VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

July 23, 2009

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No.	09-301
NL&OS/ETS	R2
Docket No.	50-338
License No.	NPF-4

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNIT 1
LICENSE AMENDMENT REQUEST
ONE TIME EXTENSION OF THE COMPLETION TIME FOR
1 "J" - 480 VOLT AC DISTRIBUTION SUBSYSTEM

Pursuant to 10 CFR 50.90, Dominion requests an Amendment of the Facility Operating License, in the form of changes to the Technical Specifications to Facility Operating License Number NPF-4 for North Anna Power Station Unit 1. The proposed change will revise Technical Specification (TS) 3.8.9 Condition A, by adding a Note to allow a one-time 72 hour Completion Time to interrogate a failed breaker and the associated Unit 1 "J" 480 Volt emergency bus switchgear to ensure the emergency power system continues operation in a reliable condition for the remainder of the operating cycle. The request proactively addresses the condition after North Anna Unit 1 experienced a small fire in a breaker for a control rod drive mechanism (CRDM) fan.

Dominion requests that the proposed change be processed on an expedited basis. Although there is no evidence of damage in the emergency bus switchgear that would question operability, Dominion plans to proactively de-energize the bus (i.e., Motor Control Centers (MCCs) 1J1-2S and 2N) to interrogate the failed breaker and ensure the MCCs 1J1-2S and 2N continue in a reliable condition for the remainder of the operating cycle. The proposed change is intended to avert the known risks from complex actions in support of plant shutdown and startup evolutions, as well as permit a proactive inspection of the Unit 1 MCCs 1J1-2S and 2N.

The proposed change is based on a deterministic analysis and a risk-informed evaluation. The deterministic analysis was compiled using data from approved Station Load Lists, simulator runs of design basis accident scenarios with the 1J1-2S / 2N MCCs de-energized, and review from Senior Reactor Operators (SROs). The risk-informed evaluation was performed in accordance with Regulatory Guides (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications." A discussion of the proposed Technical Specifications change is provided in Attachment 1. The marked-up and proposed Technical Specifications pages are provided in Attachments 2 and 3, respectively. The Deterministic Analysis is provided in Attachment 4. The Probabilistic Risk Assessment is provided in Attachment 5.

We have evaluated the proposed Technical Specifications change and have determined that it does not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for that determination is provided in Attachment 1. We have also determined that operation with the proposed change will not result in any significant increase in the amount of effluents that may be released offsite and no significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed changes. The basis for that determination is also provided in Attachment 1.

Dominion requests that the proposed Technical Specification change be reviewed on an expedited basis. The extended Completion Time will expire upon returning the Unit 1 "J" 480 Volt AC distribution subsystem to OPERABLE, but no later than 72 hours after entering the extended Completion Time of TS 3.8.9 Condition A.

If you have any further questions or require additional information, please contact Mr. Thomas Shaub at (804) 273-2763.

Very truly yours,

J. Alan Price
Vice President – Nuclear Engineering

Attachments

1. Discussion of Technical Specification Change
2. Mark-up of Unit 1 Technical Specifications Change
3. Proposed Unit 1 Technical Specifications Change
4. Supporting Documentation: Deterministic Analysis
5. Supporting Documentation: Probabilistic Risk Analysis

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. Alan Price who is Vice President – Nuclear Engineering of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 23 day of July, 2009.

My Commission Expires: 4/30/13.

Notary Public

(SEAL)

GINGER L. ALLIGOOD
Notary Public
Commonwealth of Virginia
310847
My Commission Expires Apr 30, 2013

Commitments made in this letter:

1. The following actions will be taken during the extended Completion Time to provide additional assurance that public health and safety will not be adversely affected by this request.

    - Planned maintenance that may result in the unavailability of other equipment within the scope of Maintenance Rule (a)(4) program will be prohibited. For example:
        - There will be no planned maintenance on either Units' Emergency Diesel Generators, the Unit 1 "H" emergency bus, the "F" transfer bus, the switchyard, the Alternate AC Diesel Generator, or the reserve station service transformers.
        - Protected equipment will be identified and signs posted, including but not limited to: Unit 1 "H" emergency bus, the "F" transfer bus, the Alternate AC Diesel Generator (AAC DG), and the Unit 1 "H" Emergency Diesel Generator.
        - The two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System will be maintained operable.
    - AFW pump 3A will be verified operable and signs posted.
    - The MOVs on the TDAFW and MDAFW 3B's flow path will be verified to be open, kept open, and de-energized.
    - An Operator will be designated as the AFW valve Operator and will be deployed to manually operate the de-energized AFW MOVs, if needed.
    - A sound-powered phone system will be installed for communication between the AFW pump house and the Control Room.
    - A Maintenance Operating Procedure (MOP) will be utilized to control the removal of MCCs 1J1-2S and 2N electrical loads, to ensure compliance with Technical Specifications, and to verify that safety function is maintained throughout the extended 72 hour Completion Time. This includes posting of signs for protected equipment.
    - A contingency plan will be added to applicable Electrical Maintenance procedures to expedite re-energizing the MCCs, if needed.
    - Risk awareness briefings will be conducted for maintenance and operations personnel prior to the work.
    - Maintenance will be performed around-the-clock to minimize the time spent with equipment unavailable.
    - Unit 2 charging pumps and both Auxiliary Service Water pumps will be verified operable.
    - Work that may cause a trip hazard (e.g., surveillance in instrument racks) will not be performed.
    - An Operations Standing Order will require an operator action to manually open 1-RH-MOV-1701 following a Steam Generator Tube Rupture, if needed.

Commitments made in this letter (continued):

- Pressurizer Operated Relief Valve (PORV) 1-RC-PCV-1456 will be placed in manual operation to prevent the possibility of a Small Break LOCA should the PORV cycle automatically and fail to re-seat while the PORV block valve (1-RC-MOV-1535) is de-energized open.
- Fire watches will be established in the Cable Vault and Tunnel area and the Service Water Pump House (Fire Area 12).

2. A root cause evaluation of the breaker damage will be performed to establish 1) if a common cause failure mode exists and 2) the extent of the condition. Based on the root cause evaluation results, the appropriate corrective actions will be taken.

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Attachment 1

Discussion of Technical Specification Change

North Anna Power Station Unit 1
Virginia Electric and Power Company
(Dominion)

Table of Contents

1.0 Introduction

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion) requests an amendment to Facility Operating License Number NPF-4, in the form of a change to the Technical Specifications (TS) for North Anna Power Station Unit 1. The proposed change will revise the Completion Time (CT) for Technical Specification (TS) 3.8.9 Condition A by adding a note to the CT to allow a One Time Only 72 hour CT to interrogate the failed breaker and the Unit 1 “J” 480 Volt AC distribution subsystem (Motor Control Centers (MCCs) 1J1-2N and 2S) and to ensure the distribution subsystem continues operation in a reliable condition for the remainder of the operating cycle.

The request proactively addresses the condition after North Anna Unit 1 experienced a small fire in a breaker for a control rod drive mechanism (CRDM) cooling fan. The proposed change is based on a deterministic analysis and a risk-informed evaluation. The deterministic analysis was compiled using data from approved Station Load Lists, simulator runs of design basis accident scenarios with the MCCs 1J1-2N and 2S de-energized, and review from Senior Reactor Operators (SROs). The risk-informed evaluation was performed in accordance with Regulatory Guides (RG) 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis,” [RG 01] and RG 1.177, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,” [RG 02]. Although the operability of MCCs 1J1-2N / 2S is not in question, Dominion requests an expedited review to provide an opportunity to interrogate the damaged breaker, and to fully investigate the extent of condition to ensure continued bus reliability.

The proposed change qualifies for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Therefore, no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed change.

2.0 Background

On April 22, 2009, at approximately 0500 hours, Operations personnel identified a strong odor in the North Anna Unit 1 Cable Vault area. Subsequent investigation identified that the odor was coming from circuit breaker 01-EE-BKR-1J1-2S-J1 associated with the "D" Control Rod Drive Mechanism (CRDM) Fan (1-HV-F-37D). Operations personnel locally opened the circuit breaker to place it in a safe condition. The fan had stopped approximately 30 minutes prior to the event. The fan is not safety-related and not required for safe shutdown. However, the breaker is safety-related since it serves to isolate the CRDM from the safety-related Motor Control Centers (MCCs) 1J1-2N and 2S.

When Operations personnel opened the circuit breaker cabinet a small (6-inch) flame was observed. Operations personnel used a CO2 extinguisher on the internals of the circuit breaker to quickly extinguish the fire. The initial assessment of the breaker and MCC did not identify any condition that would adversely affect emergency bus operability. However, in order to complete a more comprehensive inspection, troubleshooting and repair, if necessary, of the 480 Volt AC distribution subsystem (MCCs 1J1-2N and 2S), additional time beyond the eight hours provided by the existing CT will be necessary.

The requested one-time change is needed to interrogate the failed breaker and the Unit 1 "J" 480 Volt AC distribution subsystem (MCCs 1J1-2N and 2S) to ensure the emergency distribution subsystem continues in a reliable condition for the remainder of the operating cycle. Inspection and repair of the breaker and MCCs is estimated to take approximately fifty-two hours under the worst case scenario and consists of the following:

- Tag out the Unit 1 480 Volt AC distribution subsystem (1-EE-MCC-1J1-2N and 2S)
- Remove emergency bus circuit breaker 01-EE-BKR-1J1-2S-J1
- Perform emergency bus motor control center (MCC) inspection and cleaning
- Remove circuit breaker 01-EE-BKR-1J1-2S-J2L and J2R (located below 1-EE-BKR-1J1-2S-J1)
- Megger the 1J 480V emergency bus
- If meggering is unsatisfactory, disconnect 1J1-2N
- Re-megger MCCs, if required
- If necessary, complete clean / repair of 1-EE-MCC-1J1-2N and 2S
- Reconnect MCCs and re-megger
- Reinstall circuit breakers
- Clear tag-outs, and
- Re-energize 1-EE-MCC-1J1-2N and 2S

The extended CT will expire upon returning the Unit 1 "J" 480 Volt AC distribution subsystem (MCCs 1J1-2N and 2S) to OPERABLE, but no later than 72 hours after entering the extended CT of TS 3.8.9 Condition A.

Once the affected breaker is removed from the MCC, a root cause evaluation will be performed to assess: 1) common cause failure potential and 2) the extent of condition. Based on the root cause evaluation results, the appropriate corrective action will be taken. Currently there are 133 breakers of the same design installed in North Anna Units 1 and 2.

A Maintenance Operating Procedure will be utilized to control the removal of MCCs 1J1-2N and 2S electrical loads, to ensure compliance with Technical Specifications, and to verify that safety function is maintained throughout the extended 72 hour Completion Time. This will ensure that any loss of safety function is detected and appropriate actions implemented. This includes posting of signs for protected equipment.

We have evaluated the Unit 1 "J" 480 Volt AC distribution subsystem and supported components and determined that the safety functions of each component or system will be maintained during the extended completion time. A simulator verification for reactor trip, steam generator tube rupture and large break LOCA was performed, which verified that safety functions were maintained for each accident.

The proposed one-time, extended CT change in this license amendment request has been evaluated in accordance with Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications." The approach addresses, as documented in this report, the impact on defense-in-depth and the impact on safety margins, as well as an evaluation of the impact on risk. The risk evaluation considers the three-tiered approach as presented by the NRC in Regulatory Guide 1.177. Tier 1, "PRA Capability and Insights," assesses the impact of the proposed Completion Time changes on core damage frequency (CDF), incremental conditional core damage probability (ICCDP), large early release frequency (LERF), and incremental conditional large early release probability (ICLERP). Tier 2, "Avoidance of Risk-Significant Plant Configurations," considers potential risk-significant plant operating configurations, and Tier 3, "Risk-Informed Plant Configuration Control and Management," assesses emerging plant conditions.

Use of the extended CT will be limited to one time. Planned maintenance that may result in the unavailability/nonfunctionality of Maintenance Rule (a)(4) components will be prohibited during the extended CT. Any emergent maintenance will be controlled in accordance with 10 CFR 50.65(a)(4), Maintenance Rule. Actions listed in Section 6.2, "Defense in Depth", will be taken prior to de-energizing the MCCs to improve defense-in-depth during the extended CT duration. The CDF impact and the LERF impact, as well as the ICCDP and ICLERP associated with the proposed CT change are summarized below. The CDF and LERF values meet the acceptance criteria in RG 1.174 and RG 1.177 for the proposed change.

3.0 Proposed Technical Specification Change

3.1 Need for Proposed Change

The proposed one time change to the CT of Technical Specification 3.8.9, Condition A, is necessary to perform a thorough inspection of the MCC once the damaged breaker is removed from the cubicle and avoid an unnecessary shutdown of the plant in the event of identifying a minor degradation or discrepancy during the interrogation of the Unit 1 "J" 480 Volt AC distribution subsystem switchgear. Although there is no evidence of damage in the distribution subsystem that would question operability, Dominion plans to proactively de-energize the 480 Volt emergency distribution subsystem (MCCs 1J1-2N and 2S) to 1) perform a thorough inspection of the MCCs to ensure that the emergency distribution subsystem is and continues to be in a reliable condition for the remainder of the operating cycle, and 2) interrogate the failed breaker for common cause and determine the extent of the condition. The proposed change will permit a proactive inspection of the Unit 1 "J" 480 Volt distribution subsystem switchgear (MCCs 1J1-2N and 2S) as well as avert known risks from complex actions in support of plant shutdown and startup evolutions.

3.2 Description of Proposed Change

The following Note is to be added to the Completion Time of TS 3.8.9 Condition A:

> "The One Time Only Completion Time for maintenance on the 1J1-2N / 2S MCCs is 72 hours."

See Attachment 2 of this License Amendment Request (LAR) for the Mark-up of the Unit 1 Technical Specifications Change.

3.3 Basis for the Technical Specification Change

The proposed one time CT change from 8 hours to 72 hours to permit maintenance of the Unit 1 "J" 480 Volt AC distribution subsystem (MCCs 1J1-2N and 2S) is based on a deterministic analysis and a risk-informed analysis. These analyses are summarized in Sections 5 and 6 of this Discussion of Changes. Additional information associated with the deterministic analysis and risk-informed analysis is included in Attachments 4 and 5, respectively.

4.0 System Description

The station electric power system sources are the station service transformers, the reserve station service transformers, the alternate ac diesel generator, and the emergency diesel generators. The station service transformers are also referred to as the normal source, the reserve station service transformers are referred to as the preferred source and the emergency diesel generators as the standby source. These sources feed the distribution system (station service power system) consisting of the 4160 Volt normal and emergency switchgear, 480 Volt normal and emergency unit substations, 480 Volt normal and emergency motor control centers, and the safety-related equipment and auxiliaries necessary for safe shutdown of the reactor and power plant. In addition, the reserve station service transformers feed the intake structure consisting of two 4160 Volt buses, three 480 Volt transformers, two 480 Volt motor control centers, and one 480 Volt load center.

The reserve station service power source is connected to the distribution system by several different feeds. The feeds connect the three 34.5/4.16-kV reserve station service transformers to the following:

1. Six normal station service buses (three per unit).
2. Four emergency buses (two per unit).
3. Two intake structure buses (one per unit).

The normal buses are connected to the transformers via 5-inch overhead pipe bus and cables in tray. These feeds originate at the transformer low-side bushing and extend from the transformers to the north wall of the turbine building via the 5-inch pipe bus. The cables connect to the pipe bus at the north wall, are carried in trays over the turbine building, and are terminated in the normal bus switchgear located on the top floor of the service building. The normal bus feeds from A and B reserve transformers are physically separated from the C feed.

The four emergency buses are fed from the reserve station service transformers by 4160 Volt cable runs in underground duct, conduit, and cable tray. The first path has cables in duct from reserve station service transformers A and B to the turbine building where it continues in conduit and then in cable tray to the emergency bus feeder breakers D1 and E1, respectively. The second path is for reserve station service transformer C cables, which run in duct to the service building so that they are kept separate from the cable runs of the other two transformers. These cables then continue in conduit and cable tray to the emergency bus feeder breaker F1. Feeder breakers D1 and E1 are separated from breaker F1 by the wall separating Unit 1 switchgear from Unit 2 switchgear. The feed to the intake structure busses is from the B and C transformers through a small section of overhead bus, then through their own duct banks into the intake structure switchgear room. Unit 1 Emergency Busses "H" and "J" are fed from "C" and "A" Reserve Station Transformers, respectively.

The 4160 Volt emergency switchgear is arranged in two separate systems designated H and J. The H bus is associated with train A, while the J bus is associated with train B. The buses are physically as well as electrically separated from each other on the bottom floor of the service building. The 480 Volt emergency switchgear buses H and J are powered from the corresponding 4160 Volt emergency bus. Therefore, the Unit 1 J 480 Volt bus receives electrical power from the Unit 1 J 4160 Volt bus MCCs 1J1-2N and 2S receive electrical power from the Unit 1 J 480 Volt bus.

5.0 Deterministic Analysis

This section is a summary of the information contained within Attachment 4, "Supporting Documentation: Deterministic Analysis" of this LAR. Additional details, including schematics of affected systems and complete lists of components to be de-energized are provided in Attachment 4.

5.1 Affected Emergency Safety Functions

Components associated with the MCCs 1J1-2N and 2S support the following emergency safety functions: Emergency Core Cooling, Auxiliary Feedwater, Containment Depressurization, and Containment Isolation. These emergency safety functions are common to a number of design basis accidents. The actions following a Steam Generator Tube Rupture are uniquely effected by de-energizing the MCCs 1J1-2N and 2S, and will therefore, be discussed in more depth.

