A	HLR-AS-A	Exact		
HLR-AS-B	HLR-AS-B	Exact		
HLR-AS-C	HLR-AS-C	Same	Wording	Change in the order of the wording, no change to intent
AS-A1	AS-A1	Exact		
AS-A2	AS-A2	Exact		
AS-A3	AS-A3	Same	Reference to another SR changed from SC-A4 to SC-A3 (this is the same SR)	Text of the referenced SR is essentially the same
AS-A4	AS-A4	Same	Reference to another SR changed from SC-A4 to SC-A3 (this is the same SR)	Text of the referenced SR is essentially the same
AS-A5	AS-A5	Exact		
AS-A6	AS-A6	Exact		

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
AS-A7	AS-A7	Exact		
AS-A8	AS-A8	Exact		
AS-A9	AS-A9	Exact		
AS-A10	AS-A10	Exact		
AS-A11	AS-A11	Exact		
AS-B1	AS-B1	Exact		
AS-B2	AS-B2	Exact		
AS-B3	AS-B3	Exact		
AS-B4	AS-B4	Exact		
AS-B5	AS-B5	Exact		
AS-B5a	AS-B6	Exact		
AS-B6	AS-B7	Exact		
AS-C1	AS-C1	Exact		
AS-C2	AS-C2	Exact		
AS-C3	AS-C3	Same	Added reference to QU-E1 & QU-E2	Added reference to SRs on uncertainty and assumptions, no change to intent
HLR-SC-A	HLR-SC-A	Exact		
HLR-SC-B	HLR-SC-B	Exact		
HLR-SC-C	HLR-SC-C	Same	Wording	Change in the order of the wording, no change to intent
SC-A1	SC-A1	Exact		
SC-A2	SC-A2	Exact		
SC-A3-deleted	n/a	n/a		
SC-A4	SC-A3	Exact		
SC-A4a	SC-A4	Exact		
SC-A5	SC-A5	Exact		
SC-A6	SC-A6	Exact		
SC-B1	SC-B1	Exact		
SC-B2	SC-B2	Exact		
SC-B3	SC-B3	Exact		
SC-B4	SC-B4	Exact		
SC-B5	SC-B5	Exact		
SC-C1	SC-C1	Exact		
SC-C2	SC-C2	Exact		

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
SC-C3	SC-C3	Same	Added reference to QU-E1 & QU-E2	Added reference to SRs on uncertainty and assumptions, no change to intent
HLR-SY-A	HLR-SY-A	Exact		
HLR-SY-B	HLR-SY-B	Exact		
HLR-SY-C	HLR-SY-C	Same	Wording	Change in the order of the wording, no change to intent
SY-A1	SY-A1	Exact		
SY-A2	SY-A2	Exact		
SY-A3	SY-A3	Exact		
SY-A4	SY-A4	Same	Added knowledgeable (before plant personnel)	Interviews were performed with the responsible system engineer
SY-A5	SY-A5	Exact		
SY-A6	SY-A6	Exact		
SY-A7	SY-A7	Exact		
SY-A8	SY-A8	Exact		
SY-A9- deleted	n/a	n/a		
SY-A10	SY-A9	Exact		
SY-A11	SY-A10	Exact		
SY-A12	SY-A11	Same	Reference to another supporting requirement changed SY-A14 changed to SY-A15	Text of the referenced SR is the same
SY-A12a	SY-A12	Exact		
SY-A12b	SY-A13	Exact		
SY-A13	SY-A14	Same	Reference to another supporting requirement changed SY-A12 changed to SY-A11	Text of the referenced SR is the same
SY-A14	SY-A15	Exact		
SY-A15	SY-A16	Same	References paragraph 4.5.5 changed to 2-2.5	References overall HR Section
SY-A16	SY-A17	Same	References paragraph 4.5.2 changed to 2-2.2	References overall HR section

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
SY-A17	SY-A18	Same	SY-A20 changed to SY-A22	Text of the referenced SR is the Same
SY-A18	SY-A19	Same	Wording	
SY-A18a	SY-A20	Same	DA-C13 changed to DA-C14	Text of the referenced SR is the Same
SY-A19	SY-A21	Exact		
SY-A20	SY-A22	Exact		
SY-A21	SY-A23	Exact		
SY-A22	SY-A24	Same	DA-C14 changed to DA-C15	Text of the referenced SR is the Same
SY-B1	SY-B1	Same	Note (1) changed to reference [2-4]	Referenced NUREG is the same
SY-B2	SY-B2	Exact		
SY-B3	SY-B3	Exact		
SY-B4	SY-B4	Exact		
SY-B5	SY-B5	Exact		
SY-B6	SY-B6	Exact		
SY-B7	SY-B7	Exact		
SY-B8	SY-B8	Exact		
SY-B9-deleted	n/a	n/a		
SY-B10	SY-B9	Exact		
SY-B11	SY-B10	Exact		
SY-B12	SY-B11	Exact		
SY-B13	SY-B12	Exact		
SY-B14	SY-B13	Exact		
SY-B15	SY-B14	Exact		
SY-B16	SY-B15	Exact		
SY-C1	SY-C1	Exact		
SY-C2	SY-C2	Exact		
SY-C3	SY-C3	Same	Wording, added reference to QU-E1 & QU-E2	Added reference to SRs on uncertainty and assumptions, no change to intent
HLR-HR-A	HLR-HR-A	Exact		
HLR-HR-B	HLR-HR-B	Exact		
HLR-HR-C	HLR-HR-C	Exact		
HLR-HR-D	HLR-HR-D	Exact		
HLR-HR-E	HLR-HR-E	Exact		
HLR-HR-F	HLR-HR-F	Exact		
HLR-HR-G	HLR-HR-G	Exact		

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
HLR-HR-H	HLR-HR-H	Exact		
HLR-HR-I	HLR-HR-I	Same	Wording	Change in the order of the wording, no change to intent
HR-A1	HR-A1	Exact		
HR-A2	HR-A2	Exact		
HR-A3	HR-A3	Exact		
HR-B1	HR-B1	Exact		
HR-B2	HR-B2	Exact		
HR-C1	HR-C1	Exact		
HR-C2	HR-C2	Exact		
HR-C3	HR-C3	Exact		
HR-D1	HR-D1	Same	Note 1 and note 2 changed to references 2-5 and 2-6	Referenced NUREGs are the same
HR-D2	HR-D2	Exact		
HR-D3	HR-D3	Exact		
HR-D4	HR-D4	Exact		
HR-D5	HR-D5	Exact		
HR-D6	HR-D6	Exact		
HR-D7	HR-D7	Exact		
HR-E1	HR-E1	Exact		
HR-E2	HR-E2	Exact		
HR-E3	HR-E3	Exact		
HR-E4	HR-E4	Exact		
HR-F1	HR-F1	Exact		
HR-F2	HR-F2	Exact		
HR-G1	HR-G1	Exact		
HR-G2	HR-G2	Exact		
HR-G3	HR-G3	Exact		
HR-G4	HR-G4	Exact		
HR-G5	HR-G5	Exact		
HR-G6	HR-G6	Exact		
HR-G7	HR-G7	Exact		
HR-G8-deleted	n/a	n/a		
HR-G9	HR-G8	Exact		
HR-H1	HR-H1	Exact		

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
HR-H2	HR-H2	Same	Wording, added "if the following occur"	The intent and all criteria in this SR is the Same
HR-H3	HR-H3	Exact		
HR-I1	HR-I1	Exact		
HR-I2	HR-I2	Exact		
HR-I3	HR-I3	Same	Wording, added reference to QU-E1 & QU-E2	Added reference to SRs on uncertainty and assumptions, no change to intent
HLR-DA-A	HLR-DA-A	Exact		
HLR-DA-B	HLR-DA-B	Exact		
HLR-DA-C	HLR-DA-C	Same	Wording	Change in the order of the wording, no change to intent
HLR-DA-D	HLR-DA-D	Exact		
HLR-DA-E	HLR-DA-E	Same	Wording	Change in the order of the wording, no change to intent
DA-A1	DA-A1	Exact		
DA-A1a	DA-A2	Same	Wording, changed references to other SRs	text of the referenced SRs is essentially the same
DA-A2	DA-A3	Exact		
DA-A3	DA-A4	Exact		
DA-B1	DA-B1	Exact		
DA-B2	DA-B2	Exact		
DA-C1	DA-C1	Same	DA-A3 is now DA-A4, moved references from Notes to References list, added NUREG-1715 and NUREG/CR-6928 to part (a), added line to say "see NUREG/CR-6823 [2-1] for a listing of additional data sources"	Text of the referenced SRs is essentially the Same, NUREG/CR-6928 is the primary source for generic probability failures in the DB PRA

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
DA-C2	DA-C2	Same	DA-A2 and DA-A3 change to DA-A3 & DA-A4	Text of the referenced SRs is essentially the Same
DA-C3	DA-C3	Exact		
DA-C4	DA-C4	Exact		
DA-C5	DA-C5	Exact		
DA-C6	DA-C6	Exact		
DA-C7	DA-C7	Exact		
DA-C8	DA-C8	Exact		
DA-C9	DA-C9	Exact		
DA-C10	DA-C10	Exact		
DA-C11	DA-C11	Exact		
DA-C11a	DA-C12	Exact		
DA-C12	DA-C13	Same	Added “knowledgeable” (before plant personnel)	This SR states that interviews should be conducted in the case that "reliable estimates or the start and finish times are not available." For DB reliable start and finish times for all significant maintenance activities are available through the history of the Online Risk Monitor, the Plant Narrative Logs, Potential Conditions Adverse to Quality Reports, and Condition Reports. Therefore no specific interviews have been conducted.
DA-C13	DA-C14	Same	Wording, added phrase "that is a result of a planned, repetitive activity", added example of intersystem unavailability	Intent of SR unchanged
DA-C14	DA-C15	Exact		
DA-C15	DA-C16	Exact		
DA-D1	DA-D1	Exact		
DA-D2	DA-D2	Exact		