Unless otherwise stated, the safety functions of the inoperable equipment listed can be accomplished by redundant trains, none of which rely on the MCCs 1J1-2N and 2S for power. In certain instances, actions will be taken either prior to de-energizing the MCCs or following a design basis accident, to assist in accident mitigation. The LCOs associated with these de-energized components are addressed in Section 5.2, "Limiting Conditions of Operation" below.

5.1.1 Emergency Core Cooling System

While MCCs 1J1-2N and 2S are out for maintenance, the following equipment associated with the Emergency Core Cooling System (ECCS) will be inoperable. For the following equipment, a redundant train can provide the safety function and no additional actions will be required. However, a precautionary measure to be taken to maintain the integrity of a LHSI pump will be discussed.

- "J" train of Safeguards ventilation exhaust
- "J" train of Boron Injection Tank heat tracing
- "J" train Boron Injection Tank flow path MOVs
- "J" train High Head Safety Injection MOVs
- "J" train Low Head Safety Injection MOVs

5.1.1.1 Low Head Safety Injection

Following an actuation of the ECCS, the "H" and "J" bus Low Head Safety Injection (LHSI) suction headers are swapped over from the Refueling Water Storage Tank (RWST) to the containment sump to provide long-term cooling. However, the "J" bus MOVs in the flow path between the RWST and containment will be de-energized and unable to realign. To prevent damage, the "J" bus LHSI pump (1-SI-P-1B) will be secured when the RWST level reaches 15%.

5.1.2 Auxiliary Feedwater

Four Auxiliary Feedwater (AFW) pump discharge MOVs (1-FW-MOV-100A, B, C, & D) will be de-energized when MCCs 1J1-2N and 2S are taken out for maintenance. Two of these valves (1-FW-MOV-100B & D) are in the flow path of two AFW pumps (1-FW-P-2 & 3B), which provide cooling to the A and B Steam Generators. Two of these valves (1-FW-MOV-100A & C) are normally isolated and, therefore, do not affect operability.

The two valves in the flow path of the pumps (1-FW-MOV-100B & D) are normally kept open and will remain open when de-energized. These MOVS are normally manually operated from the control room. They are considered to remain operable when de-energized because they can be manually operated locally, to throttle or isolate AFW flow to the Steam Generators, as needed. The flow path to the Steam Generators remains operable and fulfills the design function during this evolution.

Prior to beginning maintenance, an operator will be designated as the AFW valve operator and a sound-powered phone system will be utilized in the AFW pump house in order to provide direct communication with the control room. In the event of an AFW actuation, this operator will be deployed to the AFW pump house and will receive direct instruction from the Control Room regarding operation of these MOVs.

5.1.3 Containment Depressurization Actuation

In the event of a Containment Depressurization Actuation (CDA) the following equipment will be inoperable due to MCCs-1J1-2N and 2S being de-energized for maintenance. For the following equipment, a redundant train can support the safety function of these trains and no additional actions will be required.

- "J" train Quench Spray pump MOVs,
- "J" train of chemical addition flow path to Quench Spray.
- "J" train of Casing Cooling MOVs, which provides NPSH to "B" outside Recirculation Spray pump,
- "J" train Recirculation Spray pumps MOVs,
- "J" train of Service Water MOVs to Recirculation Spray heat exchangers, and
- "J" train of Service Water MOV to isolate Component Cooling heat exchangers.

5.1.3.1 Quench Spray

Under these conditions, one train of Quench Spray (QS) will be inoperable, as will one train of the Chemical Addition flow path into the system. The redundant train of QS and the other Chemical Addition flow path will continue to provide the safety-related function.

5.1.3.2 Recirculation Spray

The inoperable components in the Recirculation Spray (RS) System result in two sub-systems of one train of RS heat exchangers being inoperable. The redundant train of RS heat exchangers will continue to provide CDA heat removal.

5.1.3.3 Service Water

The inoperable MOVs in the Service Water (SW) System result in one loop of SW and one isolation MOV to the Component Cooling Water (CCW) heat exchangers being inoperable. The redundant loop of SW will provide necessary cooling and the redundant MOV will continue to provide isolation capability, if needed.

5.1.4 Containment Isolation

The "J" train Reactor Coolant Pump Seal Water Return containment isolation valve will be de-energized during this evolution (1-CH-MOV-1381). A redundant valve (1 CH-MOV-1380) is able to perform the safety function. Section 5.0, "Limiting Conditions of Operation," of Attachment 4, "Supporting Documentation: Deterministic Analysis" addresses this operability determination in more detail.

5.1.5 Steam Generator Tube Rupture

Two features are credited for post-accident mitigation of a Steam Generator Tube Rupture (SGTR) that are not credited in any other design basis accident: Pressurizer Power-Operated Relief Valves (PORVs) and Residual Heat Removal (RHR).

Pressurizer PORVs may be used to depressurize the RCS. However, when the MCCs 1J1-2N / 2S are de-energized, the block valve upstream of the "J" train PORV will be inoperable. The block valve (1-RC-MOV-1535) serves to isolate the "J" train PORV (1-RC-PCV-1456). The block valve will be de-energized open. With the block valve inoperable, LCO 3.4.11, Action D will be entered with a 72 hour Completion Time. This LCO also requires that the PORV be placed under "manual" operation. This manual operation will prevent cycling of the "J" train PORV so as to prevent a Small Break LOCA from occurring should the PORV fail to close after being automatically cycled open.

In the event of an SGTR, the redundant PORV (1-RC-PCV-1455C) will be used, if necessary, to depressurize the RCS. This valve and its block valve are powered by the "H" train, and will be unaffected when the MCCs 1J1-2N and 2S are de-energized. Figure 1 in Attachment 4 shows a simplified schematic of the pressurizer. The block valve to be de-energized and its associated PORV are labeled.

Additionally, the "J" train RHR suction MOV (1-RH-MOV-1701) will be de-energized. This valve is normally closed and will be de-energized closed. In the event of a SGTR, this MOV can be manually operated locally, if necessary, during the cooldown process. This valve is located inside of containment and is usually manually operated from the control room. Manual operation of the RHR suction MOVs is described in UFSAR Section 5.5.4.3.4.

5.2 Limiting Conditions of Operation

Forty-three of the 97 potentially de-energized loads would require entry into either a TS or Technical Requirements Manual (TRM) Limiting Condition for Operation (LCO), if they failed independent of de-energizing the MCCs 1J1-2N and 2S. When the MCCs 1J1-2N and 2S are de-energized, none of the LCOs will be entered. Instead, LCO 3.0.6[1] will be entered for all components listed. The only exception to this will be LCO 3.4.11 for the PORV block valve (1-RC-MOV-1535).

5.3 Summary

[1] LCO 3.0.6 states: "When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered" provided a satisfactory evaluation is performed in accordance with TS 5.5.14. This evaluation will be performed immediately prior to de-energizing the MCCs 1J1-2N and 2S.

Components associated with the MCCs 1J1-2N and 2S support the following emergency safety functions: Emergency Core Cooling, Auxiliary Feedwater, Containment Depressurization, and Containment Isolation. Unless otherwise stated, the safety functions of the inoperable equipment can be accomplished by redundant trains, none of which rely on the MCCs 1J1-2N and 2S for power. In certain instances, actions will be taken either prior to de-energizing the MCCs or following a design basis accident in order to ensure the successful actuation of these systems, as needed.

6.0 Probabilistic Risk Analysis

6.1 Risk Assessment

A risk-informed evaluation to determine the impact of the proposed change on plant risk was performed in accordance with Regulatory Guides 1.174 and 1.177.

The Tier 1 and Tier 2 results are discussed below. Tier 3 requirements ensure that the risk impact of all off normal configurations are assessed and managed as required by 10 CFR 50.65(a)(4).

The North Anna PRA model N105A [NB 01] was used for performing the quantitative evaluation of the delta risk impact due to the proposed one time extension. A discussion of the technical adequacy of the model to support this one time CT extension is provided in Section 3.4 of the PRA analysis contained in Attachment 5. The quantitative evaluation is supplemented by a qualitative assessment of the impact of the proposed change on the external events' contribution to the risk.

6.1.1 Method of Analysis and Results - Tier 1: PRA Capability and Insights

Regulatory Guide 1.174 [RG 01] and Regulatory Guide 1.177 [RG 02] are the applicable regulatory guides for preparation of the risk assessment.

Regulatory Guide 1.174 provides guidance on developing a risk-informed licensing submittal. This document classifies potential increases in Core Damage Frequency (ΔCDF) and Large Early Release Frequency (ΔLERF).

Regulatory Guide 1.177 provides additional guidance specific to risk-informed Technical Specification changes. This Regulatory Guide provides guidance on the acceptable Incremental Conditional Core Damage Probability (ICCDP) and Incremental Conditional Large, Early Release Probability (ICLERP). These risk metrics are the result when a risk increase, defined as the frequency of core damage or large radionuclide release per year, are integrated over the time of the proposed Technical Specifications Completion Time (CT). The thresholds for ICCDP and ICLERP in Regulatory Guide 1.177 are 5E-7 and 5E-8, respectively.

The effect on risk of the proposed increase in CT for restoration of the 1J1 MCC has been evaluated using the NRC's three-tier approach suggested in RG 1.177:

Tier 1 – PRA Capability and Insights,
Tier 2 - Avoidance of Risk-Significant Plant Configurations, and
Tier 3 - Risk-Informed Configuration Risk Management

Although RG 1.177 requires the evaluation of the proposed change on the total risk (i.e., on-line and shutdown risk), this evaluation only quantifies the on-line risk. This is appropriate since the shutdown risk will not be impacted as a result of the proposed change. For this one time extension, the maintenance will take place while the unit is on-line and not during shutdown.

In Tier 1, the impact of the proposed TS change on the figures of merit (CDF, ICCDP, LERF, and ICLERP) are assessed by considering (1) the PRA insights and findings and (2) the validity of the model.

<u>Risk Metrics</u>

ΔCDF_{AVE} = change in the annual average CDF due to an expected unavailability of MCCs 1J1-2N and 2S that could result from the increased CT. This risk metric is compared against the criteria of RG 1.174 to determine whether a change in CDF is regarded as risk significant. This metric is a function of the baseline annual average CDF, CDF_{BASE}.

$\Delta LERF_{AVE}$ = change in the annual average LERF due to an expected unavailability of MCCs 1J1-2N and 2S that could result from the increased CT. Similar to ΔCDF_{AVE}, RG 1.174 criterion was also applied to judge the significance of changes in this risk metric.

ICCDP = incremental conditional core damage probability with MCCs 1J1-2N and 2S out-of-service for an interval of time equal to the proposed CT (i.e., 72 hours). This risk metric is used as suggested in RG 1.177 to determine whether a proposed CT has an acceptable risk impact.

ICLERP = incremental conditional large early release probability with MCCs 1J1-2N and 2S out-of-service for an interval of time equal to the proposed CT. Similar to ICCDP, RG 1.177 criteria were also applied to judge the significance of changes in this risk metric.

The baseline Core Damage Frequency (CDF) and Large Early Release Frequency (LERF), from the N105A internal events and flooding, zero-maintenance model, are CDF = 4.49E-6/year and LERF = 7.83E-7/year, from the North Anna model notebook QU.2, Rev. 3, page 4.

The North Anna N105A model includes both internal events and flooding. The fire risks have been analyzed as well. The seismic and tornado risk contributions were evaluated and found to be negligible. These tables provide the combined CDFs and LERFs in detail.

Table 1 CDF and LERF Summary for Tier 1 Analysis								
	CDF (yr^{-1})				**LERF (yr^{-1})**			
	Internal Events & flooding	**Fire**	**Seismic & Tornado**	**Total**	**Internal Events & flooding**	**Fire**	**Seismic & Tornado**	**Total**
Baseline (zero maintenance model)	4.49E-6	3.91E-6	Not quantified	8.40E-6	7.83E-7	Not quantified	Not quantified	7.83E-7
1-EE-MCC-1J1 out of service	1.07E-5	1.0E-5	Not quantified	2.07E-5	1.57E-6	Base case + 4.11E-7	Not quantified	1.57E-6 + Base case fire + 4.11E-7
The base case fire risk was taken from the NAPS non-seismic IPEEE (p. 1-7). Fire LERF was not quantified. No seismic IPEEE was performed for North Anna and thus no numbers can be reported. The annual average tornado risk was screened out because of its low value.								

Table 2 Results of Tier 1 Analyses		
	Core Damage Risk	**Large Early Release Risk**
Internal Events Risk	CDF = 8.40E-6/yr***	LERF = 7.83E-7/yr*** + Fire LERF
Risk with 1-EE-MCC-1J1 OOS	CDF = 2.07E-5/yr	LERF = 1.57E-6/yr+Base case LERF + 4.11E-7/yr
Risk increase	CDF = 1.23E-5/yr	LERF = 7.9E-7/yr + 4.11E-7/yr = 1.20E-6/yr
Risk increase per CT entry (72 hours)	ICCDP = 1.01E-7	ICLERP = 9.86E-9
Annual Average Risk Increase with 1J1 OOS	CDF = 1.01E-7/yr	LERF = 9.86E-9/yr
* RG 1.177 classifies a change as "Small" when the ICCDP is < 5E-7 and the ICLERP is < 5E-8.		
** RG 1.174 classifies these changes as "Small" due to their combined baseline risk and expected risk increase.		
*** CDF values are Internal Events, Internal Flooding and Fire hazards. The LERF value is for the Internal Events, Internal Flooding, and Fire Hazards.		

There have been several previously approved, risk-informed Technical Specifications changes at North Anna. These changes and their cumulative risk impacts are tabled below, in addition to the currently proposed TS change.

Table 3
Summary of Approved or Pending NAPS Risk-Informed Technical Specifications Changes

Risk-informed TS Change	Reference	Annual CDF Increase	Annual LERF Increase
1J1 MCC 72-hour CT (one time only)	TSCR #N-075 (NAPS.RA.LI.8)	No permanent impact	No permanent impact
14-day underground fuel oil storage tank CT (one time only)	N-058 (NAPS.RA.LI.6)	No permanent impact	No permanent impact
RPS and ESF actuation system analog channel surveillance test internal extensions from monthly to quarterly and Completion Time extensions	TSCR #N-038 (ET NAF 98-0200, Rev. 0)	3E-07/yr (1% of baseline risk)	Not quantified
Supplemental RPS/ESFAS functions	TSCR #N-038 supplemental (SM-1317 Table 1 & SM-1290, Rev. 0)	3E-09/yr	3E-10/yr
7-day inverter Completion Time extension	TSCR #N-012 (SM-1360)	8.1E-08/yr	4.6E-10/yr
14-day Emergency Diesel Generator (EDG) CT	TSCR #318B (SM-0969, Rev. 0)	1.3E-06/yr	1.3E-07/yr *
14-day N2 backup supply for PORVs	TSCR #323 (ET NAF 95-0018, Rev. 0 & ET NAF 98-0202, Rev. 0)	Not quantified	Not quantified
Total		1.7E-6/yr	1.3E-07/yr

* Not quantified. LERF was conservatively estimated as 10% of the CDF change.

The cumulative ΔCDF of the proposed and previously approved TS changes is 1.7E-6/yr and the cumulative ΔLERF is 1.3E-7/yr. According to RG 1.177, the cumulative annual increases in CDF and LERF are still "small."

6.1.2 PRA Model Applicability and Quality

The latest PRA model, N105A, has been used to analyze the risk of the proposed change. The N105A model, which evaluates internal events and flooding, was released in March, 2007. The internal events (including the internal flooding hazard) PRA model is maintained and updated under the PRA configuration control program in accordance with Dominion procedures. Plant changes, including physical and procedural modifications as well as changes in performance data, are reviewed for applicability and the PRA is updated to reflect such changes on a regular schedule by qualified personnel, with independent reviews and approvals.

In order to verify and improve the quality of the North Anna PRA model, an independent review of the NAPS internal events at power was performed in 2001 by the Westinghouse Owner's Group (WOG). The peer review is documented in the

Westinghouse PRA peer review report [REPORT 05]. All of the "A" and "B" Findings and Observations (F&Os) have either been resolved or found to have no impact upon the proposed Technical Specification (TS) change. Documentation of the resolution of the B-significance F&Os is provided in PRA Notebook volume [NB 03].

Additionally, the NAPS PRA model was also subjected to a self assessment against Capability Category II requirements of the ASME Standard for PRA, including Addendum B [STD 03] and Regulatory Guide (RG) 1.200 Revision 1 [RG 03]. The review was conducted by a team of experts with experience in performing NEI PRA Certifications and ASME PRA Standard Reviews. The scope of this assessment was to compare the current PRA model against ASME standard RA-S-2002 (including RA-Sa-2003 and RA-Sb-2005) to determine if each of the requirements of Capability Category II had been met and sufficiently documented. The approach of the assessment was to develop a comprehensive list of all potential areas for improvement and to be aggressive in pursuing model enhancement by conservatively characterizing a Supporting Requirement (SR) as "Not Met" if one or more areas for improvement were identified. This conservative philosophy is different than that which is used for PRA model peer reviews that are performed in accordance with NEI 05-04, Revision 2, where "findings" and "suggestions" are used to characterize such observations. Using this conservative philosophy, although the preponderance of evidence point to meeting the applicable SR at Category II level, the assessment characterized a number of SRs as not meeting Capability Category II requirements. Based on a review of the findings and suggestions listed in the assessment many of the instances where a SR was indicated as "Not Met" could be characterized as a "suggestion" using the guidance in NEI 05-04.

To support the Bus 1J1 CT extension, consistent with the industry practices, the "Not Met" SRs were reviewed to:

- Identify those unmet SRs that do not have an impact on the risk insights provided in support of this application (e.g., documentation-only issues).
- Identify potential sensitivity studies that can be performed to ensure that the risk insights are not significantly affected by the "Not Met" findings.

As a result of this review, the following conclusions were reached:

i. A significant number of unmet SR issues pertained to documentation only. Enhancements to the documentation would not change the model and, therefore, would have no impact on the analysis performed in support of this application.

ii. A number of unmet SRs related to initiating event identification. For example, events related to the process used to identify plant systems that have the potential to cause an initiating event. However, although new initiating events may be identified, based on the experience of dealing with this comment, it is judged that 1) the accident progression for these potential initiating events is similar to the progression for initiating events already included in the model; and, 2) the frequency of these newly identified potential initiating events is lower than the existing initiating event frequencies. Therefore, the impact on this analysis is negligible.

iii. A number of additional unmet SRs pertained to the Accident Sequence (AS) element. One issue that resulted in characterizing an AS related SR as not meeting Capability Category II is that the basis for some system success criteria is not documented and that, as a result of developing the documentation, changes could occur. No expected changes or outliers were identified, so resolution of this item likely would not impact the analysis results. A few items related to the completeness of accident sequence modeling, but these items were for insignificant sequences, e.g., ATWS after a LOCA. The last item was that sources of uncertainty were not documented. Based on the discussion above, it is not expected that resolving the unmet SRs for the AS element with the potential for model changes would alter the findings of this analysis.

iv. The overall quality of a High Level Requirement (HLR) was found to be more than adequate for this one time application. For example, out of the 22 SRs related to AS HLR, one was found not to be applicable, 9 SRs were found to meet capability Category II, and 9 out of the 12 unmet SRs were found to be documentation related. One of the three unmet SR pertained to the uncertainty evaluation, and the other two unmet SRs pertain to specific examples of area of improvements and not necessarily indicative of a systemic problem.

v. A few unmet SRs were assessed to have no impact on the CDF/LERF estimate. For example, SC-B1 SR is characterized as "Not-Met" because, in reviewer's opinion, data used to develop the success criteria for seal LOCA and offsite power recovery appears to be dated. The "Not Met" characterization seems to be overly conservative because the reviewers also found that "the resulting success criteria seem to be reasonable as compared to those used in other plants." Additionally, the reviewers also stated that, "it is not certain if the fault tree models themselves may have more up-to-date models than the documentation indicates." Another example is DA-C12, where although unavailability data is based on plant-specific data and is documented in the appropriate notebook, the SR is characterized as unmet because it was not clear to the reviewers that both units' data was being used to compute unavailability. Also, the reviewers concluded that using a floor value of 1.0E-6, which was used for components that are not expected to be taken out of service, and had no observed unavailability, is too low. This particular issue does not have any impact on this one time CT extension since maintenance activities on all other risk significant components is prohibited.

vi. A number of SRs were characterized as not met due to the same apparent cause. For example, although the reviewers found that the intent of SY-A17 and SY-A19 were generally met, these SRs were characterized as not meeting capability Category II because load sequencing for the diesels is not included in the model, nor are there any assumptions or referenced calculations pertaining to load sequencing. The impact of this potential omission on this one time extension of CT for 1J1 MCC is judged to be negligible. Similar discussion also applies to SRs SY-A11 and SY-A13, where inadvertent SI actuation was judged to be inadequately modeled.

vii. Certain unmet SRs related to identification, screening, and modeling of pre-initiator operator errors. Numerous pre-initiator operator errors are included in the PRA model. Although a rigorous analysis of such events could result in the identification of additional items, pre-initiator operator errors are typically not important to the overall PRA results so it is not expected that resolving the unmet SRs for the pre-initiator HR element with the potential for model changes would alter the findings of this one-time CT extension. Additionally, any change in the risk estimation would impact the base case as well as the 1J1 case. Therefore, the overall risk-insights with respect to the change in risk for this one-time extension of the CT are judged to be unchanged.

viii. A number of unmet SRs related to post-initiator operator actions. None of these items noted any major weaknesses, so it is not expected that resolving the unmet SRs for the post-initiator HR element with the potential for model changes would alter the findings of this analysis. Additionally, similar to the pre-initiator SRs, any change on the risk estimation would impact the base case as well as the 1J1 case. Therefore, the overall risk-insights with respect to the change in risk for this one-time extension of the CT are judged to remain unchanged.

ix. Unmet items related to internal flooding are either due to documentation or have the potential to equally impact the base case and the 1J1 LAR case, with no impact on the delta calculations. Therefore, they are not expected to impact the risk insights for this proposed one-time extension of CT for 1J1 MCC.

In 2007, the N105A North Anna PRA model was successfully used to implement a one-time, 14-day Completion Time for the Underground Fuel Oil Storage Tanks.

Based on the discussion presented above, and the considerable effort made to incorporate the latest industry insights into the PRA as well as results of self-assessments and Peer Reviews, Dominion is confident that the current NAPS PRA model meets the expectations for PRA technical adequacy for this one-time CT extension.

6.1.3 Internal Events and Flooding Analysis

This analysis used the zero-maintenance N105A model, with 1-RC-MOV-1536 failed closed, as its base case. Appendix B in Attachment 5 provides additional information about modeling changes that were made to the base N105A model to perform the risk calculations.

The dominant accident sequences were reviewed for the case with the 1J1 480 VAC MCC unavailable. The results are as follows:

- The Steam Generator Tube Rupture (SGTR) initiating event contributes 64% (1.0E-6/yr) of the Large Early Release Frequency (LERF) with the 1J1 MCC unavailable. This configuration is LERF-limiting, rather than CDF-limiting, because the SGTR is a containment bypass event.

- The most limiting SGTR sequences include a failure to cool down and depressurize the RCS. This sequence contributes 21% (3.3E-7/yr) to LERF and is independent of the MCC outage.
- The next two most limiting SGTR sequences include failures of the High Head Safety Injection (HHSI) system and failure of feedwater isolation. These sequences total 21% (3.3E-7/yr) of LERF. The HHSI failures occurred due to the MCC outage on one train and coincident, random failures on the opposite train.
- Several SGTR sequences include failure of the Residual Heat Removal (RHR) system due to the tagout of the 1J1 MCC. This bus powers the normally-closed isolation valve 1-RH-MOV-1701 from the Reactor Coolant System to the RHR system inside containment. If recovery of the MOV is unsuccessful, then decay heat is removed via the secondary system and any secondary faults will result in an offsite release. These sequences total 13% (2.0E-7/yr) of LERF.

- The most limiting LERF sequence which is not a SGTR is the vessel rupture, contributing 18% (2.7E-7/yr) to LERF. This sequence is independent of the proposed MCC Completion Time.

- There are no other sequences involving the MCC tagout that contribute more than 5% to the overall LERF. The Core Damage Frequency (CDF) impact of the proposed CT is less limiting than its LERF impact.

6.1.4 Common Cause Issues

No common cause analysis has been performed. The original breaker failure has, to date, not revealed any characteristics of common cause concerns. The proposed CT is strictly a precautionary measure to inspect for potential damage to surrounding equipment. There is no evidence of potential common cause vulnerability and, thus, none has been modeled.

The current PRA model does not include any common cause faults in the emergency electrical power distribution system, other than those associated with the diesel generators.

6.1.5 Fire Analysis

[REPORT 02] documented the original IPEEE fire analysis for North Anna. It screened out all but four areas as insignificant contributors to core damage risk. The NAPS fire PRA model was developed using the following approach:

Fire areas of potential risk significance were identified using the initial qualitative and quantitative screening steps defined in the FIVE methodology document.

Those fire areas, which did not screen out, were subject to detailed modeling as described in various guidance documents (e.g.; NUREG-2300, NUREG-2815 or NSAC-

181). The COMPBRN IIIe code was used for all deterministic modeling of intra-area fire propagation. Inter-area fire propagation analysis was not required based on the review of the fire area boundaries performed to address the Fire Risk Scoping Study, NUREG/CR-5088 issues.

Fire frequencies in particular locations accounted for both generic experience (US plant experience obtained from the EPRI Fire Event Data Base) and area specific fixed ignition sources. The contribution of transient fuels and sources was accounted for by addressing plant specific procedures for the control of combustibles and ignition sources, as well as for periodic inspections for transients.

No credit was taken in the analysis for the detection and suppression of fires (i.e., fires were allowed to burn until they self extinguished).

Fire Risk Scoping Study Issues were addressed through specifically tailored walkdowns as defined in the FIVE methodology, including seismic fire interactions, effects of fire suppressant on safety-related equipment, fire barrier effectiveness and control systems interactions.

6.1.5.1 Approach for Assessing Change in Fire-Hazard-Induced Risk due to the proposed CT Extension

The current NAPS fire PRA model, which was developed in support of the Individual Plant Examination for External Events (IPEEE) study, is a vulnerability fire PRA model. Therefore, it contains a number of assumptions and assertions that are meant to effectively, yet, quickly identify areas of vulnerability. Also, the quantification part of the model used the average internal events model to quantify the risk of the postulated damage. In this application of the fire PRA model, to reflect the plant configuration during the proposed application, the assumptions and assertions of the IPEEE fire model are reviewed and adjusted, if necessary, and the case specific internal events PRA model (i.e., no-maintenance model with MCC 1J1 unavailable "N105A-TM0") was run to quantify the risk. The approach used in this evaluation is sometimes overly conservative. This conservative approach was used to ensure that the semi-qualitative nature of the assessment was well compensated for by the conservatisms included in other sections. Since this is not a vulnerability study (i.e., the relative risk is not important), the use of conservatisms in some areas to compensate for potential non-conservatisms in other parts is appropriate.

The following steps were followed to assess the potential change in the fire-hazard-induced risk due to the proposed CT extension:

a) The qualitatively and quantitatively screened fire areas were reviewed to assess whether the basis for screening would be changed due to unavailability of the 1J1 MCC.
b) Those screened fire areas that were found to have higher CCDP value were re-analyzed to determine their contribution to the CDF figure of merit, given the proposed plant configuration (i.e., the average base model was not used since this

is a one-time extension).

c) The contribution for those areas that were retained in the base case analysis for detailed analysis, were re-evaluated to assess any potential increase in CDF.
d) The Fire CDF so calculated was added to the internal events CDF to recalculate the impact on the CDF figure of merit.
e) The Delta CDFs so calculated, in combination with a qualitative evaluation of the containment performance, were used to quantify the increase in the LERF figure of merit.

6.1.5.2 Qualitative Screened Fire Areas

This screening was performed in several steps. In the first step, all plant areas which did not contain any susceptible Appendix R shutdown equipment (in any of their compartments) were screened out. Next, for the areas remaining, the requirement, or not, for a plant shutdown was determined assuming that 1) all Appendix R safe shutdown equipment and cables in a given area (including all compartments) are damaged and 2) the normal alternate shutdown path (as defined within the Appendix R framework) is unavailable. A demand for shutdown was assumed unless it could be shown with confidence that the fire would not cause an automatic trip or plant operating conditions or Technical Specifications would not require a shutdown within 8 hours.

If the fire does not create a demand for safe shutdown using the equipment assumed to be unavailable or damaged by the fire, then the fire area and all its compartments were screened out. (Note: It was not necessary to assume a loss of offsite power as in the case in Appendix R studies, unless there is some potential for the postulated fire inducing such an event.)

For this analysis, the second criterion (i.e., the criterion that the normal alternate shutdown path, as defined within the Appendix R framework, is unavailable) is adjusted as follows:

a) given the train supported by 1J1 MCC is unavailable and
b) the train supported by the H bus is available

Crediting Condition B (i.e., crediting availability of the train supported by the H bus) is judged to be appropriate for this application because prior to performing the work on the 1J1 MCC, the availability of the redundant train (i.e., the train supported by the 1H bus) will be verified.

It is also noted that in a later revision of the FIVE methodology [REPORT 13] fire areas or compartments should not be screened out unless it can be shown that Appendix R equipment is not damaged <u>and</u> there is no demand for shutdown. This revision was issued several months after the NAPS screening analysis was completed. For the original NAPS analysis the impact of the changes is that several fire areas eliminated in the initial qualitative screening analysis would require further evaluation. The IPEEE study, however, concluded that these fire areas would have been eliminated in the quantitative screening analysis phase. As a result, the number of areas requiring detailed analysis would not change. This conclusion was substantially verified in the fire analysis which was performed for Surry under the revised rules. For this analysis, since all the screened fire areas were re-evaluated, the original NAPS IPEEE qualitative screening criteria did not have a significant impact.

Based on a review of the Dominion calculation file, documenting the qualitative analysis [NB 05], the following evaluation has been performed:

- A number of fire areas were qualitatively screened out because they did not contain any safe shutdown equipment AND fire in these areas would not result in a forced plant shutdown. Such areas included Turbine Building Lube Oil Room (Fire compartment TB-LOR), Service Building East (Fire zone Z-34), Service Building West (Z-35), Service Building Stairwell (Zone 54), Technical Support Center (Zone 46A), and Auxiliary Building Boiler Room (Zone 22). For this application, it is concluded that the contribution of these areas to the fire-induced risk would not change since the safe shutdown components supported by 1J1 MCC would not be challenged.
- A number of fire areas were qualitatively screened out because they did not contain any Safe Shutdown components. However, a fire in these areas could result in a forced shutdown (manual/automatic trip or forced shutdown within 8 hours). These areas included Unit 1 Normal Switchgear Room (Fire area 5-1, fire compartment NSR-1) and Unit 1 Motor Generator Set house (Fire Area 3-1, Z-27-1, MGSH-1), where a loss of offsite power and a reactor trip initiating events can be postulated, respectively. For such areas, to calculate the change in the fire induced CDF, the fire frequency for the major fixed ignition sources in the area (i.e., fire frequency for electrical cabinets in case of fire area 5-1 and fire frequency for MG Set for fire area 3-1) is multiplied by the CCDP for the postulated initiating event with 1J1 MCC failure probability set to 1.0. Note that this is a conservative delta CDF calculation on the basis that the base case fire-induced CDF is assumed to be negligible. Also, disregarding the fire frequency from the other potential ignition sources in each area is justified since, the duration of the 1J1 MCC unavailability is limited and no significant maintenance activities will be performed.
- A number of fire areas were screened out because, although the area contained one or more safe shutdown components, the potential for a plant trip could be ruled out. Such area included Charging Pumps 1A, 1B, and 1C fire areas (fire area 11A, 11B, and 11C). In this application, the change in the fire risk due to the proposed CT extension is calculated as follows:

    - For areas where the affected safe shutdown components of concern are supported by the 1J bus, the change in risk is considered to be negligible since the 1J1 MCC outage would not add any additional risk (that is the equipment in these areas are assumed damaged by fire and availability or unavailability of the 1J1 MCC is considered to be irrelevant). In some cases (e.g., fire area 9B-1 (EDG 1J compartment), this consideration may be an oversimplification. However, this oversimplification is judged to be justified on the basis that the risk from fires in such areas is dominated by the fire-induced loss of safe shutdown components in the area. Additionally, based on a review of the final re-screening results, this potential oversimplification did not result in a change in the results, since the fire areas containing the H-supported train were screened out (e.g., the 1H EDG compartment remained screened out).

- For areas where the affected safe shutdown equipment of concern is supported by the 1H bus, the risk is calculated by multiplying the fire frequency by the appropriate CCDP, unless it could be clearly shown that the impact of the 1J1 MCC unavailability on the fire-induced risk for the area is negligible. If quantification was deemed to be necessary, then the fire frequency is set to be equal to the ignition frequency for the major fixed ignition sources in the area. The CCDP is the "no maintenance" CDF with guaranteed failure of the 1J1 MCC, including fire vulnerable components, and the appropriate initiating event frequency set to 1.0 and the frequencies for the other initiating events set to zero.

- A number of areas were qualitatively screened out on the basis that at least two alternate shutdown paths were available following a fire in these areas. For example, Emergency Diesel Generator (EDG) Room 1H was screened out since the offsite power and 1J EDG were unaffected by the fire. Another example is that a major part of the Turbine Building was screened out because all EDGs and at least two Motor Driven Auxiliary Feed Water pumps were unaffected by a postulated fire in this area. In this application, the change in the contribution of fires in such area is evaluated as follows:
  - For areas where the affected safe shutdown equipment of concern are supported by the 1J bus, the change in risk is considered to be negligible since the 1J1 MCC outage would not add any additional risk (that is the equipment in these areas are assumed damaged by fire and availability or unavailability of the 1J1 MCC is considered to be irrelevant). For example, the change in fire risk for the 1J EDG room is considered to be negligible. Again, this oversimplification is judged to be justified on the basis that the risk from fires in such areas is dominated by the fire-induced loss of safe shutdown components in the area. Additionally, based on a review of the final rescreening results, this potential oversimplification did not result in a change in the results, since the fire areas containing the H-supported train remained screened out (e.g., the 1H EDG compartment remained screened out).
  - For areas where the affected safe shutdown equipment of concern are supported by the 1H bus, the risk is calculated by multiplying the fire frequency for the major fixed ignition sources in the area with the CCDP for an appropriate surrogate initiating event. Note the CCDP is based on "no maintenance" CDF with the failure probabilities for the 1J1 MCC and fire vulnerable components in the area set to 1.0. For example the risk for the 1H EDG room is calculated by estimating the CCDP, assuming a forced manual shutdown event (using Turbine Trip as a surrogate) and failure probabilities of 1J1 MCC and 1H EDG set to 1.0. Then, the initiating event frequency is set equal to the fire initiating event frequency for the EDG ignition source. For the Turbine Building fire area, a loss of MFW event is postulated. The loss of MFW

CCDP, given guaranteed failure of the 1J1 MCC, is then multiplied by the fire initiation source frequency.