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
DA-D3	DA-D3	Exact		
DA-D4	DA-D4	Exact		
DA-D5	DA-D5	Exact		
DA-D6	DA-D6	Same	Wording, for CC II added the phrase "in a manner" prior to "consistent with the component boundaries"	Intent of SR unchanged
DA-D6a	DA-D7	Exact		
DA-D7	DA-D8	Exact		
DA-E1	DA-E1	Exact		
DA-E2	DA-E2	Exact		
DA-E3	DA-E3	Same	Added reference to QU-E1 & QU-E2	Added reference to SRs on uncertainty and assumptions, no change to intent
HLR-QU-A	HLR-QU-A	Exact		
HLR-QU-B	HLR-QU-B	Exact		
HLR-QU-C	HLR-QU-C	Exact		
HLR-QU-D	HLR-QU-D	Change	Added LERF	LERF added to the model following 2008 Self-Assessment and assessed via Focused Scope Peer Review
HLR-QU-E	HLR-QU-E	Exact		
HLR-QU-F	HLR-QU-F	Same	Wording	Change in the order of the wording, no change to intent
QU-A1	QU-A1	Exact		
QU-A2a	QU-A2	Exact		
QU-A2b	QU-A3	Exact		
QU-A3	QU-A4	Exact		
QU-A4	QU-A5	Exact		
QU-B1	QU-B1	Exact		
QU-B2	QU-B2	Exact		
QU-B3	QU-B3	Exact		
QU-B4	QU-B4	Exact		

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
QU-B5	QU-B5	Change	AVOID changed to DO NOT; reference changed from Note to References	Circular logic broken using the guidance from NUREG/CR-2728, with care to ensure no unnecessary conservatisms or nonconservatisms were introduced into the model logic
QU-B6	QU-B6	Exact		
QU-B7a	QU-B7	Exact		
QU-B7b	QU-B8	Exact		
QU-B8	QU-B9	Exact		
QU-B9	QU-B10	Exact		
QU-C1	QU-C1	Exact		
QU-C2	QU-C2	Exact		
QU-C3	QU-C3	Exact		
QU-D1a	QU-D1	Exact		
QU-D1b	QU-D2	Exact		
QU-D1c	QU-D3	Exact		
QU-D2 -deleted	n/a	n/a		
QU-D3	QU-D4	Exact		
QU-D4	QU-D5	Exact		
QU-D5a	QU-D6	Exact		
QU-D5b	QU-D7	Exact		
QU-E1	QU-E1	Same	Removed "key" from key sources	Efforts made to capture all sources of uncertainty, and not just key sources, utilizing NUREG 1855 and EPRI 1016737.
QU-E2	QU-E2	Same	Removed "key" from key sources	All assumptions are documented in the respective notebooks
QU-E3	QU-E3	Same	HR-G9 is now HR-G8, IE-C13 is now IE-C15	Text of the referenced SRs is essentially the Same

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
QU-E4	QU-E4	Change	Changed from three distinct capability categories to a single CC I/II/III	The DB Quantification Notebook identifies assumptions and sources of uncertainty, and assess how they impact the overall model and results
QU-F1	QU-F1	Exact		
QU-F2	QU-F2	Exact		
QU-F3	QU-F3	Exact		
QU-F4	QU-F4	Same	Wording (took out the examples "such as..)	The intent to document the model assumptions and sources of uncertainty is unchanged. The DB documentation includes discussions on these items.
QU-F5	QU-F5	Exact		
QU-F6	QU-F6	Exact		
HLR-LE-A	HLR-LE-A	Exact		See Note 1
HLR-LE-B	HLR-LE-B	Exact		See Note 1
HLR-LE-C	HLR-LE-C	Exact		See Note 1
HLR-LE-D	HLR-LE-D	Exact		See Note 1
HLR-LE-E	HLR-LE-E	Exact		See Note 1
HLR-LE-F	HLR-LE-F	Same	Removed "key" from key sources	See Note 1
HLR-LE-G	HLR-LE-G	Same	Wording	See Note 1
LE-A1	LE-A1	Exact		See Note 1
LE-A2	LE-A2	Exact		See Note 1
LE-A3	LE-A3	Exact		See Note 1
LE-A4	LE-A4	Exact		See Note 1
LE-A5	LE-A5	Exact		See Note 1
LE-B1	LE-B1	Exact		See Note 1
LE-B2	LE-B2	Exact		See Note 1
LE-B3	LE-B3	Exact		See Note 1
LE-C1	LE-C1	Same	Wording	See Note 1
LE-C2a	LE-C2	Exact		See Note 1
LE-C2b	LE-C3	Exact		See Note 1
LE-C3	LE-C4	Exact		See Note 1
LE-C4	LE-C5	Exact		See Note 1
LE-C5	LE-C6	Exact		See Note 1

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
LE-C6	LE-C7	Exact		See Note 1
LE-C7	LE-C8	Exact		See Note 1
LE-C8a	LE-C9	Exact		See Note 1
LE-C8b	LE-C10	Exact		See Note 1
LE-C9a	LE-C11	Exact		See Note 1
LE-C9b	LE-C12	Same	LE-C9a changed to LE-C11	See Note 1
LE-C10	LE-C13	Exact		See Note 1
LE-D1a	LE-D1	Exact		See Note 1
LE-D1b	LE-D2	Exact		See Note 1
LE-D2	LE-D3	Exact		See Note 1
LE-D3	LE-D4	Exact		See Note 1
LE-D4	LE-D5	Exact		See Note 1
LE-D5	LE-D6	Exact		See Note 1
LE-D6	LE-D7	Exact		See Note 1
LE-E1	LE-E1	Exact		See Note 1
LE-E2	LE-E2	Exact		See Note 1
LE-E3	LE-E3	Exact		See Note 1
LE-E4	LE-E4	Same	Table numbers changed	See Note 1
LE-F1a	LE-F1	Same	Table numbers changed	See Note 1
LE-F1b	LE-F2	Exact		See Note 1
LE-F2	n/a	n/a	Deleted in 09	See Note 1
LE-F3	LE-F3	Same	Added characterize to SR previously just had Identify	See Note 1
LE-G1	LE-G1	Exact		See Note 1
LE-G2	LE-G2	Exact		See Note 1
LE-G3	LE-G3	Exact		See Note 1
LE-G4	LE-G4	Same	Wording	See Note 1
LE-G5	LE-G5	Exact		See Note 1
LE-G6	LE-G6	Exact		See Note 1
Section 3 (Internal Flooding)				
HLR-IF-A	HLR-IFPP-A	Exact		See Note 2
n/a	HLR-IFPP-B	n/a	Documentation	See Note 2
IF-A1	IFPP-A1	Exact		See Note 2
IF-A1a	IFPP-A2	Exact		See Note 2

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
IF-A1b	IFPP-A3	Exact		See Note 2
IF-A3	IFPP-A4	Exact		See Note 2
IF-A4	IFPP-A5	Exact		See Note 2
n/a	IFPP-B1	n/a	Documentation SR added (flooding split into sections)	See Note 2
n/a	IFPP-B2	n/a	Documentation SR added (flooding split into sections)	See Note 2
n/a	IFPP-B3	n/a	documentation SR added (flooding split into sections)	See Note 2
HLR-IF-B	HLR-IFSO-A	Exact		See Note 2
n/a	HLR-IFSO-B	n/a		See Note 2
IF-B1	IFSO-A1	Exact		See Note 2
IF-B1a	IFSO-A2	Exact		See Note 2
IF-B1b	IFSO-A3	Same	IF-B1 & IF-B1a changed to IFSO-A1 & IFSO-A2	See Note 2
IF-B2	IFSO-A4	Exact		See Note 2
IF-B3	IFSO-A5	Exact		See Note 2
IF-B3a	IFSO-A6	Same	Removed note	See Note 2
n/a	IFSO-B1	n/a	Documentation SR added (flooding split into sections)	See Note 2
n/a	IFSO-B2	n/a	Documentation SR added (flooding split into sections)	See Note 2
n/a	IFSO-B3	n/a	Documentation SR added (flooding split into sections)	See Note 2
HLR-IF-C	HLR-IFSN-A	Same	Wording	See Note 2
n/a	HLR-IFSN-B	n/a		See Note 2
IF-C1	IFSN-A1	Exact		See Note 2
IF-C2	IFSN-A2	Exact		See Note 2
IF-C2a	IFSN-A3	Exact		See Note 2
IF-C2b	IFSN-A4	Exact		See Note 2
IF-C2c	IFSN-A5	Same	SR numbers change	See Note 2
IF-C3	IFSN-A6	Exact		See Note 2
IF-C3a	IFSN-A7	Exact		See Note 2
IF-C3b	IFSN-A8	Same	IF-A1a changed to IFPP-A2	See Note 2

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
IF-C3c	IFSN-A9	Exact		See Note 2
IF-C4	IFSN-A10	Exact		See Note 2
IF-C4a	IFSN-A11	Exact		See Note 2
IF-C5	IFSN-A12	Exact		See Note 2
IF-C5a	IFSN-A13	Exact		See Note 2
IF-C6	IFSN-A14	Exact		See Note 2
IF-C7	IFSN-A15	Exact		See Note 2
IF-C8	IFSN-A16	Exact		See Note 2
IF-C9	IFSN-A17	Exact		See Note 2
n/a	IFSN-B1	n/a	Documentation SR added (flooding split into sections)	See Note 2
n/a	IFSN-B2	n/a	Documentation SR added (flooding split into sections)	See Note 2
n/a	IFSN-B3	n/a	Documentation SR added (flooding split into sections)	See Note 2
HLR-IF-D	HLR-IFEV-A	Exact		See Note 2
n/a	HLR-IFEV-B	n/a		See Note 2
IF-D1	IFEV-A1	Exact		See Note 2
IF-D2	n/a	n/a		See Note 2
IF-D3	IFEV-A2	Exact		See Note 2
IF-D3a	IFEV-A3	Exact		See Note 2
IF-D4	IFEV-A4	Exact		See Note 2
IF-D5	IFEV-A5	Exact		See Note 2
IF-D5a	IFEV-A6	Exact		See Note 2
IF-D6	IFEV-A7	Exact		See Note 2
IF-D7	IFEV-A8	Exact		See Note 2
n/a	IFEV-B1	n/a	Documentation SR added (flooding split into sections)	See Note 2
n/a	IFEV-B2	n/a	Documentation SR added (flooding split into sections)	See Note 2
n/a	IFEV-B3	n/a	Documentation SR added (flooding split into sections)	See Note 2
HLR-IF-E	HLR-IFQU-A	Exact		See Note 2
n/a	HLR-IFQU-B	n/a		See Note 2