Note that for performing the above calculations, the internal events model is used (i.e., a new Fire PRA model has not developed). The major limitation of using the internal events model is that the human error probability (HEP) estimates used in the internal events model may not be appropriate for the fire scenario of concern. The major concerns with the appropriateness of HEP estimates, and the way these concerns are addressed in this evaluation, are as follows:

1) The fire event may present additional stress/distraction for operators that may not be present for a similar initiating event in the internal events model. To address this concern, the CCDPs values for the fire scenarios are increased by a factor of 2 or 5, depending on the severity of the postulated damage. Increasing the CCDP by a factor of 2 or 5 is conservative since an increase in failure probability of one operator action for one initiating event is not usually expected to increase the CCDP significantly.
2) The fire event may present plant physical conditions that may prevent operators from performing the credited operator action. To address this concern, a list of operator actions that in the internal events analysis are credited to be performed outside of the areas that were retained in the original fire PRA analysis for detailed evaluation (i.e., the control room, the emergency switchgear room, general portion of the Auxiliary Building, and the cable/vault tunnel) is obtained. The success of these operator actions would be set to zero (i.e., the failure probability will be set to 1.0), if the postulated fire event is evaluated to prevent the action from taking place.

It should be noted that in the IPEEE study, the circuits for the automatic actuation of components were not traced and as a result operator actions were credited to manipulate those valves that are required to change state in response to an initiating event.

The following information was used for evaluating the potential contribution of fires in each area to CDF.

<u>Turbine Trip Initiating Event</u>

From N105A-TM0 model -

T23 frequency = 9.18E-1/yr
CDF contribution given guaranteed failure of 1J1 MCC = 3.941E-7/yr
CCDP for internal events = 3.941E-7/9.18E-1 =4.3E-7

It is recognized that the above CCDP is based on the internal events model. As such failure probabilities of some operator actions (HEPs), credited in the internal events model, may be higher for a similar fire induced initiating event. However, since the postulated initiating event would not present a significantly different challenge to the operators, independent of its cause of occurrence, the HEP estimates would not increase significantly. Nevertheless, as a conservative measure, the CCDP value is increased by a factor of 2 to account for a potential additional fire-event-induced stress. Also, note that the fire-induced initiating event frequency assigned to each area is the total fire frequency for the significant fixed ignition sources in the area and no credit for the severity factor is applied. This is an additional conservatism.

Therefore, CCDP for fires = 8.6E-7

LOOP initiating Event

From N105A-TM0 model -
T1 frequency = 3.74E-2/yr
CDF contribution given guaranteed failure of 1J1 MCC = 4.615E-7/yr
CCDP for internal events = 4.615E-7/3.74E-2 = 1.2E-5

Similar to the Turbine Trip case above, it is recognized that the CCDP is based on the internal events model. As such failure probabilities of some operator actions (HEPs), credited in the internal events model, may be higher for a similar fire induced initiating event. Similarly, however, since the postulated initiating event would not present a significantly different challenge to the operators, independent of its cause of occurrence, the HEP estimates would not increase significantly. Nevertheless, as a conservative measure, the CCDP value is increased by a factor of 5 in this case to account for a potential additional fire-event-induced stress in combination with a LOOP event. Also, note that the fire-induced initiating event frequency assigned to each area is the total fire frequency for the significant fixed ignition sources in the area and no credit for the severity factor is applied. This is an additional conservatism.

Therefore, CCDP for fires = 6.0E-5

Results

Based on the approach shown above (which is a combination of the qualitative and quantitative screening), with the following plant configuration all but one of the IPEEE screened fire areas/compartments would screen out:

1. All accident mitigating systems and functions, other than those supported by the 1J1 MCC, are operable (i.e., the no-maintenance model is used)
2. 1J1 MCC is out of service

This is not unexpected since the contribution of 1J1 MCC being unavailable is well compensated for by restricting unavailability on all other accident mitigating components.

The general area of the Turbine Building is the only compartment that could not be screened out. The change in the contribution of this compartment to the fire risk due to 1J1 MCC being unavailable is estimated in the detailed analysis section of Attachment 5.

Note that based on a comparison of the local manual operator actions credited in the internal events model (Attachment F) and the postulated damage in each area, it was concluded that none of the postulated scenarios will require the failure probability of the local manual operator action to be set to 1.0.

6.1.5.3 Quantitatively Screened Fire Areas

Based on a review of NAPS calculation file, NAPS Fire Quantitative Screening, none of the fire areas retained after the qualitative analysis were quantitatively screened out.

6.1.5.3.1 Determination of Impact in the IPEEE Non-Screened Fire Areas

In the IPEEE study, detailed analysis was performed for four potentially significant fire areas which could not be eliminated as part of the qualitative and quantitative screening process. For these areas, IPEEE supporting documentation, which includes documentation of the changes that were made to reflect the postulated damage in each one of the defined compartments in each unscreened fire area, was reviewed. The information available in Section 6 includes the postulated initiating event and a list of basic events that were assigned certain failure probabilities (mostly 1.0). Based on a review of the basic event lists, the following criteria were used:

1. If the basic event representing the 1J1 MCC was assigned to fail as a consequence of a fire in a compartment, then the impact of the fire in that compartment was considered to be non-consequential on the proposed increase in the CT for 1J1 MCC.
2. If a basic event representing a component that supports the 1J1 MCC (e.g., 1J Bus) was assigned to fail as a consequence of a fire in a compartment, then the impact of the fire in that area was considered to be non-consequential on the proposed increase in the CT for 1J1 MCC.
3. If neither of the above two conditions applied, then ideally the IPEEE fire PRA model would have been solved twice. Once, with all the basic events representing the maintenance-induced unavailability of the component supported by the H bus set to zero and the second time, the above generated model with and 1J1 MCC failure probability set to zero. The first run would be made to ensure that a more representative delta is calculated, since for this evaluation, the operability of the redundant train will be verified prior to performing work on the 1J1 MCC. However, based a review of the IPEEE documentation and the state of the NAPS fire PRA model, it is judged that the best practical approach is to use the results of the IPEEE's detailed analysis to gain a conservative estimate of the risk increase.

Also, note that for the 1J1 MCC case, one additional area (general section of the Turbine Building) survived the qualitative screening. The contribution from this area is also further analyzed in this section but in this case, since the damage was limited, the current model was used to calculate the delta.

6.1.5.3.2 Determination of Delta CDF in the Cable Vault & Tunnel (CV&T)

The IPEEE study separated the CV&T into three compartments: (1) Tunnel which includes the area outside of the Emergency Switchgear Room (ESGR) labeled "cable vault" on most diagrams, (2) the Electrical Penetration Room (Elec Pen), which has also been called the cable vault area, and (3) Rod Drive Room.

Estimation of delta CDF

Based on a review of Table 4-1.3-1, "Fire Area/Compartment Safe Shutdown Equipment Detail Worksheet" for CV&T fire area, for an App. R fire, a number of Unit 2 components (e.g., Unit 2 Charging pumps) are relied upon to mitigate the consequences of a fire in this area. In the IPEEE fire PRA, this fire area was divided into a number of compartments and each compartment was separately analyzed. The analyses of impact of the 1J1 MCC unavailability on the risk for each compartment are discussed below.

Delta CDF for the Service Building CV&T Compartment

Based on a review of page 61 of NAPS calculation file 5T45.NF/08.1, this subcompartment does not contain significant number of components supported by the 1J1 MCC. Therefore, Condition 3 of the above set of criteria applies. Two runs were made. The CCDP estimates for the base case, with 1J1 MCC available, and the 1J1 case, with the 1J1 MCC assumed unavailable, are estimated to be 2.26E-7 and 4.29E-7, respectively. Again, since the internal events PRA model is used for this evaluation, the CCDP values are multiplied by a factor of 5 to account for potential additional operator stress by the fire in this compartment. Therefore, the base case and 1J1 MCC case CCDPs are 1.13E-6 and 2.15E-5, respectively and the delta CCDP is 1.01E-5. The initiating event frequency for this fire scenario is 1.56E-4 per year. Therefore, the change in CDF is 1.58E-9 per year.

Delta CDF for the Electrical Penetration Compartment

Based on a review of page 69 of NAPS calculation file 5T45.NF/08.1 in the IPEEE study, Condition 1 of the above set of rules applies (i.e., 1J1 MCC would be affected in this subcompartment). Therefore, there is no delta CDF contribution from this subcompartment.

Delta CDF for the Rod Drive Room Compartment

This compartment is physically located above the electrical penetration area. The compartment contains two 480V electrical buses 1H1 and 1J1. For the IPEEE study, this compartment was further subdivided into Control Rod Drive Room General Area (GA), Control Rod Drive Room 1H1 Bus area (1H1), Control Rod Drive Room 1J1 Bus area (1J1). The evaluation for each subcompartment is provided below:

- In the IPEEE study, the fires initiating in the general area do not contribute significantly to the CDF due to limited impact of the postulated fires on accident mitigating functions. For this evaluation, it is also concluded that the fires in the GA will not have a significant impact on the proposed CT extension on the basis of low fire frequency (1.9E-3 per year) and limited damage.
- In the IPEEE study, fires originating in the 1J1 subcompartment were the most significant contributors to the CDF. For this analysis, the unavailability of the 1J1 MCC would not have an impact on the delta CDF.
- In the IPEEE study, fires originating in the 1H1 subcompartment were the second most significant contributors to the CDF. Based on a review of the event trees for this area, the fire frequency for this subcompartment is 2.86E-4 per year and its contribution to CDF is 2.41E-7, which means the IPEEE CCDP for this area is 2.41E-7/2.86E-4 or 8.6E-4. Based on a review of page 85 of 5T45.NF/08.1,

MCC 1-EE-1J1-2 is included the components that are affected by a fire in this subcompartment. Also, based on a review of Table 6.4-1 and 6.4-2 of 5T45.NF/08.1, as well as Table 4-1.3-2 of 5T45.NF2, the following observations are made:

- o The major non-fire-induced failures which contribute to the CDF are loss of AFW and MFW functions.
- o Random failures or maintenance-induced unavailability of the TDAFW pump (AFW-P-2) and the 3B MDAFW pump are included amongst the top cutsets for fires originating in this subcompartment.
- o The Appendix R credits Unit 2 high head safety injection function in this area and this function is not affected by a fire in this subcompartment.

Based on these observations, it is concluded that the inoperability of the 1J1 MCC would not result in a significant increase in the conditional probability of core damage sequences in this area on the following bases:

- The availability/reliability of the cooling function provided by the TDAFW or MDAFW pump 3B are not significantly impacted by the unavailability of the 1J1 MCC because neither the pumps nor the flow path are affected. The pumps are not powered by the MCC. The MOVs on the flow path are normally open, kept open, and will be open and de-energized. Also, these valves can manually be operated locally, thus providing or isolating flow to the SGs (See deterministic evaluation, Table 1, note 2). Also, note that an Appendix R fire in the CV&T area is postulated to result in a loss of 1J1 MCC. Therefore, procedures are in place for this area to manually operate these valves in an event of fire in the CV&T area.
- The availability/reliability of the high head safety injection function is not significantly impacted because for fires in this subcompartment, the Unit 2 HHSI is credited. Again, as noted in the first bullet, an Appendix R fire in the CV&T area is postulated to result in a loss of 1J1 MCC. Therefore, the safe shutdown for this area already includes an evaluation of the HHSI flow path due to the unavailability of the 1J1 MCC. Although the fire PRA scenarios may assume a more limited damage and as such the 1J1 MCC unavailability may have an impact on the CDF estimate, it is judged that such scenarios are a subset of all events. Therefore, the CCDP multiplier and the verification of the availability of the alternate path compensates for any optimistic consideration with respect to the reliability of any manual action that may be credited for the HHSI flow path.

Based on the above observations, discussions and conclusions, it is asserted that the additional impact of the 1J1 MCC unavailability on the consequences of a fire in this subcompartment is limited. However, as a bounding analysis, it is conservatively assumed that the consequences can be represented by increasing the IPEEE CCDP estimate (calculated above as 8.6E-4) by a factor of 5. Therefore, the CCDP, CDF, and delta CDF for this subcompartment, with 1J1 MCC unavailable are estimated as 4.3E-3, 1.23E-6 per year, and 9.9E-7 per year, respectively.

Total Delta CDF
The total delta CDF is dominated by the delta CDF in the Control Rod Drive Room 1H1 Bus area (1H1), which is conservatively estimated as 9.9E-7 per year. It should be additionally noted that due to the work being done (which results in the 1J1 MCC series being de-energized) as well as fire watch compensatory measures which will be deployed in this area, the fire initiation frequency is smaller than that used in the IPEEE study.

6.1.5.3.3 Determination of Delta CDF in the ESGR

In the IPEEE study the switchgear room was divided into the following four compartments:

The 1H Switchgear Room (1H SWGR)- This room contains Emergency Switchgear, transformers, motor control centers, and cable race ways associated with Train H; battery Room 1-II is also located in this room. In addition there are minimal set of cable race ways (conduits only) associated with Train J, which are referred to as “crossover” raceways.

The 1J Switchgear Room (1J SWGR)- This compartment contains Emergency Switchgear, transformers, motor control centers, and cable race ways associated with Train J; battery Room 1-IV is also located in this room. In addition there are minimal set of cable race ways (conduits only) associated with Train H, which are referred to as “crossover” raceways.

The Instrument and Relay Rack Room (IRR)- This compartment contains solid state protection and relay logic cabinets associated with both Train H and Train J. The area also contains overhead train A and B cable trays and BOP under floor cable chases. The IRR also contains all process instrumentation cables which are used to monitor plant parameters and trip or ESF conditions.

The AC Room- This compartment contains the ESGR air conditioning units and related cables.

Based on the above descriptions, it is concluded that the proposed 1J1 MCC CT extension request has no impact on the risk originating from the fires originating in the 1J SWGR compartment.

Estimation of Delta CDF
Based on a review of calculation file 5T45.NF/08.2, page 266, in the IPEEE study it was estimated that the Emergency Switchgear Room has a fire-induced core damage frequency of 3.26E-6 per year. This represented 84% of the total Unit 1 fire CDF. The IRR alone contributes 63% of the total Unit 1 fire CDF. This result is not unexpected since almost all of the power station's equipment has a cable inside of this fire area. It should be noted that the Appendix R relies on the alternate safe shutdown capability using equipment outside this area. Therefore, although the fire risk in this area is relatively high, the 1J1 MCC availability does not have a significant impact on the fire-induced CCDP since most fires would either disable both trains or the 1J bus related power or cables.

Also, the IPEEE study concluded that reliability of the secondary heat removal function was the most important function in the ESGR CDF sequences. Recovery of AFW or MFW included actions such as removing control circuit fuses from the MFW or AFW 4160 V breakers cubicles.

Based on a review of the top sequences for the ESGR fires in the IPEEE study (pages 267-279), description of these sequences provided on pages 280-291, and the credited components provided in Table 4-1.6-1 of 5T45.NF2, page 42-45, the following observations are made:

1. Consistent with the IPEEE conclusions, fire induced reduction in the reliability of the AFW function is a major contributor to the CDF.
2. In sequences ranked 1st through 7th, and the 9th ranked sequence, either both MDAFW pumps or the 3B AFW pump are assumed to be damaged and core damage results. These sequences (all postulated as a result of a fire in the IRR subcompartment) contribute 47.2% of the total fire CDF from the ESGR fire area.
3. The feed and bleed function, using HHSI system (which may be credited as a backup to the loss of AFW function) can be provided by the Unit 2 HHSI system.
4. Lower ranked sequences in the IRR room (e.g., Sequences 68 and 69) also include fire damage to both MDAFW pumps or damage to the B MDAFW pump.
5. A number of lower ranked sequences (e.g., 70th, 66th, 65th) are postulated to occur in the 1J SWGR subcompartment.
6. The availability/reliability of the cooling function provided by the TDAFW or MDAFW pump 3B is not significantly impacted by the unavailability of the 1J1 MCC because neither the pumps nor the flow path are affected. The pumps are not powered by the MCC. The MOVs on the flow path are normally open, kept open, and will be open and de-energized. Also, these valves can manually be operated locally, thus providing or isolating flow to the SGs (See deterministic evaluation, Table 1, note 2).

Based on the above, the following conclusions are made:

1. Based on observations 1-3, and 6 above, the frequencies of the high ranked sequences from the IPEEE study will not significantly increase due to the unavailability of the 1J1 MCC because the 1J1 MCC would not have an impact on reliability of the AFW function or HHSI function as modeled for fires in the ESGR fire area.
2. Based on observations 4, 5, and 6, the frequencies of the lower ranked sequences from the IPEEE study will also not significantly increase due to the inoperability of the 1J1 MCC because the 1J1 MCC would not have an impact on reliability of the AFW function or HHSI function as modeled for fires in the ESGR fire area.

Therefore, it is judged that the increase in the ESGR fire risk, due to the unavailability of the 1J1 MCC, is minimal because either the postulated fire damage in the area would disable the function provided by the MCC, or the unavailability of the MCC does not change reliability of the functions credited to mitigate the consequences of a fire in this fire area.

Total Delta CDF

To ensure that any potential impact is included, a delta CDF is conservatively estimated as follows:

The total base case fire CDF from the ESGR room = 3.26E-6 per year
Total fire frequency is 1.27E-2 per year

An average CCDP can be approximated as 3.26E-6/1.27E-2= 2.57E-4 (it is noted that this is not really a CCDP but is used as a surrogate for an average of all the CCDPs)

As a conservative measure, due to the unavailability of 1J1 MCC, the average CCDP is increased by a factor of 2 (this is highly conservative because fires initiating in the 1J1 MCC are the highest contributors to the fire risk in this area). Therefore, delta CDF for this subcompartment, with 1J1 MCC unavailable is estimated as 3.26E-6 per year.

6.1.5.3.4 Determination of Delta CDF in the General Aux Building Fire Compartment

The Auxiliary Building is a four-level structure constructed of reinforced concrete with metal siding used for the upper levels. This fire area houses both normal operating and emergency components, in particular cable and equipment for the Component Cooling Water (CCW) and Chemical Volume Control Systems (CVCS).

In the IPEEE study, the potential fires and equipment damage in the Auxiliary Building were analyzed. The majority of the fire scenarios were determined to cause only limited damage, except for fires originating from the CCW pumps.

In the base fire PRA model, the Auxiliary Building contributes less than 1% to the Unit 1 total fire CDF. The CDF for this area is not higher because, the IPEEE model credited Operator action for manipulating MOVs and manual valves, after the postulated fire was extinguished. Human error probabilities (HEPs) for manual local operation of valves were included in the fire PRA model. Since the availability of the 1J1 MCC only impacts remote capability function of supported components and, as stated here, remote operation of valves was not credited in the fire PRA model, the unavailability of the 1J1 MCC is not expected to result in a change in the estimated risk for this area.

Also, based on a review of the top sequences for the Auxiliary Building, it is noted that the reliability of the components credited for the mitigation of the postulated fire scenarios (for example, Unit 2 Charging System, Unit 2 CCW, Unit 1 CCW, AFW system) is not impacted by the 1J1 unavailability.

Therefore, based on the above evaluation and the low contribution of this area to the total Unit 1 fire CDF, it is concluded that the proposed extension of the 1J1 MCC CT will not impact the fire risk in this area.

6.1.5.3.5 Determination of Delta CDF in the Control Room

Fire scenarios which were developed in the IPEEE study for the Control Room mainly postulated a loss of control or indication provided by the control room cabinets. Typically, for most functions, the control and indications for the redundant trains are provided in close proximity of each other and are postulated to be impacted by the same fire scenario. Also, for more severe fires, the control room is expected to be evacuated and the plant to be shutdown from the Auxiliary shutdown panel, located in the Emergency Switchgear Room. An Auxiliary Monitoring Panel is also provided in the Fuel Handling Building. Emergency Diesel Generators are provided with local control panels and Component Cooling, Service Water, and RHR pumps can be operated from the Switchgear. Additionally, a number of local operator actions, such as manipulation of MOVs are required.

In the IPEEE fire PRA study, the Main Control Room fire contributes 4% to the Unit 1 total fire CDF.

Based on a review of fire scenarios delineated on pages 11 through 21 of calculation file 5T45.NF/08.4 as well as a review of fire PRA results provided on pages 119 through 129 of the same calculation file, it is concluded that the impact of the 1J1 MCC unavailability on the conditional probability of core damage for the control room fire scenarios is negligible on the basis that:

1. The control and indication for the redundant components are mostly affected by the same fire.
2. The fire PRA model credits manual local operation of a number of valves, including those that are powered by the 1J1 MCC (e.g., See a list of operator actions that are required in Auxiliary Building per Appendix A of 5T45.NF/08.4). This list was generated based on plant procedure 0-FCA-1.
3. The cooling provided by the AFW system is important in most fire scenarios. Due to the configuration of the AFW valves which are supported by the 1J1 MCC (as discussed previously), the reliability of this function is not significantly affected by 1J1 MCC unavailability.

Therefore, no delta CDF is calculated for the fires originating in the Control Room.

6.1.5.3.6 Determination of Delta CDF in the General Area of Turbine Building (TB)

The IPEEE study screened out this area based on the FIVE methodology screening rules. In this analysis, based on the analysis described in Section 3.7.1.1, this fire compartment is retained for further evaluation. The delta CDF for this fire area is conservatively calculated as follows:

Initiating Event = T1 (Loss of Offsite Power)
Fire Frequency = 2.6E-2 per year (similar to the 1J1 case)
CCDP given guaranteed failure of MDAFW 3A (See Run MCC 1J1(A)-T1-AFW-3A-FS.EQP)= 1.22E-4 (using N105A-TM model) * 5 = 6.1E-4

CDF (base) = 2.6E-2 * 6.1E-4 = 1.6E-5 per year
Delta CDF = 1.8E-5 (from the screening table) – 1.6E-5 = 2.0E-6 per year

This is conservative due to the use of a high fire frequency, postulation of loss of offsite power for the given frequency, and due to the postulated damage for the assigned fire frequency.

6.1.5.4 Total increase in Fire CDF Due to the Unavailability of 1J1 MCC

Based on the results of the assessment described in Sections 3.7.1.1, 3.7.1.2, and 3.7.1.3 in the IPEEE study, the total increase in CDF is:

Total increase = Delta CDF from the Turbine Building Fires + Delta CDF from the CV&T Fire + Delta CDF from the ESGR Fires = 2.0E-6 + 9.9E-7 +3.26E-6 = 6.25E-6 per year. As shown in Table 4.1-1, the base case fire CDF is 3.91E-6 per year. This evaluation conservatively estimates an increase of 6.25E-6 per year, resulting in a total CDF of 1.0E-5 per year.

Note that, for most areas, the qualitative assessment of the unscreened fire areas (see Sections 3.7.1.1, 3.7.1.2, etc of this document), indicates that the impact of the 1J1 MCC being out of service on the fire-induced CDF for a particular compartment is negligible. However, the risk calculation, intentionally, uses a very conservative approach to provide a quantitative estimate of the CDF increase. As a result, the estimated CDF increase is about factor of two greater than the base case fire CDF (That is, a fire CDF with 1J1 being out of service is more than two times higher than the base case fire CDF). Given the components that would be impacted by 1J1 MCC being out of service, this is highly conservative. This conservative approach was used to ensure that the internal events model or the qualitative nature of the assessment was well compensated for by the conservatism included in other sections. Since this is not a vulnerability study (i.e., the relative risk is not important), then the use of conservatisms in some area to compensate for potential non-conservatism in other parts is appropriate. Some of the conservatisms in the modeling include:

- In most cases (from the quantification point of view), other than use of the no-maintenance model in the screening stage, no credit is taken for the fact that this is a one-time extension when the work will be performed with plant configuration known and certain compensatory measures in place.
- From the fire hazard point of view and postulated initiators, the availability/reliability of the secondary heat removal function is an important factor. The availability/reliability of the cooling function provided by components supported by the 1J1-MCC (i.e., the TDAFW or MDAFW pump 3B) are not significantly impacted by the unavailability of the 1J1 MCC because the MOVs on the flow path are normally open, kept open, and will be open and de-energized. Also, these valves can be manually operated locally, thus providing or isolating flow to the SGs.
- Similar to the internal flooding hazard, due to the plant physical configuration and location of safe shutdown equipment and their associated cables, the significant

contributors to the internal fire risk include those scenarios that have the potential to disable redundant components performing the same function. Therefore, the impact of unavailability of one train is not as consequential.

6.1.6 Seismic Hazard Analysis

The North Anna IPEEE did not perform seismic risk calculations. The North Anna IPEEE used a seismic margins evaluation based upon a Review Level Earthquake (RLE) with a peak ground acceleration of 0.3g [REPORT 03]. The frequency of this event may be estimated from the EPRI Mean Seismic Hazard Curve for Surry.

The seismic CDF associated with a 1J1 bus tagout may be evaluated as follows. A severe seismic event is likely to produce a long-term LOOP that is not readily recovered, due to potential severe damage in the switchyard. Therefore, it is similar to a LOOP event due to other causes, with little or no opportunity for recovery. Based on a review of the internal events LOOP cutsets and results for the base case configuration (MCCJ1(A)-T1.EQP) and the 1J1 configuration (MCC1J1-T1.EQP, it is noted that:

- The unavailability of the 1J1 MCC increases the LOOP's CCDP by 7.5E-6. This is very small increase in the CCDP.
- The new cutsets appearing in the top 10 cutsets are where a component in the opposite train fails to perform its function. This is conservative because for the purpose of this one-time extension, the operability of the opposite train will be verified.

Since the seismic frequency is only 4.9E-5/yr (REPORT 04, p. A-10, *mean* value at 0.3 g = 300 cm/sec^2), it is concluded that the combination of potential seismic events, seismically-induced loss of other components, and a concurrent 1J1 bus outage will not contribute significantly to the cutsets in this analysis for either CDF or LERF. It is recognized that during a seismic event other failures may occur, resulting in higher LOOP CCDP. However, given the current state-of-the-art in seismic analysis, a seismic event is postulated to disable redundant components. As a result, although the CCDP value may increase, the delta CCDP is not expected to increase significantly.

Also, similar to the fire hazard, it is judged that the risk insights for this one-time extension are not impacted by this qualitative assessment of the seismic hazard, on the basis that:

1) There is a significant margin in the calculated figures of merit.
2) The expected out of service time of the MCC is about 52 hours.
3) The most significant contributor to the risk (e.g., the reduction in the capability to use RHR for the decay heat removal), is not required for most seismic induced plant transients.
4) Given the current state of knowledge in seismic PRA, where redundant components are assumed failed for a given seismic level, the risk estimate due to the unavailability of one redundant MCC would not be appreciably higher.

6.1.7 Tornado Hazard Analysis

A tornado strike is likely to produce a long-term LOOP that is not readily recovered, due to potential severe damage in the switchyard. Therefore, it is similar to a LOOP event due to other causes, with little or no opportunity for recovery. As stated in Section 3.8 in the IPEEE study, the inoperability of the 1J1 MCC is estimated to result in an increase of 7.5E-6 in the LOOP's CCDP. Since the tornado frequency is only 1.94E-4/yr (IPEEE, p. 5-8) and the fact that the unavailability of the 1J1 MCC would not have a significant impact on the operability of the functions credited for mitigating consequences of a tornado event (e.g., AFW function and the emergency power function (e.g., Emergency Diesel generators), it is concluded that, the change in risk will be insignificant.

6.1.8 Tier 2: Avoidance of Risk-Significant Plant Configurations

There is reasonable assurance that risk-significant plant equipment configurations will not occur when 1J1 MCC is out of service using the proposed TS change based on the following:

Technical Specifications and Safety Function Determination Program

Adhering to the current TS requirements will prevent many of the more risk significant configurations from being entered into. Specifically, there are requirements concerning the operability/availability of emergency buses as specified in LCO 3.8.9. Potential configurations that should be avoided while the MCC is out of service are the inoperability of the redundant emergency buses and components supported by the H bus.

The Safety Function Determination Program (SFDP) requires provisions for cross-division checks to ensure a loss of the capability, to perform a safety function assumed in the accident analysis, does not go undetected. TS LCO 3.0.6 establishes requirements regarding supported systems when support systems are found inoperable. Upon entry into TS LCO 3.0.6 an evaluation is required to determine whether there has been a loss of safety function. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of TS LCO 3.0.6. Procedure O-GOP-9.4 implements the SFDP.

Risk Management and Compensatory Actions

The risk associated with having 1J1 MCC de-energized will be managed by adhering to the requirements for online risk assessment and management as described in the Dominion procedure NF-AA-PRA-370. In addition to the risk directly associated with the MCC unavailability, the procedure requires that potentially risk significant configurations during the period of the MCC's inoperability are assessed and managed. Based on a review of dominant cutsets, these additional risk management actions and restrictions which will be used during the extended completion time include the following:

- Planned maintenance that may result in the unavailability of other equipment within the scope of Maintenance Rule (a)(4) program will be prohibited. For example:
    - There will be no planned maintenance on either units' Emergency Diesel Generators, the Unit 1 "H" emergency bus, the "F" transfer bus, the switchyard, the Alternate AC Diesel Generator, or the reserve station service transformers.
    - Protected equipment will be identified and signs posted, including but not limited to: Unit 1 "H" emergency bus, the "F" transfer bus, the Alternate AC Diesel Generator (AAC DG), and the Unit 1 "H" Emergency Diesel Generator.
    - The two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System will be maintained operable.
- AFW pump 3A will be verified operable and signs posted.
- The MOVs on the TDAFW and MDAFW 3B's flow path will be verified to be open, kept open, and de-energized.
- An Operator will be designated as the AFW valve Operator and will be deployed to manually operate the de-energized AFW MOVs, if needed.
- A sound-powered phone system will be installed for communication between the AFW pump house and the Control Room.
- A Maintenance Operating Procedure (MOP) will be utilized to control the removal of MCCs 1J1-2N and 2S electrical loads, to ensure compliance with Technical Specifications, and to verify that safety function is maintained throughout the extended 72 hour Completion Time. This includes posting of signs for protected equipment.
- A contingency plan will be added to applicable Electrical Maintenance procedures to expedite re-energizing the MCCs, if needed.
- Risk awareness briefings will be conducted for maintenance and operations personnel prior to the work.
- Maintenance will be performed around-the-clock to minimize the time spent with equipment unavailable.
- Unit 2 charging pumps and both Auxiliary Service Water pumps will be verified operable.
- Work that may cause a trip hazard (e.g., surveillance in instrument racks) will not be performed.
- An Operations Standing Order will require an operator action to manually open 1-RH-MOV-1701 following a Steam Generator Tube Rupture, if needed.
- Pressurizer Operated Relief Valve (PORV) 1-RC-PCV-1456 will be placed in manual operation to prevent the possibility of a Small Break LOCA should the PORV cycle automatically and fail to re-seat while the PORV block valve (1-RC-MOV-1535) is de-energized open.
- Fire watches will be established in the Cable Vault and Tunnel area and the Service Water Pump House (Fire Area 12).

6.1.9 Tier 3: Risk-Informed Plant Configuration Control and Management

Dominion's 10 CFR 50.65(a)(4) program fully satisfies the requirements of Regulatory Guide 1.177 Tier 3. RG 1.177, Section 2.3, states that "The licensee should develop a program that ensures that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity. A viable program would be one that is able to uncover risk-significant plant equipment outage configurations in a timely manner during normal plant operation."

The Dominion (a)(4) program performs full PRA analyses of all planned maintenance configurations in advance. Configurations that approach or exceed the NUMARC 93-01 risk limits (i.e., 1.0E-6 for CDP) are avoided or addressed by compensatory measures. Historically, North Anna rarely approaches these limits. Emergent configurations are identified and analyzed by the on-shift staff for prompt determination of whether risk management actions are needed. The configuration analysis and risk management processes are fully proceduralized in compliance with the requirements of (a)(4).

North Anna's 10 CFR 50.65(a)(4) compliance program requires analysis and management of all configuration risks. The emergency power system is included in the (a)(4) scope and any component unavailability is monitored, analyzed and managed. When a configuration approaches the (a)(4) risk limits, plant procedures direct the implementation of risk management actions in compliance with the regulation. If the configuration is planned, these steps are taken in advance.

The proposed 1J1 bus outage is not expected to approach the required risk management thresholds of the (a)(4) regulation. While combinations of unavailable equipment and/or evolutions, including a 1J1 bus outage, may approach the limits and even require risk management actions, the risks arising from these configurations will be dominated by factors other than the MCC. As a result, the risk significance of the proposed MCC-1J1 tagout does not warrant limitations upon other equipment. Nevertheless, this analysis has assumed that no concurrent, planned maintenance will be performed and the analysis will be invalidated otherwise. This limitation is applicable to planned maintenance only, and not emergent. (This analysis has accounted for the closed pressurizer PORV block valve.)

6.2 Defense-In-Depth Assessment

The proposed change to the CT maintains the system redundancy, independence, and diversity commensurate with the expected challenges to system operation. The opposite train of emergency power and the associated engineered safety equipment remain operable to mitigate the consequences of any previously analyzed accident. In addition to the Technical Specifications, the Work Management Program, and Maintenance Rule (a)(4) Program provide for controls and assessments to preclude the possibility of simultaneous outages of redundant trains and ensure system reliability. The proposed increase in the CT for the "J" 480 Volt AC distribution subsystem (MCCs 1J1-2N and 2S) will not alter the assumptions relative to the causes or mitigation of an accident.

The proposed change meets the defense-in-depth principle consisting of a number of elements. These elements and the impact of the proposed change on each follow:

- **A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved.**

  The proposed CT change has only a small calculated impact on CDF and LERF. The proposed change is not accomplished by degrading core damage prevention and compensating with improved containment integrity nor do this change degrade containment integrity and compensate with improved core damage prevention. The balance between prevention of core damage and prevention of containment failure is maintained. Consequence mitigation remains unaffected by the proposed changes. Furthermore, no new accident or transients are introduced with the requested change and the likelihood of most accidents or transients is not impacted.

- **Over-reliance on programmatic activities to compensate for weaknesses in plant design.**

  Plant safety systems are designed with redundancy so when one train is inoperable, a redundant train can provide the necessary design function. During the timeframe when the MCC 1J1 is inoperable, a redundant safety train will be maintained operable. PRA analysis indicates that there is a small calculated impact on CDF and LERF with the proposed TS change.

- **System redundancy, independence, and diversity are maintained commensurate with the expected frequency and consequences of challenges to the system.**

  The redundancy, independence, and diversity of the electrical distribution subsystem will be maintained during the extended 72-hour CT with the exception of the MCCs 1J1-2N and 2S. A Maintenance Operating Procedure (MOP) will be utilized to control the removal of MCCs 1J1-2N and 2S electrical loads, to ensure compliance with Technical Specifications, and to verify that safety function is maintained throughout the extended 72-hour CT. During the extended CT the two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System will be maintained operable.

- **Defenses against potential common cause failures are maintained and the potential for introduction of new common cause failure mechanisms is assessed.**

  Defenses against common cause failures are maintained. The extended CT requested is not sufficiently long to expect new common cause failure mechanisms to arise. In addition, the operating environment and operating parameters for the emergency bus switchgear remains constant, therefore, new common cause failures modes are not expected. In addition, redundant and backup systems are not impacted by this change and no new common cause links between the primary and backup systems are introduced. Therefore, no new potential common cause failure mechanisms have been introduced by the proposed change.

- **Independence of barriers is not degraded.**

  The barriers protecting the public and the independence of these barriers are maintained. Multiple systems or electrical distribution systems will not be taken out of service simultaneously that could lead to degradation of these barriers and an increase in risk to the public. In addition, the extended CT does not provide a mechanism that degrades the independence of the barriers; fuel cladding, reactor coolant system, and containment.