ASME/ANS RA-Sb-2005 (RG 1.200 Rev 1)	ASME/ANS RA-Sa-2009 (RG 1.200 Rev 2)	Change From 05 to 09	Description of Change	Comments
HLR-IF-F	n/a	n/a	Due to Internal flooding being split into section this is essentially the second HLR in each section	See Note 2
IF-E1	IFQU-A1	Exact		See Note 2
IF-E2	n/a	n/a		See Note 2
IF-E3	IFQU-A2	Exact		See Note 2
IF-E3a	IFQU-A3	Exact		See Note 2
IF-E4	IFQU-A4	Exact		See Note 2
IF-E5	IFQU-A5	Exact		See Note 2
IF-E5a	IFQU-A6	Exact		See Note 2
IF-E6	IFQU-A7	Exact		See Note 2
IF-E6a	IFQU-A8	Exact		See Note 2
IF-E6b	IFQU-A9	Exact		See Note 2
IF-E7	IFQU-A10	Exact		See Note 2
IF-E8	IFQU-A11	Exact		See Note 2
IF-F1	n/a	n/a	Documentation SR added (flooding split into sections)	See Note 2
IF-F2	n/a	n/a	Documentation SR added (flooding split into sections)	See Note 2
IF-F3	n/a	n/a	Documentation SR added (flooding split into sections)	See Note 2
n/a	IFQU-B1	n/a	Documentation SR added (flooding split into sections)	See Note 2
n/a	IFQU-B2	n/a	Documentation SR added (flooding split into sections)	See Note 2
n/a	IFQU-B3	n/a	Documentation SR added (flooding split into sections)	See Note 2

Notes:

1. The focused scope peer review on LE was to ASME/ANS RA-Sa-2009
2. The focused scope peer review on IF was to ASME/ANS RA-Sa-2009

PRA RAI 05 – Internal Events PRA Focused-Scope Peer Review

Attachment U of the LAR explains that focused-scope peer reviews were performed by a contractor on the LERF element of the internal events PRA in October 2011, and on the internal flooding PRA in July 2012, using the current PRA standard. The licensee's March 7, 2016, supplement to the LAR states that the focused-scope peer reviews were performed using ASME/ANS RA-Sa-2009, but does not indicate that these reviews were consistent with RG 1.200, Revision 2, or met industry and NRC criteria (e.g., NEI 05-04 guidance). Clarify whether the focused-scope peer reviews were performed using PRA standard ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2, and that the peer review process met criteria for a focused-scope peer review.

Response:

The internal flooding peer review was performed using part three of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard (ASME/ANS RA-Sa-2009) and any clarifications and qualifications provided in the NRC endorsement of the standard contained in RG 1.200, Revision 2. The peer review was performed using the process defined in NEI 05-04.

The focused-scope peer review of the LE (Large Early Release Frequency) element of the PRA followed the applicable portions of the general peer review of NEI 05-04 with the exception that the peer review was conducted remotely with a kick-off meeting being conducted in a web-meeting format. The specific high level requirements and supporting requirements were reviewed to ASME/ANS-RA-Sa-2009 standard. The only NRC clarification from RG 1.200 Rev. 2 for the LE portion of the standard consisted of eliminating the need to perform sensitivity studies in the uncertainty assessment. This clarification would not impact the results of the review, thus the focused-scope peer review for LE was performed using the PRA standard ASME/ANS-RA-Sa-2009 as qualified by RG 1.200, Revision 2.

In summary, the peer review of the internal flooding PRA model and the focused-scope peer review of the LE element followed the NEI 05-04 process reviewing the applicable portions of the ASME/ANS-RA-SA-2009 standard consistent with RG 1.200, Revision 2.

PRA RAI 06 – Minimum Joint Human Error Probability

NUREG-1921 discusses the need to consider a minimum value for the joint probability of multiple HFEs in human reliability analyses. NUREG-1921 refers to Table 2-1 of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," dated April 2005 (ADAMS Accession No. ML051160213), which recommends that joint human error probability (HEP) values should not be below 1E-5. Table 4-4 of EPRI 1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," provides a lower limiting value of 1E-6 for sequences with a very low level of dependence. Therefore, the guidance in

NUREG-1921 allows for assigning joint HEPs that are less than 1E-5, but only through assigning proper levels of dependency.

Confirm that each joint HEP value used in the fire PRA below 1E-5 includes its own justification that demonstrates the inapplicability of the NUREG-1792 lower value guideline (i.e., that the criteria for independent HFEs are met). Provide an estimate of the number of these joint HEP values below 1.0E-5, discuss the range of values, and provide at least two different examples where this justification is applied. Alternatively, confirm that joint HEPs of 1E-05 will be used in the integrated analysis provided in response to PRA RAI 03.

Response:

The lower value guidance in NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines – Final Report," was not followed during development of the fire PRA model to support the LAR. However, a sensitivity study has been performed, and using the lower limiting value of 1E-05 will increase overall CDF approximately 1.3 percent, and increase overall LERF approximately 0.5 percent. Therefore, rather than justifying the use of combined joint HEP values less than 1E-05, a lower limit joint HEP value of 1E-05 will be used in response to PRA RAI 03, and LAR Table W-3 will be updated accordingly.

PRA RAI 07 – Assumed Cable Routing

RG 1.174 includes consideration of change in risk as part of the acceptance guidelines for risk-informed applications. The cable selection analysis, F&Os ES-A3-01, and PRM-B10-01, refer to a UNL logic list, and in the latter case, states that "equipment not cable traced were assumed to be failed using the UNL tag." F&O FSS-E4-01 refers to cables that are failed due to assumed cable routing. The assumption that untraced cables are failed is a conservative approach for modeling untraced cables for the post-transition plant model, but can lead to underestimation of the change in risk when used in the compliant plant model.

a) Describe the extent of untraced fire PRA cables and how they were treated in the fire PRA. Explain how they were treated in the compliant and post-transition plant models.

Response:

The analysis identified four systems not credited due to lack of cable routing information. These systems include instrument air, main feedwater, turbine plant cooling water, and circulating water. These systems were not credited in the fire PRA because of low fire risk importance. All untraced systems are assumed to be in the worst failure mode for the fire PRA. The unlocated (UNL) equipment was failed for both the compliant and post-transition plant models.

In addition to the four systems, FHAR Appendix C-3 identified a number of cabinets as having inadequate breaker/fuse coordination. Per the FHAR, the associated cables were not specifically identified or located, but they were known to remain within the same fire area as the cabinet itself. The analysis began with the Appendix R fire barriers and partitioned them into smaller groups. For example, Appendix R fire area K was broken down into NFPA 805 fire compartments K-01 and K-02. Therefore, for the cabinets listed in FHAR Appendix C-3, cabinet failures were assumed for any fire that damaged additional equipment. For fires in those compartments containing non-coordinated load cables from the cabinet, the cabinet is assumed to be failed. This includes failure of cabinet C3615 in K-01 and K-02, C3616 in J-01 and J-02, C3630 in R-01, and cabinets C6708, C6709, C6714, and C6715 failed in fire compartments DD-01, FF-01, FF-02, FF-03, and HH-01.

PRA RAI 07

b) Explain how assumptions made about untraced cables do not significantly contribute to the underestimation of the transition change in risk. Alternatively, confirm that an acceptable approach will be used in the integrated analysis requested in PRA RAI 03.

Response:

The assumption of failure of the equipment identified in response to PRA RAI 07(a) may overestimate compliant plant risk, and lead to an underestimation of transition change in risk. This is because a compliant plant may not actually fail all of this equipment in every scenario. It is therefore possible that the approach used in the LAR underestimates transition change in risk. However, completely removing the failures of the equipment identified in response to PRA RAI 07(a) would likely underestimate compliant plant risk, and overestimate transition change in risk, as it is likely that some of the equipment would still fail due to fire impacts.

Therefore, in response to PRA RAI 03, the results for both methods (keeping the assumptions and removing the assumptions) will be presented as a sensitivity study in order to characterize the potential effect on the transition change in risk.

The sensitivity study will result in an additional table (Table W-4) where the compliant plant model with conservative assumptions removed will be a replacement of the current compliant plant model (R4 model) in Table W-3.

The columns in Table W-4 will include: Fire Compartment; Compartment Description; Transitioning Plant CDF/LERF; As-Built Plant CDF/LERF; Compliant Plant with Assumptions Removed (identified as R5) CDF/LERF; Delta CDF/LERF (representing the risk increase associated with not fixing the VFDRs); Risk Offset CDF/LERF; and the Net Delta CDF/LERF. This will present the requested delta risk estimates without artificially increasing the compliant plant risk. Keeping Table W-3 as-is with the conservative assumptions included in the as-built, compliant, and post-transition model

will represent the upper bound of the compliant plant CDF/LERF. Including an additional table, Table W-4, will remove the assumptions in the compliant plant model and result in a lower bound approximation of the compliant plant CDF. This will then bound the transition change in risk. The updated Table W-3 and new Table W-4 will be provided in response to PRA RAI 03.