- **Defenses against human errors are maintained.**

  The 1J1 480 Volt AC distribution subsystem (MCCs 1J1-2N and 2S) powers the normally closed isolation valve 1-RH-MOV-1701 from the Reactor Coolant System to the RHR system inside containment. The calculation performed in support of this one time CT extension credits an operator action to open 1-RH-MOV-1701 to maintain the risk estimates for a 72-hour CT within acceptable regulatory thresholds. This action is credited to exist in the mitigation of a steam generator tube rupture (SGTR) initiating event. The SGTR initiating event, if not mitigated by this operator action, would have a higher contribution to LERF. A standing order has been developed and the operators will be instructed regarding the manual action.

  The 1J1 480 Volt AC distribution subsystem (MCCs 1J1-2N and 2S) powers the normally open AFW MOVs (1-FW-MOV-100B & D) which are in the flow path of the two AFW pumps (1-FW-P-2 and 3B) that provide cooling to the A and B Steam Generators. The calculation performed in support of this one time CT extension credits an operator action to throttle or isolate AFW flow to the Steam Generators, as needed to maintain the risk estimates for a 72-hour CT within acceptable regulatory thresholds. This action is credited to exist in the mitigation of a steam generator tube rupture (SGTR) initiating event. The SGTR initiating event, if not mitigated by this operator action, would have a higher contribution to LERF.

With the inclusion of the standing order to address the manual operator action, it is concluded that defense-in-depth against human error was not impacted by the proposed change for a one time extended Completion Time.

6.3 Safety Margin Assessment

The overall margin of safety is not decreased due to the increased CT for the "J" electrical distribution system since the system design and operation are not altered by the proposed increase in CT.

The safety analysis acceptance criteria stated in the Updated Final Safety Analysis

Report (UFSAR) is not impacted by the change. Redundancy and diversity of the electrical distribution system will be maintained with the exception of the MCCs 1J1-2N and 2S. The proposed change will not allow plant operation in a configuration outside the design basis. The electrical distribution system requirements credited in the accident analysis will remain the same. It was concluded that safety margins were not impacted by the proposed changes.

6.4 Dominant Accident Sequences

The dominant accident sequences were reviewed for the case with the MCCs 1J1 2N and 2S unavailable. The results are as follows.

- The Steam Generator Tube Rupture (SGTR) initiating event contributes 64% (1.0E-6/yr) of the Large Early Release Frequency (LERF) with the MCCs 1J1- 2N and 2S unavailable. This configuration is LERF-limiting, rather than CDF-limiting, because the SGTR is a containment bypass event.
  - The most limiting SGTR sequences include a failure to cool down and depressurize the RCS. This sequence contributes 21% (3.3E-7/yr) to LERF and is independent of the MCC outage.
  - The next two most limiting SGTR sequences include failures of the High Head Safety Injection (HHSI) system and failure of feedwater isolation. These sequences total 21% (3.3E-7/yr) of LERF. The HHSI failures occurred due to the MCC outage on one train and coincident, random failures on the opposite train. Little or no credit for flowpath recovery was taken.
  - Several SGTR sequences include failure of the Residual Heat Removal (RHR) system due to the tagout of the MCCs 1J1-2N and 2S. This bus powers the normally-closed letdown isolation valve 1-RH-MOV-1701 from the Reactor Coolant System to the RHR system inside containment. If recovery of the MOV is unsuccessful, then decay heat is removed via the secondary system and any secondary faults will result in an offsite release. These sequences total 13% (2.0E-7/yr) of LERF.

- The most limiting LERF sequence, which is not a SGTR, is the vessel rupture which contributing 18% (2.7E-7/yr) to LERF. This sequence is independent of the proposed MCC Completion Time.

- There are no other sequences involving the MCC tagout that contribute more than 5% to the overall LERF. The Core Damage Frequency (CDF) impact of the proposed CT is less limiting than its LERF impact.

6.5 Summary

This risk evaluation supports a one-time 72-hour CT for the North Anna 480 Volt AC distribution subsystem (MCCs 1J1-2N and 2S). The increase in annual Core Damage and Large Early Release Frequencies associated with the proposed change in the Technical Specification CT are characterized as “small changes” by Regulatory Guide 1.174. The Incremental Conditional Core Damage and Large Early Release

Probabilities associated with the proposed Technical Specification CT meet the acceptance criteria in Regulatory Guide 1.177.

Sensitivity calculations were not performed because the analysis includes significant conservatism and still demonstrated substantial margin to the limits of Regulatory Guides 1.174 and 1.177. No further assessment of modeling uncertainty is required.

The Regulatory Guide 1.174 requirement to "Track Cumulative Impacts" for "small changes" is satisfied by the Dominion model maintenance program procedures.

This evaluation assumes planned maintenance that may result in the unavailability of components important to safety will be prohibited during the extended 72-hour CT of the Unit 1 "J" 480 Volt AC distribution subsystem (MCCs 1J1-2N and 2S). Otherwise, it would be necessary to impose Tier 2 restrictions as per Regulatory Guide 1.177.

7.0 Regulatory Safety Analysis

7.1 No Significant Hazards Consideration

The proposed change, a one-time extended Completion Time of Technical Specification 3.8.9 Condition A, will provide an opportunity to interrogate the damaged breaker, and to fully investigate the extent of condition and to ensure continued bus reliability for the remainder of the operating cycle.

The proposed change is based on a risk-informed evaluation performed in accordance with Regulatory Guides (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications." Dominion has evaluated whether or not a significant hazards consideration is involved with the proposed change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

   The proposed change does not alter any plant equipment or operating practices in such a manner that the probability of an accident is significantly increased. The proposed change will not alter assumptions relative to the mitigation of an accident or transient event. Manual operator actions in the event of a SGTR have been identified during the one-time extended CT for the 1J1 MCC outage. A risk-informed evaluation of these operator actions has been performed and the increase in annual Core Damage and Large Early Release Frequencies associated with the proposed change in the Technical Specification CT are characterized as "small changes" by Regulatory Guide 1.174. The Incremental Conditional Core Damage and Large Early Release Probabilities associated with the proposed Technical Specification CT meet the acceptance criteria in Regulatory Guide 1.177.

The ICCDP and ICLERP are 1.01E-7 per year and 9.86E-9 per year, respectively. These results are below the RG 1.177 limits of 5E-7 for ICCDP and 5E-8 for ICLERP.

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

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The systems' design and operation are not affected by the proposed change. The safety analysis acceptance criteria stated in the Updated Final Safety Analysis Report is not impacted by the change. Redundancy and diversity of the electrical distribution system will be maintained with the exception of the MCCs 1J1-2N and 2S. The proposed change will not allow plant operation in a configuration outside the design basis.

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

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

7.2 Environmental Assessment

This amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) as follows:

(i) The amendment involves no significant hazards consideration.

As described above, the proposed change involves no significant hazards consideration.

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

The proposed change does not involve the installation of any new equipment, or the modification of any equipment that may affect the types or amounts of effluents that may be released offsite. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

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

The proposed change does not involve plant physical changes, or introduce any new mode of plant operation. Therefore, there is no significant increase in individual or cumulative occupational radiation exposure.

Based on the above, Dominion concludes that the proposed change meets the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.22 relative to requiring a specific environmental assessment by the Commission.

8.0 Conclusion

The proposed change, a one-time extended CT of Technical Specification 3.8.9 Condition A, will provide an opportunity to interrogate the damaged breaker, and to fully investigate the extent of condition and to ensure continued bus reliability for the remainder of the operating cylce. The risk-informed evaluation concludes that the increase in core damage and large early release frequencies associated with the proposed change are 1.01E-7 per year and 9.86E-9 per year, respectively, which are characterized as "very small changes" by RG 1.174. The incremental conditional core damage and large and early release probabilities associated with the proposed change are each within the acceptance criteria in RG 1.177.

The initial assessment of the breaker and switchgear did not identify any evidence that would affect the operability of the MCCs 1J1-2N and 2S. However, in order to interrogate the failed breaker and the Unit 1 MCCs 1J1-2N and 2S and ensure the emergency power system continue in a reliable condition for the remainder of the operating cycle, additional time beyond the eight hours provided by the existing CT will be necessary.

The Facility Safety Review Committee (FSRC) has reviewed the proposed change to the Technical Specifications and has concluded that it does not involve a significant hazards consideration and will not endanger the health and safety of the public.

9.0 References

[NB 01] "North Anna Power Station Units 1 And 2 Probabilistic Risk Assessment Model Notebook Part III, PRA Model Development Category QU - Quantification Volume QU.2, Model Quantification Results," Revision 2 North Anna N105A Model, March 2007.

[NB 02] "North Anna Power Station Units 1 and 2 Probabilistic Risk Assessment Model Notebook Part IV, Appendix A, PRA Quality Summary Notebook," Revision 0 for N105A model, August 2007.

[REPORT 02] "Individual Plant Examination of Non-Seismic External Events and Fires - North Anna Power Station Units 1 And 2," Virginia Electric and Power Company, 1994.

[REPORT 03] "North Anna Power Station Units 1 and 2 Report on Individual Plant Examination of External Events (IPEEE) – Seismic Prepared in Response to USNRC Generic Letter 88-20 Supplements 4 and 5," May 1997.

[REPORT 04] NUREG-1488, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," April 1994.

[REPORT 05] North Anna Power Station Probabilistic Safety Assessment Peer Review Certification Report, July, 2001.

[RG 01] Regulatory Guide 1.174, Revision 1, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis," November 2002.

[RG 02] Regulatory Guide 1.177, Revision 0, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications," August 1998.

[RG 03] Regulatory Guide 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, January 2007."

[STD 01] ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," April 2002.

[STD 02] ASME RA-Sa-2003, "Addenda to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," December 2003.

[STD 03] ASME RA-Sb-2005, "Addenda to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," December 2005.

Attachment 2

Mark-up of Unit 1 Technical Specifications Change

North Anna Power Station Unit 1
Virginia Electric and Power Company
(Dominion)

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems–Operating

LCO 3.8.9 The following distribution subsystems shall be OPERABLE:

a. The Train H and Train J AC, DC, and AC vital buses; and

b. The necessary AC, DC and AC vital buses on the other unit for each required shared component.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more LCO 3.8.9.a AC electrical power distribution subsystem(s) inoperable.	A.1 ---------NOTE-------- Enter applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources–Operating," for DC train(s) made inoperable by inoperable distribution subsystem(s). ---------------------- Restore AC electrical power distribution subsystem(s) to OPERABLE status.	----NOTE---- The One Time Only Completion Time for maintenance on the 1J1-2N/2S MCCs is 72 hours. -------- 8 hours AND 16 hours from discovery of failure to meet LCO

Attachment 3

Proposed Unit 1 Technical Specifications Change

North Anna Power Station Unit 1
Virginia Electric and Power Company
(Dominion)

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems—Operating

LCO 3.8.9 The following distribution subsystems shall be OPERABLE:

a. The Train H and Train J AC, DC, and AC vital buses; and

b. The necessary AC, DC and AC vital buses on the other unit for each required shared component.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more LCO 3.8.9.a AC electrical power distribution subsystem(s) inoperable.	A.1	----------NOTE-------- Enter applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources—Operating," for DC train(s) made inoperable by inoperable distribution subsystem(s). ---------------------- Restore AC electrical power distribution subsystem(s) to OPERABLE status.	------NOTE----- The One Time Only Completion Time for maintenance on the 1J1-2N/2S MCCs is 72 hours. -------------- 8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO

Attachment 4

Supporting Documentation
(Deterministic Analysis)

North Anna Power Station Unit 1
Virginia Electric and Power Company
(Dominion)

Table of Contents

1.0 Background

On April 22, 2009, a small breaker fire occurred inside a breaker cubicle located in the North Anna cable vault. The breaker (1-EE-BKR-1J1-2S-J1) supplies a non-safety-related load (a Control Rod Drive Mechanism fan) but the breaker itself is safety-related. The safety-related function is to isolate the non-safety-related load from the adjacent safety-related Motor Control Center (MCC 1J1-2S) in the event of an electrical fault. An operability evaluation has been performed for MCC 1J1-2S and it remains operable. A Root Cause Evaluation (RCE) of this event is in-progress. The RCE Team has identified a number of possible causes and needs to perform a hands-on inspection of the breaker to determine the exact cause, and thus the extent of condition.

To accomplish the inspection of breaker 1J1-2S-J1, both MCC 1J1-2N and MCC 1J1-2S must be de-energized because they share a common supply breaker. The current Technical Specification (TS) Completion Time (CT) for this condition is 8 hours. Removal of the breaker, inspection of the breaker and adjacent MCC, and replacement of the breaker will require approximately 52 hours, under the worst case scenario. As such, Dominion is requesting a one-time extension of 72 hours to this CT.

The following is a deterministic analysis of what components will be de-energized, the systems that will be affected, the associated Limiting Condition of Operation (LCO) entry requirements, and any actions that will be taken to reduce risk during the breaker inspection.

2.0 Introduction

North Anna has two trains of Emergency Safety Function (ESF) equipment. These trains are designated “H” and “J”. De-energizing the MCCs 1J1-2N and 2S will affect 97 loads, some of which effect safety-related equipment on the “J” train. Separate MCCs provide power to the “H” trains of these safety systems. Roughly half of the loads to be de-energized would require entry into a TS or Technical Requirements Manual (TRM) LCOs if they were made inoperable by means other than de-energizing the MCCs 1J1-2N and 2S.

This deterministic analysis will first present a summary of actions and conditions that will ensure the MCCs 1J1-2N and 2S can be safely and successfully de-energized. This will be followed by a general overview of the affected emergency safety functions. Following this, tables will identify all components to be de-energized and, where applicable, will detail associated operability concerns, redundant trains of equipment, and sources of electrical power to that redundant equipment.

The following data was obtained from approved Station Load Lists, simulator runs of design basis accident scenarios with the MCCs 1J1-2N and 2S de-energized, and review from Senior Reactor Operators (SROs).

3.0 Assurances of Safety and Success

When the MCCs 1J1-2N and 2S are de-energized, there will be a number of measures in place to ensure the evolution is completed in a safe and successful fashion, focusing primarily on maintaining electric power to "H" trains of equipment. These measures also include actions to be performed prior to de-energizing the MCCs 1J1-2N and 2S, control of the work activity, and emplacement of a contingency plan should an event occur during maintenance.

Prior to de-energizing the MCC, protected equipment will be identified and marked. The loads associated with the "H" trains of equipment are normally fed by the "C" Reserve Station Service Transformer. Back-up power is available through the 1 "H" Emergency Diesel Generator, the "F" transfer bus, and the Alternate AC Diesel Generator. This equipment is currently performing satisfactorily and will be designated protected equipment for the duration of the evolution. Further, there is to be no work planned in the switchyard during this 72 hour period.

Operations and Maintenance, will receive job-specific training prior to beginning the evolution. Maintenance staff will perform a full pre-work review of the entire evolution in the days preceding the scheduled maintenance. The morning of, there will be a second pre-job brief to reinforce points from the pre-work review. A contingency plan will also be added to applicable Electrical Maintenance procedures that will expedite re-energizing the MCCs, if needed.

To minimize the risk of fire damage during the maintenance activity, fire watches will be set in the cable vault and tunnel, and the Service Water Pump House (SWPH). A fire in the cable vault and tunnel could damage the MCCs which feed the "H" train of emergency equipment. A fire in the SWPH could result in the loss of all four normal SW pumps leaving only two Auxiliary SW pumps operable. These fire watches provide greater protection against such consequences.

4.0 Affected Emergency Safety Functions

Components associated with the MCCs 1J1-2N and 2S support the following emergency safety functions: Emergency Core Cooling, Auxiliary Feedwater, Containment Depressurization, and Containment Isolation. These emergency safety functions are common to a number of design basis accidents. The actions following a Steam Generator Tube Rupture are uniquely affected by de-energizing the MCCs 1J1-2N and 2S, and will therefore be discussed in more depth.

Unless otherwise stated, the safety functions of the inoperable equipment listed will be accomplished by redundant trains, none of which rely on the MCCs 1J1-2N and 2S for power. Actions to be taken either prior to de-energizing the MCCs or following a design basis accident, to assist in accident mitigation, are included as needed. The LCOs associated with these de-energized components are addressed in the "Limiting Conditions of Operation" section of this document.

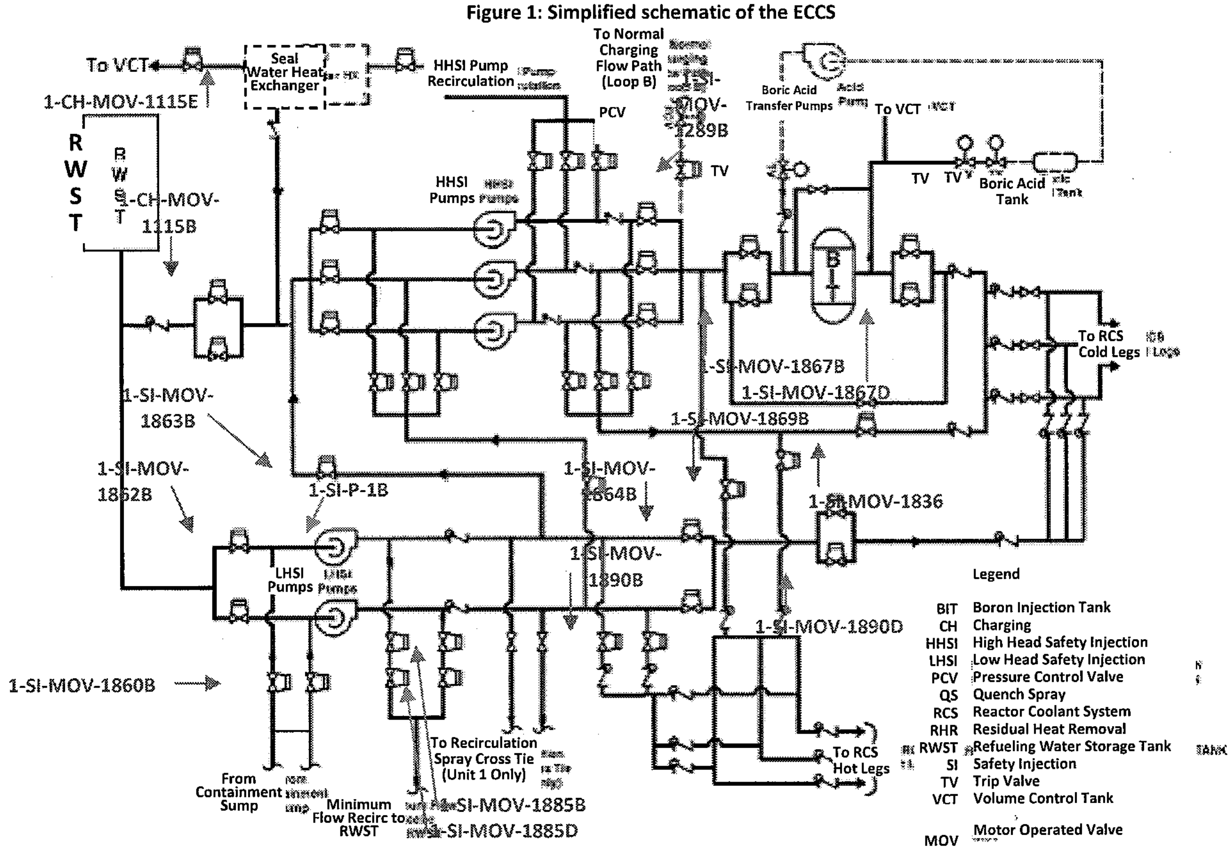
4.1 Emergency Core Cooling System

While MCCs 1J1-2N and 2S are out for maintenance the following equipment associated with the Emergency Core Cooling System (ECCS) will be inoperable. For the following equipment, redundant trains can provide the safety function and no additional actions will be required. However, a precautionary measure to be taken to maintain the integrity of a LHSI pump will be discussed. The Background section of the ECCS TS Bases is provided in Appendix A.

- "J" train of Safeguards ventilation exhaust
- "J" train of Boron Injection Tank Heat Tracing
- "J" train Boron Injection Tank flow path MOVs
- "J" train High Head Safety Injection MOVs
- "J" train Low Head Safety Injection MOVs

Figure 1 shows a simplified schematic of the ECCS System; all MOVs that will be de-energized and have an associated LCO are labeled.

Figure 1: Simplified schematic of the ECCS

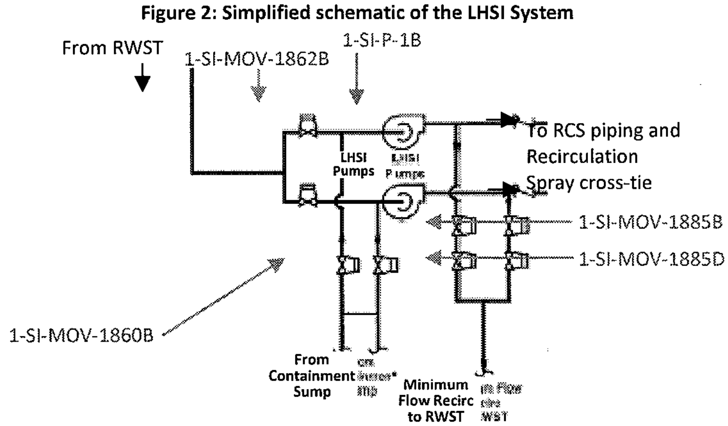


4.1.1 Low Head Safety Injection

Following an actuation of the ECCS, the "H" and "J" bus Low Head Safety Injection (LHSI) suction headers are swapped over from the Refueling Water Storage Tank (RWST) to the containment sump as inventory in the RWST decreases. This is done to provide long-term cooling. However, the "J" train MOVs in the flow path between the RWST and containment will be de-energized and unable to realign. To prevent damage, the "J" bus LHSI pump (1-SI-P-1B) will be secured when the RWST level reaches 15%.

Figure 2 shows a simplified schematic of the LHSI System; the impacted pump and all de-energized MOVs that have an associated LCO are labeled. The valve that opens the flow path from the RWST to the suction of 1-SI-P-1B (1-SI-MOV-1862B) will be de-energized open. The valve that opens the flow path between the containment sump and the suction of 1-SI-P-1B (1-SI-MOV-1860B) will be de-energized closed. The two valves that allow for a recirculation loop to be established between the LHSI pump and the RWST (1-SI-MOV-1885B, -1885D) will be de-energized open.

Figure 2: Simplified schematic of the LHSI System



Under normal operating conditions, the containment sump valve (1-SI-MOV-1860B) would open and then the three valves allowing flow to / from the RWST (1-SI-MOV-1885B, -1885D, & -1862B) would close, thus completing swapover of the LHSI header. However, this will not be possible when the MCCs 1J1-2N and 2S are de-energized. Under these conditions, the "J" train will not be able take suction from the containment sump and instead the "H" train will provide the safety function of the LHSI system.

4.2 Auxiliary Feedwater

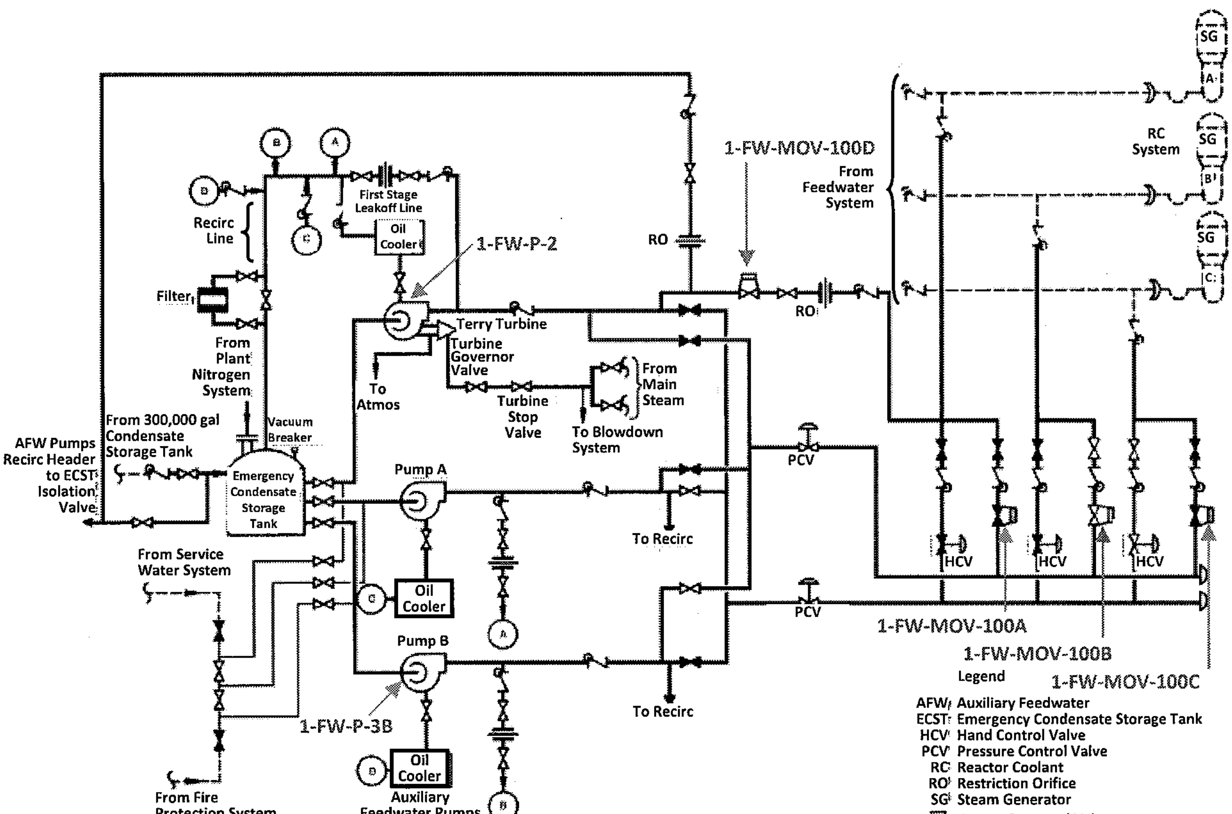
Four Auxiliary Feedwater (AFW) pump discharge MOVs (1-FW-MOV-100A, B, C, & D) will be de-energized when MCCs 1J1-2N and 2S are taken out for maintenance. Two of these valves (1-FW-MOV-100B & D) are in the flow path of two AFW pumps (1-FW-P-2 & 3B), which provide cooling to the A and B Steam Generators. Two of these valves (1-FW-MOV-100A & C) are usually isolated and will be isolated when de-energized; therefore, they do not affect operability.

The two valves in the flow path of the pumps (1-FW-MOV-100B & D) are normally kept open and will be open when de-energized. These MOVS are usually manually operated from the control room. They are considered to remain operable when de-energized because they can be manually operated locally, to throttle or isolate AFW flow to the Steam Generators, as needed. The flow path to the Steam Generators remains operable and fulfills the required design function during this evolution.

Prior to beginning maintenance, an Operator will be designated as the AFW valve operator and a sound-powered phone system will be utilized in the AFW pump house in order to provide direct communication with the control room. In the event of an AFW actuation, this operator will be deployed to the AFW pump house and will receive direct instruction from the Control Room.

Figure 3 shows a simplified schematic of the AFW System; the affected pumps and the valves to be de-energized that have an associated LCO are labeled. The Background section of the AFW TS Bases is provided in Appendix B.

Figure 3: Simplified schematic of the AFW System



4.3 *Containment Depressurization Actuation*

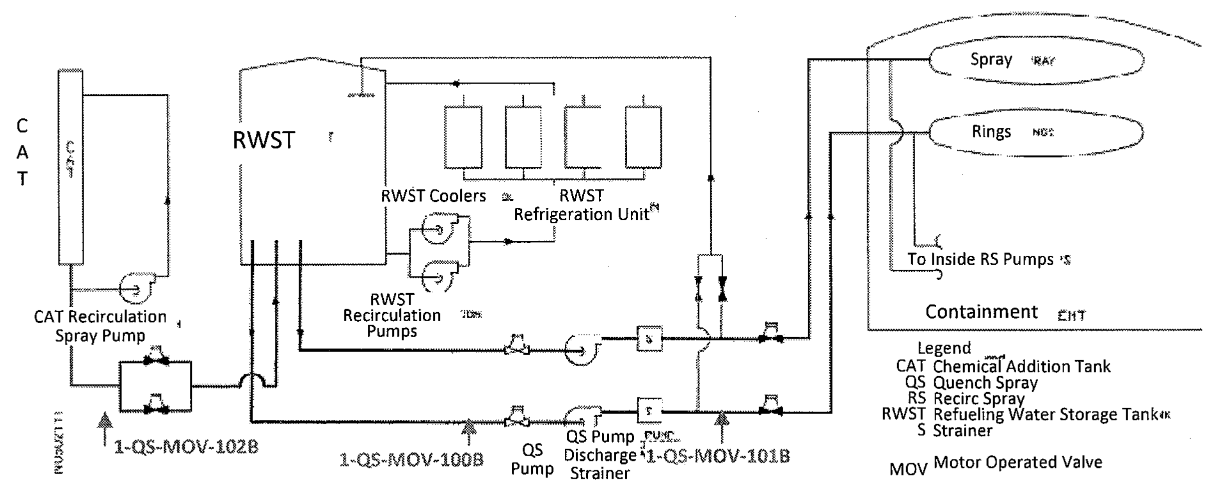
In the event of a Containment Depressurization Actuation (CDA) the following equipment will be inoperable due to the MCCs 1J1-2N and 2S being de-energized for maintenance. For the following equipment, redundant trains can support the safety function of these trains and no additional actions will be required.

- "J" train Quench Spray pump MOVs,
- "J" train of chemical addition flow path to Quench Spray.
- "J" train of Casing Cooling MOVs, which provides NPSH to "B" outside Recirculation Spray pump,
- "J" train Recirculation Spray pumps MOVs,
- "J" train of Service Water MOVs to Recirculation Spray heat exchangers, and
- "J" train of Service Water MOV to isolate Component Cooling heat exchangers.

4.3.1 Quench Spray

Under these conditions, one train of Quench Spray (QS) will be inoperable, as will one train of the Chemical Addition flow path into the system. The redundant train of QS and the other Chemical Addition flow path can provide the safety-related function. These inoperabilities are shown in Figure 4. The Background section of the Quench Spray TS Bases is provided in Appendix C.

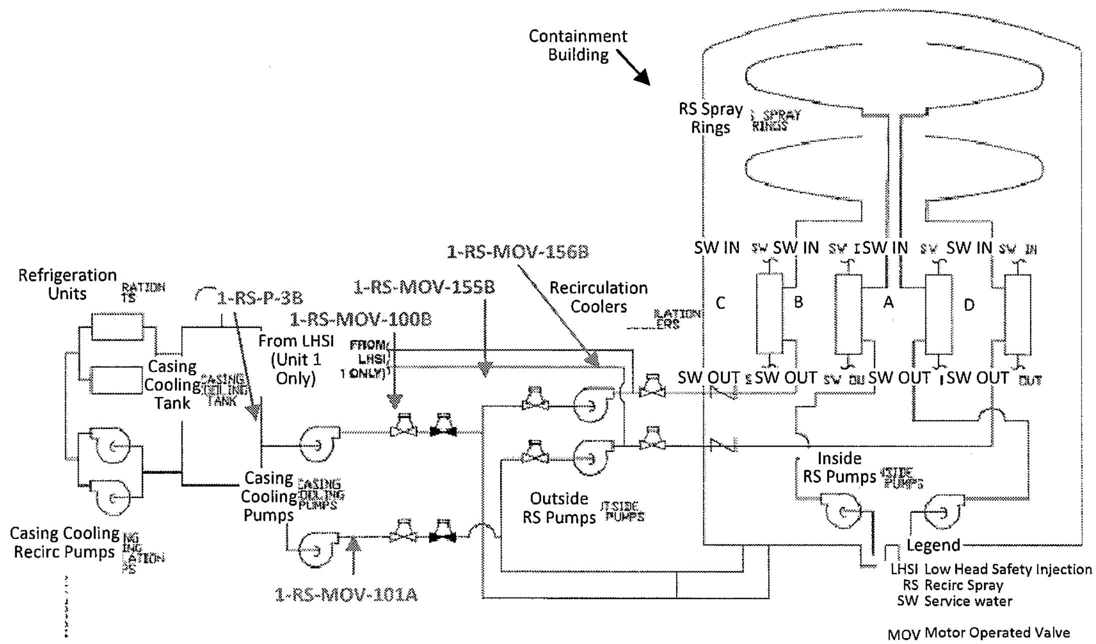
Figure 4: Simplified schematic of the Quench Spray System



4.3.2 Recirculation Spray

The inoperable components in the Recirculation Spray (RS) System result in two sub-systems of one train of RS heat exchangers being inoperable. The redundant train of RS heat exchangers can provide CDA heat removal. Figure 5 shows a simplified schematic of the RS System; de-energized components that have an associated LCO are labeled. The Background section of the RS System TS Bases is provided in Appendix D.

Figure 5: Simplified schematic of the Recirculation Spray System



4.3.3 Service Water

The inoperable MOVs in the Service Water (SW) System result in one loop of SW and one isolation MOV to the Component Cooling Water (CCW) heat exchangers being inoperable. The redundant loop of SW will provide necessary cooling and the redundant MOV can provide isolation, if needed. The de-energized components from the SW System that have an associated LCO are shown in Figures 6 and 7. The Background section of the SW System TS Bases is provided in Appendix E.