PRA RAI 08 – State-of-Knowledge Correlation

Section W.3.1 of the LAR states that risk estimates provided in LAR, Table W-3, are based on point estimates rather than mean values, but that an uncertainty analysis was performed as a sensitivity study to support use of point estimates. In the variant case of the sensitivity study, mean values from parametric probability distributions were used, and a state-of-knowledge correlation was performed. The study results show that the difference between the CDF and LERF values, based on point estimates and mean values are minimal. LAR, Section W.3.1, explains that the uncertainty analysis was performed using the software program UNCERT and that ignition frequency, weighting factors, and non-suppression probability were assumed to be independent. The LAR does not explain what fire parameters were correlated in the state-of-knowledge correlation. In addition, the UNCERT software tool is typically used for performing internal events uncertainty analysis, as opposed to fire event uncertainty analysis.

a) Clarify which fire parameters were addressed and correlated in the state-of-knowledge correlation. Provide justification for any circuit failure probabilities that were not addressed in the state-of-knowledge correlation, or confirm that this correlation will be included in the integrated analysis provided in response to PRA RAI 03.

Response:

During the state-of-knowledge correlation evaluation, distributions on equipment failure rates, initiating events, and HEP/dependencies were evaluated. In order to account for a more fire-specific correlation, the circuit failure mode likelihood analysis will be updated from what is presented in NUREG/CR-6850 Section 10 Option 1 (Failure Mode Probability Estimate table) to the updated guidance provided in NUREG/CR-7150, “Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE).” NUREG/CR-7150 contains the alpha and beta factors that can be used to create beta distributions for spurious event probabilities given different circuit configurations, which can then be used to evaluate the state-of-knowledge correlation. The correlation will be provided in response to PRA RAI 03.

The computer program UNCERT was developed to handle state-of-knowledge correlation evaluations as it generates a random value for each input parameter, and then derives the basic event probability from the inputs. Therefore, use of UNCERT to address the state-of-knowledge correlation is appropriate.

PRA RAI 08

b) Provide an implementation item for LAR, Table S-2, to update the use of point estimates and exclusion of state-of-knowledge correlation in the PRA maintenance and upgrade process, or justify why it is not needed. The justification should explain how DBNPS will ensure that the difference in estimated risk values based on using point values versus using mean values is still insignificant relative to the risk criteria for self-approval of post-transition changes (i.e., risk increase less than 1E-7/year for CDF and less than 1E-8/year for LERF).

Response:

An implementation item for LAR, Table S-2 is not required to update the use of point estimates and exclusion of the state of knowledge correlation (SOKC) in the PRA maintenance and update process because the technical adequacy is controlled to the standard within each notebook as they are the record that is maintained. The current procedure for the maintenance and upgrade process does not specify this level of detail. However, the fire PRA uncertainty notebook contains the evaluation of the SOKC uncertainty analysis which will be re-evaluated for every model update. The re-evaluation during each update will ensure that the difference in estimated risk values based on using point values versus using mean values is still insignificant relative to the risk criteria for self-approval of post-transition changes.

PRA RAI 09 – Fire Modeling of Inaccessible Floor Space

LAR, Table J-2 (p. J-19), states that, "[t]ransient and hot work fires were not postulated in locations within fire compartments that were considered inaccessible." Inaccessible areas are defined, in part, as, "An area where access is prohibited or extremely difficult due to the presence of a permanent fixture (as defined above) and there is no credible reason to expect transient material to accumulate (e.g., areas on top of half height rooms, confined areas behind a floor to ceiling stack of cable trays with no expected reason for access)."

Guidance in NUREG/CR-6850 pertaining to exclusion of locations for transient fire analysis states:

> **It is assumed that transient fires may occur at all areas of a plant unless precluded by design and/or operation, such as inside a BWR [boiling-water reactor] drywell or torus during power operation. Administrative controls significantly impact the characteristics and likelihood of transient fires, but they do not preclude their occurrence, since there is industry evidence of failure to follow administrative control procedures.**

The exclusion of locations from transient fire analysis because there is no credible reason to expect accumulation of transient material does not appear to

meet the guidance in NUREG/CR-6850. Furthermore, areas on top of half-height rooms and confined areas behind a floor-to-ceiling stack of cable trays are not visible during normal operation, and transient material accumulation could go unnoticed. Additionally, excluded transient fires near cable tray stacks could represent important risk contributors.

Justify that exclusion of locations because there is no credible reason to expect transient material, such as on top of half-height rooms and in confined areas behind a floor-to-ceiling stack of cable trays, is consistent with NRC guidance on transient fire analysis. Alternatively, confirm that these excluded transient locations will be included in the integrated analysis provided in response to PRA RAI 03.

Response:

For any locations that excluded transient fire analysis that are not explicitly precluded by design or operation (for example, the top of half-height rooms or behind floor-to-ceiling cable tray stacks), FENOC will review and include transient fire scenarios (analyze target impacts). The risk contribution of these scenarios will be included in the integrated analysis provided in the response to PRA RAI 03.

PRA RAI 10 – Reduced Transient Heat Release Rates

Attachment J of the LAR (p. J-20) identifies three fire compartments (D-01, BF-01, and BG-01) for which the transient fire heat release rate is reduced from 317 kilowatts to 69 kilowatts, or 142 kilowatts, per the guidance endorsed by the June 21, 2012, memo from Joseph Giitter, NRC, to Biff Bradley, NEI, "Recent Fire PRA Methods Review Panel Decisions and EPRI 1022993, 'Evaluation of Peak Heat Release Rates in Electrical Cabinets Fires'" (ADAMS Accession No. ML12171A583). LAR, Attachment J, explains that walkdowns and review of combustible materials, flammable liquids, and activities in these areas provide a basis for assuming that the reduced heat release rate is consistent with the specific attributes and considerations applicable to those locations. LAR, Attachment J, also explains that transient combustible control permits are required prior to taking any combustible or flammable material into these areas and that compensatory actions will be implemented as appropriate. The criteria that triggers the need for a combustible control permit for fire compartments BF-01 and BG-01 is wood scaffolding or flammable liquids greater than one gallon.

a) Explain why combustible control permits are not required for solid combustibles other than wood scaffolding and how administrative controls justify the reduced heat release rates used for Fire Compartment BF-01 and BG-01 without restricting such combustibles.

Response:

Fire compartments BF-01 and BG-01 will be updated to utilize the 98th percentile HRR recommended by NUREG/CR-6850 (317 kW). As these compartments will no longer use reduced transient heat release rates, transient combustible control permits are no longer required. LAR Table J-2 is hereby revised to delete fire compartments BF-01 and BG-01 as fire compartments crediting reduced heat release rates for transient fires. FENOC will include the risk contribution of these scenarios, with the updated HRR, in the integrated analysis provided in response to PRA RAI 03.

PRA RAI 10

b) Given that combustible and flammable materials are allowed to be taken into the three fire compartments for which a reduced heat release rate is credited, explain what kind of compensatory measures will be taken when such materials are taken into these fire compartments.

Response:

Fire compartments BF-01 and BG-01 will no longer utilize a reduced transient HRR, as discussed in the response to PRA RAI 10(a). For fire compartment D-01, there are compensatory measures in the administrative procedures discussed below.

The containment entry procedures provide the requirements for entering containment (fire compartment D-01) at power. Foreign material exclusion and fire watch requirements are included. These requirements ensure all materials taken into the fire compartment are accounted for when leaving, thereby reducing the probability of leaving these materials unattended while at-power. Materials are typically attended by personnel who entered with them.

The control of transient combustibles procedures restrict where transient combustibles can be stored within the fire compartment. Additionally, containment daily inspection and containment closeout inspection procedures provide the requirements for inspections of all accessible portions of fire compartment D-01 while at power. These inspections provide additional assurance that transient combustibles will not be left in the fire compartment while at-power.

Due to the compensatory measures including administrative procedures that limit the transient combustibles entering the fire compartment at-power, and limit transient combustibles left unattended in a fire compartment and ensure that any combustibles that enter the fire compartment are removed prior to closeout, the reduced transient HRR of 69 kW for fire compartment D-01 is appropriate.

PRA RAI 13 - Calculation of the Change in Risk

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA 805 further states that the change in public health risk arising from transition from the current fire protection program to an NFPA 805 based program, and all future plant changes to the program, shall be acceptable to the NRC. RG 1.174 provides quantitative guidelines on CDF and LERF and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of risk-informed changes. LAR, Section W.2.1, provides a high-level description of how the change in risk associated with VFDRs is determined, but additional information is needed to fully understand the approach

a) Describe the types of model adjustments made to remove different types of VFDRs from the compliant plant model, such as adding events or logic, or use of surrogate events. Clarify whether the approach is consistent with guidance in FAQ 08-0054, "Demonstrating Compliance with Chapter 4 of National Fire Protection Action 805, Revision 1" (see ADAMS Accession No. ML15016A280 and associated references therein). In addition, identify any major changes made to the fire PRA models or data for the purpose of evaluating VFDRs.

Response:

If a fire compartment had a post transition CDF less than (<) 5E-07, the risk to core damage from that fire compartment was set to 0 for the compliant case to bound the net delta risk required to be calculated. Therefore, the net delta risk equals the fire compartment post transition risk. If a fire compartment had a post transition CDF greater than or equal to (>/=) 5E-07, then a quantitative evaluation was performed.

Modeling adjustments made to remove different types of VFDRs from the compliant plant model include one of two options. The first option involves modeling the cables as being removed from the ZOI to represent a re-route of cables such that they are no longer located within the fire compartment of interest. This is done for a majority of the VFDRs that were identified.

The second option involves the use of operator actions to recover from the VFDR. For example, since instrument air is assumed to be failed, action to control the atmospheric vent valve (AVV) from the MCR (or primary control station) is lost. Therefore, instead of removing failure from all of the assumed targets such as power and control cables for the instrument air compressors, a RA such as locally throttling the AVV was used. This operator action was then set to false to represent perfect cognition and execution, therefore preventing this issue from contributing to core damage.

There was no addition of logic or change to data to represent the compliant case.

Each fire risk evaluation states: “The methodology for the Fire Risk Evaluation is based on the requirements in NFPA 805 and the guidance in Section 5.3 of NEI 04-02, as modified by Frequently Asked Question 08-0054, and Regulatory Guide 1.206.”
The six step process identified in FAQ 08-0054 was followed. Steps one through three of the FAQ are satisfied through the safe shutdown reports created for each fire compartment. Step four, Performance Based Evaluations, is discussed below. Steps five and six are completed in LAR Tables B-3 and 4-3.

Step four of FAQ 08-0054 provides guidance for licensees using the fire risk evaluation approach on how to calculate the increase in risk of a VFDR rather than modify the plant to bring it into compliance with the deterministic requirements of NFPA 805. The guidance directs licensees to perform the delta risk assessments on a fire area basis, which is consistent with NFPA 805 and Regulatory Guide 1.205. The FAQ recommends a team review of each VFDR. Those team reviews were performed early in the process such that if an investigation was required to identify how the VFDR would impact the fire PRA, it was kept for completeness and resolved either qualitatively or quantitatively in the fire risk evaluation process. The FAQ also states that the licensee can use a bounding approach, or a more detailed delta risk calculation. As stated previously, a bounding approach was taken for fire compartments with low risk (< 5E-07), while a more detailed delta risk calculation was performed for those with higher risk (>/= 5E-07). The delta risk was then compared against the acceptance criteria provided in Regulatory Position C.2.2.4 of RG 1.205. Finally, the risk offset is applied to only the fire risk as all other initiators are not quantified.

FAQ 07-0030 helps provide clarity on how to determine recovery actions required for compliance. This is a multi-step process. The primary control station is defined as being either the MCR or the auxiliary shutdown panel. Step two was followed as the population of recovery actions to resolve VFDRs was identified in response to the individual VFDR. Step three requires an evaluation of the additional risk due to use of the recovery action. This step is further covered in response to PRA RAI 14. Step four feasibility was assessed during the defense-in-depth (DID) expert panel meetings conducted as part of the fire risk evaluation process. The HRA was performed as described in step five.

As stated previously, there was no addition of logic or change to data to represent the compliant case, only failures were set to success to properly model the VFDR safety function resolution. Individual VFDRs were not assessed as the impact would not be resolved without recovering the full safety function. In addition, the values reported in LAR Attachment W are reflective of the entire set of VFDRs being resolved through the use of removing components from the ZOI or setting an operator action to false in order to mitigate the fire impacts on multiple pieces of equipment.

PRA RAI 13

b) For the MCR abandonment scenarios, describe the model adjustments that were made to remove the VFDRs to create the

compliant plant model. Clarify the statement in LAR Section W.2.1, that the compliant plant model for these scenarios "includes setting the contribution from control room abandonment due to loss of habitability to zero." Describe any differences in how VFDRs were identified for MCR abandonment areas compared to non-abandonment areas.

Response:

For the MCR, VFDRs were treated identically for the variances identified due to loss of safety function. An additional VFDR was identified for the MCR that would lead operators to abandon the MCR. For that VFDR, the assumption was made for the compliant case that the operators would have success at cooling down the plant and impact from additional failures were removed, thus the contribution from MCR abandonment on the compliant plant model is 0 (there is no longer a need to abandon the MCR).

PRA RAI 13

c) Describe the types of VFDRs identified, and discuss whether and how the VFDRs identified, but not modeled in the fire PRA, impact the risk estimates. Describe the qualitative rational for excluding VFDRs from the change-in-risk calculations.

Response:

The types of VFDRs identified include substantial barrier qualification, or lack of cable separation that could lead to a loss of a NSPC such as RCS inventory and pressure control, decay heat removal via the steam generators, reactivity control, process monitoring, and vital auxiliaries. VFDRs were identified if there was a potential impact on the above NSPC. Further analysis revealed that for VFDRs excluded using qualitative rationale there was no impact to the plant core damage or large early release frequency.

A summary of the VFDRs assessed as epsilon, and rationale for excluding them from impacting the PRA is as follows:

a) Thermal stressing a dry steam generator (SG): Analysis demonstrated that due to the height of the auxiliary feed water nozzles, water can be added to a hot, dry, depressurized SG without danger of excessive thermal stresses.
b) Spurious operation of containment spray (CS): Analysis demonstrated that if the CS pumps would start, all water would collect in the sump and operators could go to recirculation. The potential of a cold water shock to the reactor pressure vessel is a non-issue because it is a non-repetitive stress. A potential failure mode that is under further review is a spurious start of the CS pumps and a concurrent isolation of the

containment vacuum breakers that may result in containment buckling. This failure mode is still under review and will be included with the response to PRA RAI 02(g).

c) RCS inventory lost through small lines that do not cause a rapid loss of inventory: These lines are within the capacity of the make-up pump, and therefore the VFDRs are not considered a failure in the PRA success criteria.
d) Failure of pressurizer heaters to turn on or turn off: If the pressurizer heaters fail to turn on there is a potential to lose sub-cooling margin. In the event that sub-cooling margin is lost, the PRA credits natural circulation as well as the boiler – condenser mode of operation such that successful accident mitigation can occur without pressurizer heater control. If the pressurizer heaters fail to turn off (or spuriously start) there is the potential to over-pressurize the pressurizer and lift the PORV. The PORV lift due to a loss of main feedwater is now modeled for every fire because of the assumed loss of main feedwater, and therefore bounds this failure mode.
e) Loss of indications: Operator actions are either not credited in the PRA (and therefore has no impact on the success criteria), or there is redundant indication such that the impact is negligible.
f) Spurious start of pumps that have recirculation lines: These can't inject as the RCS is at a higher pressure, and therefore has no impact on the PRA.
g) Loss of pressurizer spray: This alone does not contribute to core damage or prevent the operation of other systems.
h) Failed equipment not credited in the PRA such as control room emergency ventilation (CREV), pressurizer auxiliary spray, or EDG week tank transfer pump: Upon a CREV failure, the control room heatup is a slow evaluation and operators would eventually establish portable cooling. Pressurizer auxiliary spray is not credited in the PRA and has two locked valves preventing injection due to failure of other valves. The DG day tanks are sufficient for an emergency DG run time of 24 hours. Therefore, the EDG week tanks are not modeled in the PRA.
i) The VFDRs against barriers that were analyzed by a performance-based evaluation were demonstrated to be adequate to withstand the fire effects of the hazard, and assigned a CDF/LERF value of negligible.

PRA RAI 14 - Additional Risk of Recovery Actions

LAR Attachment W, provides the total additional risk of RAs for each fire area but does not provide a description of how the additional risk of RAs was calculated. Footnote 3 to LAR, Attachment W, Table W-3, states:

> **There are many operator actions that are modeled in the PRA. In order to verify all operator actions are captured, the Additional Risk of RAs is equal to the All VFDR resolved case, and the delta is reported without credit for the risk offset; therefore, the threshold is exceeded. However, this value is not used for compliance with RG 1.174 or RG 1.205.**

The meaning of this statement is not completely clear. For example, the CDF and LERF values reported in LAR, Table W-3, in the "All VFDRs Fixed' column (which is presumably associated with the risk of the compliant plant) are not the same as

the CDF and LERF values presented in the "Additional Risk of RAs" column as stated in Footnote 3. Thus, it is not clear how the additional risk of RAs was calculated.

RAs to reduce risk are identified for certain fire compartments in LAR, Attachment G, Table G-1, but in some cases, no corresponding additional risk of RA values were reported in LAR Attachment W, Table W-3, for these same fire compartments (e.g., BF-01, BG-01, G-01, HH-01, UU-01). In addition, LAR Attachment G, Results of Step 3, states that "[a]ssessment of potential adverse effects of operator actions is addressed in the development of operator actions in the fire compartment specific Fire Risk Evaluations." It is not clear whether any potential adverse effects associated with operator actions were identified, and if they were identified, how they were treated in the fire PRA.

a) Explain how the additional risk of RAs was calculated, and confirm that the approach is consistent with guidance in FAQ 07-0030 and RG 1.205.

Response:

If a fire compartment had a post transition CDF < 5E-07, the risk of recovery action was set to the plant risk for the transitioning plant in order to get a conservative delta risk value.

If a fire compartment had a post transition CDF >/= 5E-07, then a quantitative evaluation was performed. In those cases, if the resolution to the VFDR safety function (primary side inventory and pressure control, secondary side decay heat removal, or support system) involved an operator manual action, the delta risk from the individual safety function was included in the delta risk from recovery action. If a compartment had an operator manual action to resolve multiple safety functions, then the sum of the safety function delta was included.

For example, in fire area S-01, there were VFDRs grouped to the secondary side DHR safety function which required an operator manual action to resolve the VFDR. There were also VFDRs grouped to the primary side inventory and pressure control safety function requiring a different operator manual action to resolve the VFDR. In this instance, the delta risk of recovery action is equal to the sum of the delta risk for both of those safety functions.

By taking the approach of setting the delta risk of recovery action to the post transition risk for the compartments found to have a CDF < 5E-07, the delta risk is set to an upper bound. Likewise, by taking the delta all case, the delta risk of recovery action is also set to an upper bound.

Since the fire compartment net delta risk is a decrease due to the addition of emergency feedwater and the FLEX equipment, the approach is acceptable per RG 1.205 Section 2.2.4.2 which states, "The NRC staff will not normally approve net risk increases in fire

areas where the previously approved recovery actions represent an additional risk above the acceptance guidelines in Regulatory Guide 1.174."

FAQ 07-0030 helps provide clarity on how to determine recovery actions required for compliance. Step one requires the definition of the primary control station. At DBNPS this is defined as being either the MCR or the auxiliary shutdown panel. Step two was followed as the population of recovery actions to resolve VFDRs was identified in response to the individual VFDR. Step three requires an evaluation of the additional risk of the use of the recovery action. Bullet two of FAQ 07-0030 was followed for the fire compartments with a transitioning fire risk greater than 5E-07 per year. This method is interpreted as meaning to evaluate the as-built fire PRA model with the operator action appropriately and explicitly modeled resulting in CDF1/LERF1. Then, eliminate the VFDR for the as-built fire PRA model to create a compliant case and evaluate (CDF2/LERF2). Once both values are known use the following equations to determine the delta CDF1-CDF2 and LERF1-LERF2.

For fire compartments with a transitioning fire risk less than 5E-07 per year the delta risk of recovery action is equal to the transitioning fire risk since the compliant plant risk is conservatively assumed to be negligible (0.00E+00) in order to obtain the largest delta risk.

Step four feasibility was assessed during the DID expert panel meetings conducted as part of the fire risk evaluation process. Step five HRA was performed in accordance with Part four of the ASME/ANS RA-Sa-2009 PRA Standard as endorsed by Regulatory Guide 1.200, Revision 2.

Regulatory Guide 1.205 directs the licensee to evaluate the risk increase due to the use of recovery actions when included in the risk change for each fire area. By providing the additional risk of recovery for each fire area in the LAR, Attachment W, Table W-3, the approach was consistent with the guidance provided.

While the 5E-07 per year compartment risk criteria was established based on the anticipation that summing the delta risk from all fire compartments will result in the total delta risk being consistent with RG 1.174 "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," the total delta risk on a compartment basis is not critical. The cumulative change in risk for the plant compared with RG 1.174 is the ultimate quantitative acceptance criteria. The risk results of the compartments analyzed suggest that none of these compartments are challenging to a successful transition to NFPA 805 compliance with regard to delta risk in the context of RG 1.174 and would not require further compartment specific plant design improvements on this basis.

PRA RAI 14

b) Reconcile the apparent inconsistencies in the LAR in which RAs to reduce risk are identified for fire compartments in LAR, Attachment

G, Table G-1, but no additional risk of RAs is reported in LAR, Attachment W, Table W-3, for these same fire compartments.

Response:

For compartments with a CDF less than the truncation value for quantification of 1E-10, a value of 0.00E+00 was reported (BF-01, BG-01, G-01, HH-01, UU-01, and so on). For those fire compartments, since the risk was of negligible impact, the risk of recovery actions was set to the value of the post-transition plant. In the cases where the CDF contribution was less than 1E-10, the risk of recovery actions was set to 0. In cases where the CDF contribution was greater than 1E-10 but less than 5E-07, the risk of recovery action was set to the post-transition plant value. In cases where the risk was greater than 5E-07 per year, a quantitative evaluation of the VFDRs was performed, and the delta risk of recovery action was assigned to the corresponding result.

Because of the approach taken, the risk reduction from the RA is not clear for those fire compartments with a CDF/LERF below truncation limits; however it can still be determined as having a negligible impact. To resolve this documentation issue, a note will be added to the bottom of LAR Table W-3 to more clearly explain why the risk of recovery action is assigned 0.00E+00. Attachment W will be updated and provided with the response to PRA RAI 03.

Attachment G will be reviewed to verify actions identified as risk recovery actions were modeled appropriately. A list of risk recovery actions per fire compartment will be provided in response to PRA RAI 03 in order to clarify any potential discrepancies between LAR attachments G and W.

PRA RAI 14

c) Explain whether any potential adverse effects associated with operator actions were identified, and if they were identified, how they were treated in the fire PRA.

Response:

A review of all MCR annunciators was completed during development of the operator actions credited in the fire PRA. Per that review, the fire PRA HRA concluded no new events were identified. Therefore, no new operator actions causing adverse effects were included into the fire PRA model.

In addition, implementation item DB-0572 includes updating the abnormal operating procedures (AOPs) for severe fires for the recovery actions evaluated. The actions that are ultimately credited in the AOPs will be re-reviewed per implementation item DB-1943 and included in the fire PRA when completed. It is not anticipated that procedure changes will cause adverse impacts; however, an implementation item has already been created to review the changes.

PRA RAI 15 – Large Risk Reduction Credit

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA 805 further states that the change in public health risk arising from transition from the current fire protection program to an NFPA 805-based program, and all future plant changes to the program, shall be acceptable to the NRC. RG 1.174 provides quantitative guidelines on CDF and LERF and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of risk-informed changes. The NRC staff notes that conservative calculations of the compliant plant CDF and LERF can lead to a non-conservative calculation of the ΔCDF and ΔLERF.

Appreciable risk reduction credit (i.e., large negative CDF and LERF values) is presented in LAR, Attachment W, Table W-3, for non-VFDR related modifications referred to as the "risk offset." The risk offset is defined to be the risk reduction resulting from modifications that do not bring a fire compartment into compliance with NFPA 805 deterministic requirements. Section 3.2.5 of RG 1.205 states that risk decreases may be combined with risk increases for the purposes of evaluating combined changes in accordance with regulatory positions presented in Sections 1.1 and 1.2 of RG 1.174, Revision 2, but that the total increase and total decrease in the ΔCDF and ΔLERF should be provided. LAR, Table W-3, appears to report the total decrease associated with non-VFDR modifications in the form of risk offset values, but does not appear to report the total increase associated with unresolved VFDRs.

a) Provide the total risk increase associated with unresolved VFDRs and the total risk decrease associated with non-VFDR modifications.

Response:

The total risk increase associated with unresolved VFDRs is identified in LAR Table W-3 under column heading "FRE Delta Risk." This value is 2.05E-03 per year for CDF and 6.56E-05 per year for LERF.

The total risk decrease associated with non-VFDR modifications is also listed in Table W-3 under column heading "Risk Offset." This value is 2.34E-03 per year for CDF and 7.46E-05 per year for LERF.

These values will be updated in response to PRA RAI 03, but the column identifiers will remain unchanged.

PRA RAI 15

b) Summarize the risk-significant scenarios for fire areas in the compliant plant model that are most significantly impacted by risk reduction modifications, and discuss the contribution of fire-induced failures for those scenarios.

Response:

Changes will be made to the compliant plant model in the response to PRA RAI 03, and the review of the risk-significant scenarios for the compliant plant model will be provided in response to PRA RAI 03. This will be performed similarly to how the post transition model scenarios are discussed in LAR Table W-1. The table will present the dominant scenario, scenario description, percent contribution, and dominant failures contributing to the sequence resulting in core damage.

PRA RAI 15

c) Discuss the impact of any important modeling assumptions contributing to the risk significant scenarios for important fire areas in the compliant plant model. Specifically, include discussion of conservative modeling assumptions made in the compliant plant model that may artificially reduce the calculated change in risk.

Response:

For the compliant plant model, no failures were added, only removed. There are some significant assumptions (untraced circuits, items identified as tier three, and unanalyzed circuits) that may overestimate compliant plant risk and lead to an underestimation of the transition change in risk. This is because a compliant plant may not actually fail all of this equipment in every scenario. It is therefore possible that the approach used in the LAR underestimates the change in risk. However, completely removing these failures (untraced circuits, items identified as tier three, and unanalyzed circuits) would likely underestimate the compliant plant risk, and overestimate the transition change in risk, as some of the equipment may still fail due to fire impacts. Results from both methods (retaining all failures and removing all failures in the compliant plant model) will be presented to bound the transition change in risk. As discussed in the responses to PRA RAIs 07(b) and 15(d), the updated Table W-3 and new Table W-4 will be provided in response to PRA RAI 03.

A more detailed description of the identified conservatisms is as follows:

1) Conservatisms within cable selection: As identified in PRA RAI 07, not all systems were selected for cable selection as this would take a considerable amount of resources to locate systems with little risk significance. Systems that were not included in cable selection included: instrument air, main feedwater, turbine plant

cooling water, and circulating water. Since cable tracing was not performed for these systems, the worst failure modes were included into the fire PRA.

Furthermore, as discussed in the response to PRA RAI 07, some non-coordinated cabinet loads were failed in specific fire areas.

As explained in the response to PRA RAI 15(d), failures of these systems and cabinets will be removed for the compliant plant model with conservative assumptions removed, and presented in the response to PRA RAI 03. With this approach, the compliant plant model risk will be underestimated, but can provide the lowest risk achievable given these circumstances.

2) Conservatisms within circuit failure mode likelihood analysis: In order to reduce the amount of resources analyzing every circuit where a change in position of a component could impact the fire PRA, only the spurious events that occurred within the top cutsets were evaluated while the rest were assigned a conservative value. Not refining the analysis resulted in an increase in plant risk due to the conservative value chosen.

   As explained in the response to PRA RAI 15(d), these conservative values will be removed for the compliant plant model with conservative assumptions removed, and presented in the response to PRA RAI 03. With this approach, the compliant plant model risk will be underestimated, but can provide the lowest risk achievable under these circumstances.

3) Conservatisms within detailed fire modeling: During the fire modeling scoping process, the full room burns were evaluated using the EPRI computer program FRANX. The results were then used to identify high, medium, and low risk components. The risk criteria were established on a per-compartment basis and did not follow a standard risk threshold. Relative importance measures were compared and engineering judgement was used to determine the risk thresholds in each compartment. The risk thresholds were then used to determine the amount of analysis to be performed on the component of interest. High risk components were measured from every fire source within the compartment and failed when known to be within the ZOI. Medium risk components were identified in drawings of the compartment by creating a ZOI circle on the plant drawing with a 1.5 factor applied to the ZOI radius for each ignition source. This 1.5 factor helped capture any uncertainty that could exist in the drawing. Finally, low risk components were assumed to be within the ZOI for every source within the compartment that propagated beyond the ignition source itself. Through these assumed failures, resources were allocated to support plant modifications instead of furthering the analysis. However, many additional components that contribute to core damage were failed.

   As explained in the response to PRA RAI 15(d), the low risk component failures will be removed from the ZOI in the compliant plant model with conservative assumptions removed, and presented in the response to PRA RAI 03. With this

approach, the compliant plant model risk will be underestimated, but can provide the lowest risk achievable given these circumstances.

PRA RAI 15

d) If conservative modeling of the compliant plant is identified as contributing to the underestimation of the total change in risk, demonstrate that the total risk increase associated with unresolved VFDRs is offset by the total risk decrease associated with risk reduction modifications even when the conservative modeling is removed. Alternatively, confirm that realistic modeling that does not underestimate the total change in risk will be used in the integrated analysis provided in response to PRA RAI 03.

Response:

The conservatisms listed in response to PRA RAI 15(c) increase the compliant plant risk. However, it is an underestimation in the compliant plant risk if all of these conservatisms were to be removed. Therefore, a lower bound sensitivity study will be determined in the response to PRA RAI 03. The sensitivity study will result in an additional table (Table W-4) where the compliant plant model with VFDRs resolved and assumptions removed will be a replacement of the current compliant plant model (R4 model) in Table W-3. The proposed Table W-4 is discussed in the response to PRA RAI 07(b).

The updated Table W-3 and new Table W-4 will be provided with the response to PRA RAI 03.

PRA RAI 16 – Defense-in-Depth (DID) and Safety Margin

Section 1.2 of NFPA 805 requires the fire protection standard to be based on the concept of DID, and specifies the three elements (or echelons) of DID that must be balanced. LAR, Section 4.5.2.2 provides a high-level description of how impacts on DID and safety margin were reviewed for the transition to NFPA 805 and includes a discussion of plant improvements made in response to this review. However, additional information is needed to fully understand the approach. Also, LAR, Section 4.5.2.2, states that "[f]ire protection features and systems relied upon to ensure DID were identified as a result of the assessment of DID," but LAR, Table 4-3 does not identify any fire protection systems or features to be credited for DID.

a) Explain the criteria used to determine when a substantial imbalance between DID echelons exist in the fire risk evaluations, and identify the types of plant features and administrative controls credited for providing DID for each of the three DID echelons.

Response:

The criteria used to determine when a substantial imbalance between DID echelons existed in a fire risk evaluation is outlined in the project instruction for fire risk evaluations.

A review of the impact of the change on DID was performed using the guidance from NEI 04-02. The review ensured that defense in depth was achieved when an adequate balance of the following elements (echelons) was provided:

- Prevent fires from starting
- Rapidly detect fires, control, and promptly extinguish those fires that do occur, thereby limiting damage
- Provide adequate level of fire protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

In general, the DID requirement was satisfied if the proposed change did not result in a substantial imbalance among these elements (or echelons).

The review of DID was typically qualitative and addressed each of the elements with respect to the proposed change.

Consistency with the DID philosophy was maintained if the following acceptance guidelines were met:

- A reasonable balance was preserved among 10 CFR 50.48(c) DID elements.
- Compensation for weaknesses in plant design, such as over-reliance on programmatic activities, and increased length of time or risk in performing those activities, was avoided.
- Pre-fire nuclear safety system redundancy, independence, and diversity were preserved commensurate with the expected frequency and consequences of challenges to the system and uncertainties (no risk outliers).
- Independence of DID elements was not degraded.
- Defenses against human errors were preserved.
- The intent of the General Design Criteria in Appendix A to 10 CFR 50 was maintained.

NFPA 805, Section 4.2.4.2, refers to the acceptance criteria used in the fire risk evaluation process to compare the risk associated with implementing a deterministic solution to the risk of a performance-based alternative solution. Therefore, fire protection systems and features required to demonstrate an adequate balance of DID are required by NFPA 805, Chapter 4. The VFDRs and associated values for CDF, LERF, and scenario consequences (conditional core damage probability, or CCDP) were evaluated to identify general DID echelon imbalances.

Potential methods to balance the DID features were identified and then further enhanced by a qualitative evaluation to ensure (by means of an iterative review) that an adequate balance of DID features was maintained for a fire compartment.

The iterative review of the DID echelons was performed by an expert panel, with the minimum personnel consisting of a manager, a fire protection engineer, and at least one individual with operations experience. These individuals assessed whether the DID echelons are balanced using the criteria above. In situations where there was an imbalance in the DID elements, the panel agreed on additional requirements to ensure that a balance of the DID elements was obtained. These additional requirements are documented in the fire risk evaluation to ensure the DID features (programmatic controls, physical upgrades, recovery actions, and so on) go above and beyond the existing requirement(s) with the purpose of bolstering weaknesses within the DID echelons, to maintain an overall balance.

The types of plant features and administrative controls credited to provide DID for each of the three DID echelons is shown below. The items in the left column were the existing methods used to provide a balance in DID echelons. In the event there was an imbalance in DID echelons, the items in the right column were considered. These, or other recommended changes determined by the expert panel, were included in the fire risk evaluations to ensure a balance in the DID echelons was obtained.

Considerations for Defense in Depth Determination	
Method of Providing DID	Considerations
Echelon 1: Prevent fires from starting	
• Combustible control • Hot work control	Combustible and hot work controls are fundamental elements of defense in depth, and as such, are always in place. The issue to be considered during the fire risk evaluations is whether this element needs to be strengthened to offset a weakness in another echelon thereby providing a reasonable balance. Considerations include: • Creating a new transient-free area • Modifying an existing transient-free area • Reviewing the existing procedures for the associated echelons and note any required changes The fire scenarios involved in the fire risk evaluation quantitative calculation should be reviewed to determine if additional controls should be added. Review the remaining elements of defense in depth to ensure an overreliance is not placed on programmatic activities for weaknesses in plant design.

Considerations for Defense in Depth Determination

Method of Providing DID	Considerations
Echelon 2: Rapidly detect, control, and promptly extinguish those fires that do occur thereby limiting fire damage	
• Detection system • Automatic fire suppression • Portable fire extinguishers provided for the area • Hose stations and hydrants provided for the area • Pre-fire plan	Automatic suppression and detection may or may not exist in the fire compartment of concern. The issue to be considered during the fire risk evaluation is whether installed suppression and/or detection is required for defense in depth or whether suppression and/or detection needs to be strengthened to offset a weakness in another echelon, thereby providing a reasonable balance. Considerations include: • If a fire compartment contains both suppression and detection and firefighting activities would be challenging, both detection and suppression may be required. • If a fire compartment contains both suppression and detection and firefighting activities would not be challenging, detection and manual firefighting are required (consider enhancing the pre-plans). • If a fire compartment contains detection and a recovery action is required, the detection system may be required. • If a fire compartment contains neither suppression nor detection and a recovery action is required, consider adding detection or suppression. • Review the existing procedures for the associated echelons and note any required changes. The fire scenarios involved in the fire risk evaluation quantitative calculation should be reviewed to determine the types of fires, and reliance on suppression probability should be evaluated in the area to best determine options for this element of defense in depth. Provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed.

Considerations for Defense in Depth Determination

Method of Providing DID	Considerations
Echelon 3: Provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed	
• Walls, floors, ceilings, and structural elements are rated or have been evaluated as adequate for the hazard. • Penetrations in the fire compartment barrier are rated or have been evaluated as adequate for the hazard. • Supplemental barriers (ERFBS, cable tray covers, combustible liquid dikes/drains). • Fire-rated cable. • Reactor coolant pump oil collection system (as applicable). • Guidance provided to operations personnel detailing the required success path(s), including recovery actions to achieve nuclear safety performance criteria.	If fires occur and they are not rapidly detected and promptly extinguished, then the third echelon of defense in depth would be relied upon. The issue to be considered during the fire risk evaluation is whether existing separation is adequate or whether additional measures (supplemental barriers, fire-rated cable, or recovery actions) are required to offset a weakness in another echelon, thereby providing a reasonable balance. Considerations include: • If the variance is never affected in the same fire scenario, internal fire compartment separation may be adequate and no additional reliance on recovery actions is necessary. • If the variance is affected in the same fire scenario, internal fire compartment separation may not be adequate, and reliance on a recovery action may be necessary. • If the consequence associated with the variances is high regardless of whether it is in the same scenario, a recovery action and/or reliance on supplemental barriers should be considered. • There are known modeling differences between a fire PRA and nuclear safety capability assessment (NSCA) due to different success criteria, end states. Although a variance may be associated with a function that is not considered a significant contribution to core damage frequency, the variance may be considered important enough to the NSCA to retain as a recovery action.[1] • Review the existing procedures for the associated echelons and note any required changes. The fire scenarios involved in the fire risk evaluation quantitative calculation should be reviewed for the fires evaluated and the consequences in the area, to best determine options for this element of defense in depth.

[1] An example would be components in the NSCA associated with maintaining reactor core natural circulation that are not modeled explicitly in the fire PRA since they are not part of a core damage sequence.

PRA RAI 16

b) Clarify what fire protection features and systems were relied upon to ensure DID and explain why none are identified in LAR, Table 4-3. Explain how DID is ensured, given that automatic fire detection systems are not credited for maintaining DID.

Response:

The fire protection features and systems that were relied upon to ensure DID were: 1) additional administrative procedural controls for transient combustible material in several fire compartments; and 2) automatic fire detection systems in two fire compartments.

The reason that DID was not identified in LAR Table 4-3 for any fire protection features or systems was that "R – Risk Criteria" was mistakenly selected instead of "D – Defense-in-Depth Criteria" during the development of the table. Upon review, it was determined that DID should have been identified in fire compartment rooms A-02, A-03, A-05, A-06, A-08, AB-01, AB-02, AB-04, DD-01, EE-01, T-01, D-01, and II-03. The change also reflects the decision to credit some items as being required for DID based on consideration of both the risk and DID. These changes and other associated corrections are hereby made to Table 4-3 of the LAR:

- In fire compartment A-02, in rooms 110 and 110A, "R1" is changed to "D," and in rooms 112, 116, 117, 117A, 119, 120, 121, and 122, "R1" should be deleted,
- In fire compartment A-03, room 114, "R2" should be changed to "D".
- In fire compartment A-05, room 124, "R1" should be changed to "D".
- In fire compartment A-06, room 127E, "R2" should be changed to "D".
- In fire compartment A-07, room 236, "R1" should be deleted.
- In fire compartment A-08, room 314, "R2" should be changed to "D".
- In fire compartment A-09, room 115CC, add "R*" in the "Other" column, and in room 115CC and 314CC, add "R*: Cable Tray Systems (see notes below)" to the "Comments" column.
- In fire compartment AB-01, rooms 105 and 113, "R1" should be deleted, and in room 113A, "R1" should be changed to "D".
- In fire compartment AB-02, room 127W, "R1" should be changed to "D".
- In fire compartment AB-04, rooms 225 and 226A, "R1" should be changed to "D".
- In fire compartment D-01, rooms 214, 215, 216, 218, 220, 317, 410, and 580, "D" should be added in the "Auto Detection" column.
- In fire compartment DD-01, room 422A, "D" should be added in the "Other" column and "D: Additional combustible controls" should be added in the "Comments" column.
- In fire compartment EE-01, in room 500, add "R*: Cable Tray Systems (see notes below)" in the "Comments" column, in room 501, add "D" and "R1" in the "Other" column and add "R1: Conduit embedded in concrete" in the "Comments" column, and in room 515, add "R1" in the "Other" column and "R1: Conduit embedded in

concrete" in the "Comments" column.
- In fire compartment HH-01, room 603, "R2" should be deleted.
- In fire compartment II-01, room 252, add "R1" in the "Other" column.
- In fire compartment II-03, room 333, add "D" in the "Auto Detection" and "Auto Suppression" columns.
- In fire compartment P-03, room 322, "R1" should be deleted.
- In fire compartment T-01, room 328, "D" should be added in the "Other" column and "D: Additional combustible controls" should be added in the "Comments" column.
- In fire compartment U-01, room 310, "R1" should be deleted.

The following two justification methods explain how DID is ensured given that automatic fire detection systems are not credited to maintain DID (except for the two fire compartments D-01 and II-03 as discussed below).

First Method
The initial screening criteria in the fire risk evaluation that were used to determine the overall risk significance of the individual fire compartment are the following:

- CDF > 1E-06 per year, or
- LERF > 1E-07 per year, or
- CCDP > 1E-01

If CDF, LERF, or CCDP was greater than the initial screening criteria and the automatic fire detection systems had not already been required for risk reduction, then crediting the automatic fire detection system for the fire compartment would be required. There were no fire compartments that screened in.

Second Method
The additional screening criteria in the fire risk evaluation that were used to determine if automatic fire detection systems were required for DID is shown below:

- Echelon 1 and/or 2 was credited for risk reduction, and
  - CDF > 1E-08 per year, or
  - LERF > 1E-09 per year

Using these screening criteria, the expert panel discussed if additional DID requirements were needed.

In developing the LAR, the expert panel decided that no automatic fire detection systems were needed for DID. After the LAR submittal, a second expert panel was convened, and it was decided that two fire compartments would require automatic fire detection systems for DID: fire compartments D-01 and II-03.

PRA RAI 16

c) Discuss the approach for reviewing safety margin using the NEI 04-02, Revision 2, criteria for assessing safety margin in the fire risk evaluations.

Response:

The approach for reviewing safety margin using the NEI 04-02, Revision 2, criteria in the fire risk evaluations is described in the project instruction for fire risk evaluations.

The safety margin assessment included a review of the impact of the change on safety margin. The guidelines used for making that assessment are summarized below:

- Codes and standards or their alternatives were reviewed to ensure that they were acceptable to the NRC, and
- Safety analysis acceptance criteria in the licensing bases (UFSAR, supporting analyses) were met, or it was determined that there was sufficient margin to account for analysis and data uncertainty.

The requirements related to safety margins for the change analysis are described for each of the specific analysis types used in support of the fire risk evaluation. These analyses can be grouped into the following categories:

- Fire Modeling – the instruction states:

  For fire modeling inputs to the fire PRA, document the results of the qualitative safety margin review. Include a review of the use of applicable codes and standards developed by industry and NRC staff to ensure realistic yet conservative results.

- Plant System Performance – the section states, in part:

  The methods, input parameters, and acceptance criteria used in these analyses need to be reviewed against that used for the plant design basis events. This review would serve to establish that the safety margin inherent in the analyses for the plant design basis events has been preserved in the analysis for the fire event and, therefore, satisfy the requirements of this section.

  From a safety margin perspective the evaluation of the plant system performance should address the following topics:

  - Were input parameters for plant performance analyses (for example, heat transfer coefficients, pump performance curves) altered from those used for plant design basis events such that the margin was lessened?

- Were codes and standards used to determine plant system performance acceptable to the NRC?

- PRA Logic Model – the section states, in part:

  From a safety margin perspective, the evaluation of the PRA logic model should address the following topic:

  - Were the risk-informed, performance-based processes utilized based upon NFPA 805, 2001 edition, endorsed by the NRC in 10 CFR 50.48(c)?
  - Was the fire risk evaluation process in accordance with NEI 04-02, Revision 2, which is endorsed by the NRC in RG 1.205, Revision 1?
  - Was the Fire PRA developed in accordance with NUREG/CR-6850, which was developed jointly between the NRC and EPRI?

Radiological Release (RR) RAI 01

Attachment E of the LAR states several times that, "A bounding analysis was developed that determined that in the event of a fire the radiological release would not exceed 10 CFR limits." As part of the NRC staff's audit of licensee documents, the staff reviewed Radioactive Material Release Calculation RP-15-01. However, it is not clear if this calculation is the bounding analysis referred to in LAR, Attachment E, or if this is the only bounding analysis.

For each bounding analysis referred to in LAR Attachment E, provide the title and a summary of the assumptions, methodologies, and resulting doses of the analysis.

Response:

Calculation RP-15-01 is titled "NFPA 805: Radioactive Material Release Calculation." This analysis is the only bounding analysis for LAR Attachment E and applies to each fire area (CAF, II-01, NSGSF, OS, OSGSF, T-01 and TAB) referenced in LAR Attachment E, Table E-2.

This calculation bounds the assumptions associated with a liquid and gaseous radiation release to any unrestricted area due to the direct effects of a fire and suppression activities, and ensures the resultant release would not exceed applicable 10 CFR 20.1301 and 10 CFR 20.1302 dose limits for members of the public. This is true for fire areas with no engineering controls (such as high efficiency particulate air and charcoal filters, monitored release, and so on), or areas with fire suppression, but does not involve fuel damage.

Calculation assumptions include:

- An hour-and-a-half fire duration,
- One sea-land container open during the fire,
- The size, volume, and weight of combustible waste in a sea-land container,
- Maximum curie content of dry activated waste,
- 60 percent release of liquid activity due to fire-fighting suppression activities,
- 100 gallon per minute fire suppression hose flow,
- Atmospheric dispersion factors taken from the offsite dose calculation manual (ODCM)

The calculation was performed for liquid and gaseous release due to a fire without engineering controls (described above), or due to a fire in areas with fire suppression and long-term outside radioactive material storage. The analysis includes the highest dose sea-land container with dry active waste shipped during the year 2014, and worst case sea-land container based on an administrative limit of 80 percent of the 2 meter dose rate shipping criteria (10 mRem/hr).

For gaseous and liquid releases, the nuclide analysis is based on meeting the administrative 15 mRem/hr dose rate at 1 meter and at 8 mRem/hr dose rate at 2 meters. The sea-land container gaseous and liquid release assumptions were input into the computer program RadMan. RadMan is industry-accepted software, which characterizes and classifies, manifests, and documents packaged radioactive waste. The nuclide characterization data provided by RadMan was then utilized to perform the radioactive release calculations in accordance with the ODCM.

Compliance was demonstrated by ensuring the concentrations of radioactive material released in liquid effluents to an unrestricted area is limited to the concentrations specified in 10 CFR Part 20 for radionuclides, other than dissolved or entrained noble gases. Other assumptions used in the calculation includes a dilution volume of 12 million gallons in the training center pond and a release fraction for contaminated fire water run-off is 60 percent of the activity released as liquid. The run off to reach an unrestricted area will need to travel to the storm sewer drain, which releases to the training center pond, which discharges to marsh pool number three. The results of the calculations for the fraction of liquid release to an unrestricted area, based on the annual average concentration calculated, was found to be below the values specified in Table 2 of Appendix B to Part 20.

Compliance was also demonstrated by ensuring that potential exposure from iodine-131, tritium, and all radionuclides in particulate form with half-lives greater than eight days is less than or equal to 1500 mRem per year averaged over one hour to any organ. The maximum dose to any organ was less than the 100 mRem to members of the public, in accordance with 10 CFR 20.1301 limits.

RR RAI 02

LAR, Attachment E, Table E-1, states that compartment MC-01 should be screened in for analysis of radiological containment and monitoring actions associated with firefighting operations. However, LAR, Table E-2, states that compartment MC-01 was screened out and does not provide the results of the engineered controls review for liquid and gaseous effluents associated with firefighting activities.

Clarify whether compartment MC-01 should be screened in for analysis of radiological containment and monitoring actions associated with firefighting. If so, provide the results of the engineered controls review for compartment MC-01.

Response:

Compartment MC-01 was reviewed for radiological releases in accordance with the project instruction for radiological release transition review during fire suppression activities. The review states, “The pre-fire plan identifies the area around BWST and PWST [primary water storage tank] as a radiological controlled area. This is based on dose and is not considered a potential for contaminated fire suppression water or smoke.” The conclusion of this review is, “Although the fire compartment is located near radiologically controlled areas, the fire compartment does not have the potential for contamination therefore it satisfies the NFPA-805, section 4.3 criteria.”

LAR, Attachment E, Table E-1 was incorrect, and is hereby changed to screen out compartment MC-01.