Figure 6: Simplified schematic of the SW System and RS heat exchangers

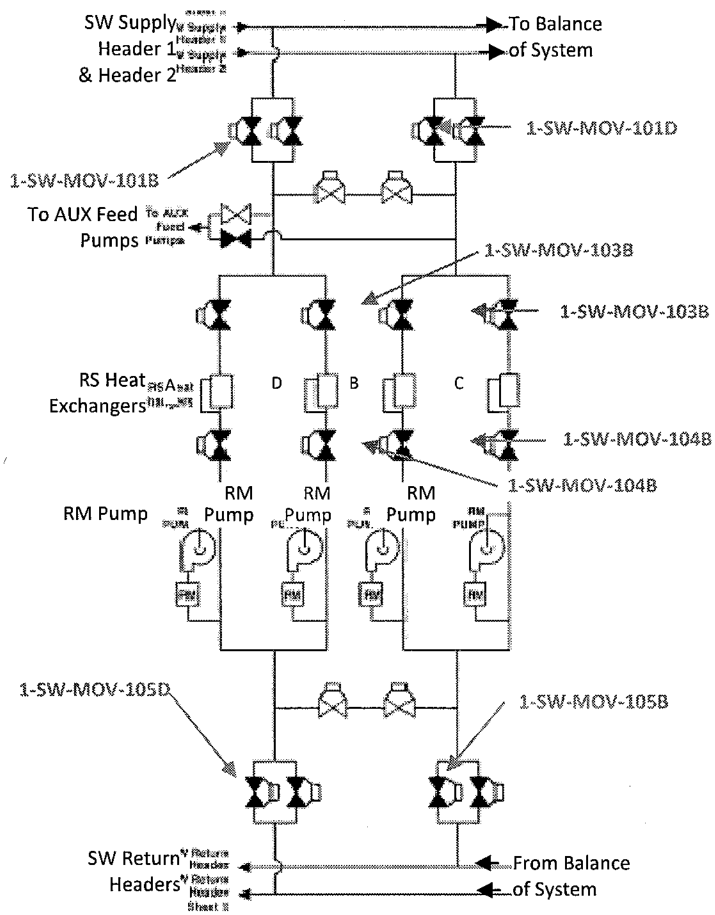
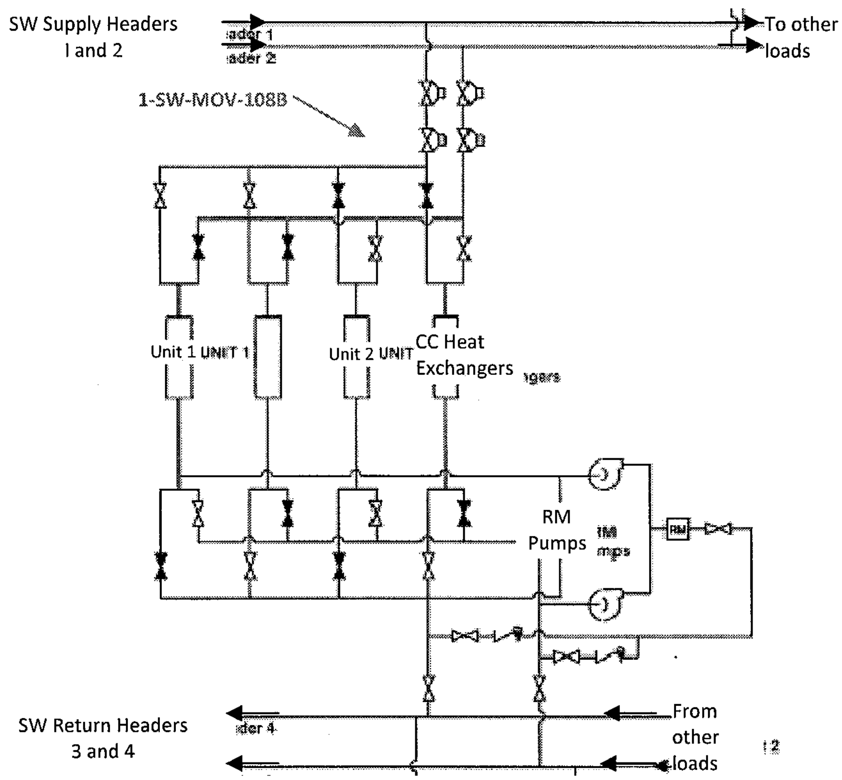


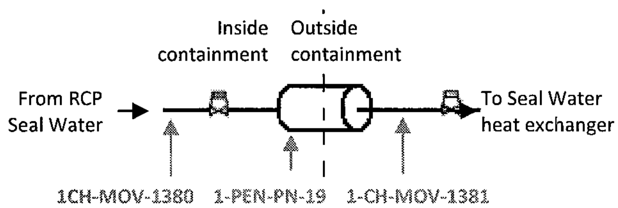
Figure 7: Simplified schematic of the SW System and CC heat exchangers



4.4 Containment Isolation

The "J" train Reactor Coolant Pump Seal Water Return containment isolation valve will be de-energized during this evolution (1-CH-MOV-1381). A redundant valve (1 CH-MOV-1380) is able to perform the safety function. Section 5, "Limiting Conditions of Operation," addresses the operability determination associated with this valve. Figure 8 is a simplified schematic of the containment penetration; the affected MOVs and penetration are labeled.

Figure 8: Simplified schematic of the containment penetration



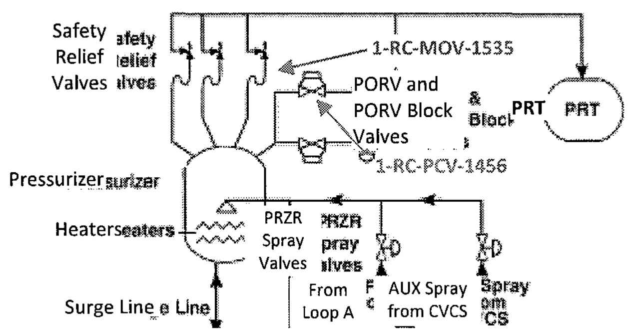
4.5 Steam Generator Tube Rupture

Two features are credited for post-accident mitigation of a Steam Generator Tube Rupture (SGTR) that are not credited in any other design basis accident: Pressurizer Power-Operated Relief Valves (PORVs) and Residual Heat Removal (RHR). The Background sections of their TS Bases are provided in Appendices F and G.

Pressurizer PORVs may be used to depressurize the RCS. However, when the 1J1-2N / 2S MCCs are de-energized, the block valve upstream of one of the two PORVs will be inoperable. The block valve (1-RC-MOV-1535) serves to isolate the "J" train PORV (1-RC-PCV-1456). The block valve will be de-energized open. With the block valve inoperable, LCO 3.4.11, Action D will be entered with a 72 hour Completion Time. This LCO also requires that the PORV be placed under "manual" operation. This manual operation will prevent cycling of the "J" train PORV so as to prevent a Small Break LOCA from occurring should the PORV fail to close after being cycled open.

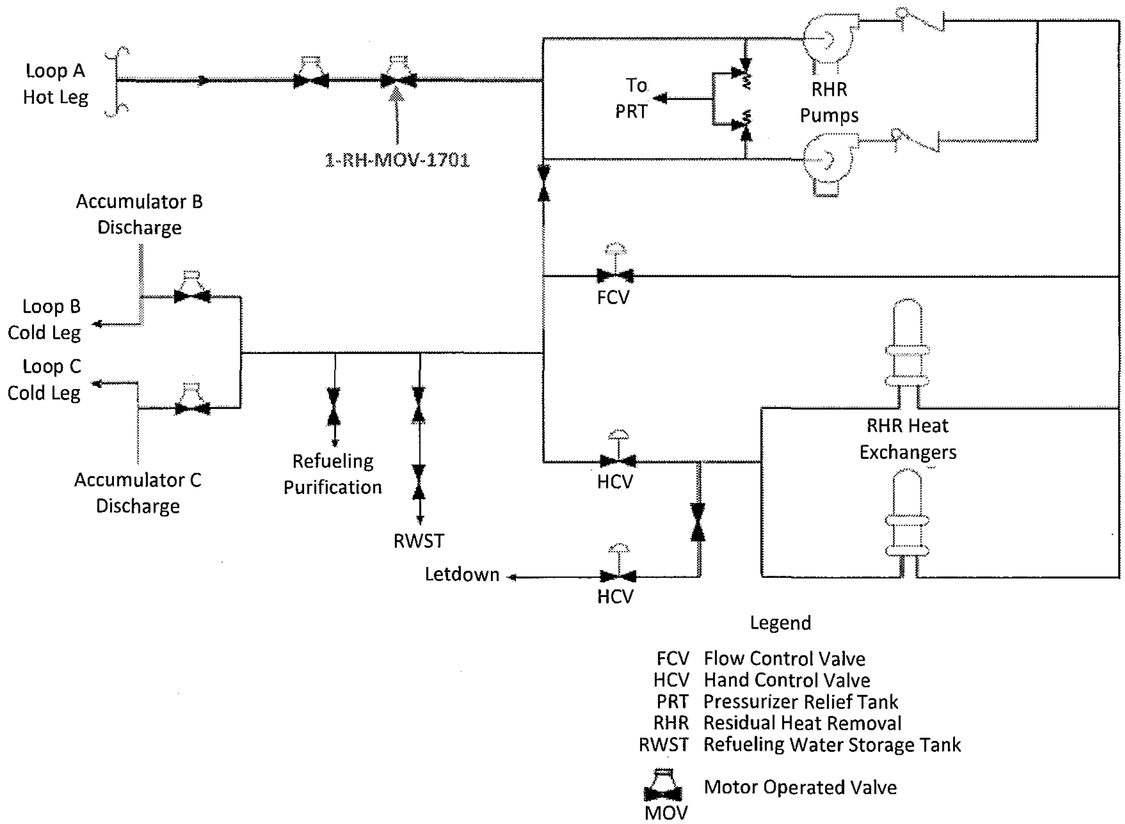
In the event of an SGTR, the redundant PORV (1-RC-PCV-1455C) will be used, if necessary, to depressurize the RCS. This valve and its block valve are powered by the "H" train, and will be unaffected when the MCCs 1J1-2S and 2N are de-energized. Figure 9 shows a simplified schematic of the pressurizer; the block valve to be de-energized and its associated PORV are labeled.

Figure 9: Simplified schematic of the Pressurizer



Additionally, the "J" train RHR suction MOV (1-RH-MOV-1701) will be de-energized. This valve is normally closed and will be de-energized closed. In the event of a SGTR, this MOV will be manually operated locally, if necessary, during the cooldown process. This valve is located inside of containment and is usually manually operated from the control room. Section 5.5.4.3.4 of the North Anna UFSAR discusses the manual operation of this valve. Figure 10 shows a simplified schematic of the RHR system; the valve to be manually operated is labeled.

Figure 10: Simplified schematic of the Residual Heat Removal



5.0 Limiting Conditions of Operation

Forty-three of the potentially de-energized loads would require entry into either a TS or Technical Requirements Manual (TRM) LCO, were they to fail independent of de-energizing the MCCs 1J1-2N and 2S. Table 1, on the following page, lists all components that would require such an entry. Where applicable, the list also includes CTs, redundant components, and a brief description of the component. The power supplies to all redundant components listed in Table 1, are the MCCs 1H1-2S and 2N. When the MCCs 1J1-2N and 2S are de-energized, none of these LCOs will be entered. Instead, LCO 3.0.6[1] will be entered for all components listed. The only exception to this will be LCO 3.4.11 for the PORV block valve (1-RC-MOV-1535).

Note: The individual components listed in Table 1 comprise the trains of equipment that were discussed in the Affected Emergency Safety Functions section.

Table 1: List of components requiring Tech Spec or TRM LCO entry upon independent failure.

Mark Number	Description	LCO	CT	Redundant Component

[1] LCO 3.0.6 states: "When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered" provided a satisfactory evaluation is performed in accordance with TS 5.5.14. This evaluation will be performed immediately prior to de-energizing the 1J1-2N / 2S MCCs.

1-SI-MOV-1867B	Isolation inlet MOV to 1-SI-TK-2, Boron Injection Tank	3.5.2	72 h	1-SI-MOV-1867A
1-SI-MOV-1867D	Boron Injection Tank outlet	3.5.2	72 h	1-SI-MOV-1867C
1-CH-MOV-1115B	Charging pump suction from Refueling Water Storage Tank	3.5.2	72 h	1-CH-MOV-1115D
1-CH-MOV-1115E	Charging pump suction from Volume Control Tank	3.5.2	72 h	1-CH-MOV-1115C
1-SI-MOV-1869B	Normal charging header discharge to Hot Leg	3.5.2	72 h	1-SI-MOV-1869A
1-SI-MOV-1863B	"B" Low Head SI discharge to charging pump suction	3.5.2	72 h	1-SI-MOV-1863A
1-SI-MOV-1864B	"B" Low Head SI pump discharge to Cold Leg	3.5.2	72 h	1-SI-MOV-1864A
1-SI-MOV-1890B	"B" Low Head SI pump discharge to Hot Leg	3.5.2	72 h	1-SI-MOV-1890A
1-SI-MOV-1890D	Low Head SI discharge to Cold Leg	3.5.2	72 h	1-SI-MOV-1890C
1-SI-MOV-1836	Alternate header discharge to Cold Legs	3.5.2	72 h	1-SI-MOV-1869A
1-CH-MOV-1289B	Normal charging header isolation valve	3.5.2	72 h	1-CH-MOV-1289A
1-QS-MOV-101B	"B" Quench Spray pump discharge	3.6.6	72 h	1-QS-MOV-101A
1-QS-MOV-100B	"B" Quench Spray pump suction	3.6.6	72 h	1-QS-MOV-100A
1-SW-MOV-103B	"B" Recirc Spray heat exchanger supply	3.6.7	72 h	1-SW-MOV-103A
1-SW-MOV-103C	"C" Recirc Spray heat exchanger supply	3.6.7	72 h	1-SW-MOV-103D
1-SW-MOV-104B	"B" Recirc Spray heat exchanger return	3.6.7	72 h	1-SW-MOV-104A
1-SW-MOV-104C	"C" Recirc Spray heat exchanger return	3.6.7	72 h	1-SW-MOV-104D
1-RS-MOV-155B	"B" outside Recirc Spray pump suction	3.6.7	72 h	1-RS-MOV-155A
1-RS-MOV-156B	"B" outside Recirc Spray pump discharge	3.6.7	72 h	1-RS-MOV-156A
1-QS-MOV-102B	Chemical Addition Tank outlet	3.6.8	72 h	1-QS-MOV-102A
1-SW-MOV-108B	"A" Service Water supply header to Component Cooling hx	3.7.8	72 h	1-SW-MOV-108A
1-HV-F-40B	Safeguards exhaust fan	3.7.12	7 d	1-HV-F-40A
1-SW-MOV-101B	"A" Service Water hdr supply to Recirc Spray hx	Note 1	------	1-SW-MOV-101A
1-SW-MOV-101D	"B" Service Water hdr supply to Recirc Spray hx	Note 1	------	1-SW-MOV-101C
1-SW-MOV-105B	Recirc Spray hx return to "B" Service Water hdr	Note 1	------	1-SW-MOV-105A
1-SW-MOV-105D	Recirc Spray hx return to "A" Service Water hdr	Note 1	------	1-SW-MOV-105C
1-FW-MOV-100B	MOV header to feed "B" steam generator	Note 2	------	N/A
1-FW-MOV-100D	Terry (steam-driven) turbine discharge to "A" steam generator	Note 2	------	N/A
1-CH-MOV-1381	Reactor Coolant Pump seal water return	Note 3	------	1-CH-MOV-1380
1-RC-MOV-1535	MOV isolation for 1-RC-PCV-1456	Note 4	------	1-RC-MOV-1536
1-SI-MOV-1885D	"B" Low Head SI pump recirc	Note 5	------	N/A
1-SI-MOV-1885B	"B" Low Head SI pump recirc	Note 5	------	N/A
1-SI-MOV-1862B	"B" Low Head SI pump suction from Refueling Water Storage Tank	Note 5	------	1-SI-MOV-1862A
1-SI-MOV-1860B	"B" Low Head SI pump suction from containment sump	Note 5	------	1-SI-MOV-1860A
1-RH-MOV-1701	RHR suction MOV	Note 6	------	N/A
1-RS-MOV-100B	Casing Cooling pump discharge	Note 7	------	1-RS-MOV-101B
1-RS-MOV-101A	Casing Cooling pump discharge	Note 7	------	1-RS-MOV-100A
1-RS-P-3B	"B" Casing Cooling pump	Note 7	------	1-RS-P-3A
1-SI-EHR-2B	Boron Injection Tank (BIT) heater	TRM 3.5.1	30 d	1-SI-EHR-2A
1-EP-CB-11AR	BIT heat trace transformer 11R	TRM 3.5.1	30 d	1-EP-CB-11AN
1-EP-CB-11BR	BIT heat tracing distribution panel	TRM 3.5.1	30 d	1-EP-CB-11BN

Note 1: This will result in one Service Water loop being inoperable and two sub-systems of one train of Recirculation Spray being inoperable. LCO 3.7.8 Action C has a CT of

72 hours. LCO 3.6.7 Action B, for the Recirculation Spray system, also has a CT of 72 hours.

Note 2: This combination of inoperable components renders the discharge MOVs (1-FW-MOV-100B & -D) inoperable. These valves are in the flow path of two AFW pumps (1-FW-P-2 & -3B). These valves are normally kept open and will be open when de-energized. They are considered to remain operable because they can be manually operated locally, thus providing or isolating AFW flow to the Steam Generators, as needed. In the event of an AFW actuation, local operators will be deployed. The remaining two valves (1-FW-MOV-100A & -C) are usually isolated and do not affect operability.

Note 3: If this valve were to fail independent of the MCCs being de-energized, LCO 3.6.3 Action A would be entered with a CT of 4 hours. However, there is a redundant valve (1-CH-MOV-1380) which can accomplish the same safety function. Ability to accomplish the safety function allows implementation of LCO 3.0.6, which states, "When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered." Only the LCO for the support system, in this case the MCCs 1J1-2N and 2S, is required to be entered.

Note 4: This valve (1-RC-MOV-1535) is a block valve up stream of the "J" train PORV (1-RC-PCV-1456). It will be de-energized open. At that time, LCO 3.4.11, Action D will be entered. This LCO requires the PORV to be placed under manual operation within one hour and requires the block valve to be restored to operable within 72 hours. The "H" train of PORV and block valve (1-RC-PCV-1455C and 1-RC-MOV-1536, respectively) will be operable to provide their safety functions, if needed.

Note 5: Following a Safety Injection, this LHSI suction header flow path must be secured during swapover from the Refueling Water Storage Tank to the containment sump. To prevent equipment damage, the associated pump (1-SI-P-1B) will be secured when swapover criteria is met.

Note 6: This MOV will be manually operated locally following a SGTR, if necessary. This MOV is located in containment and is normally manually operated remotely.

Note 7: These valves serve two functions. Early-on in an accident they allow for the delivery of Casing Cooling water to the outside Recirculation Spray pumps. After the Casing Cooling tank inventory is depleted, they serve to isolate the flow path from containment to prevent backflow of radioactivity through the line. Train "B" comprises a pump (1-RS-P-3B), one valve normally maintained open (1-RS-MOV-101B), and one valve normally maintained closed (1-RS-MOV-100B). When de-energized, the "B" train will only be able to maintain its containment isolation function, due to the normally closed valve remaining closed.

The "A" train, however, will maintain its ability to perform both functions. The pump (1-RS-P-3A) and the normally closed valve (1-RS-MOV-100A) are both powered by a "H" MCC and will therefore be functional. The normally open valve (1-RS-MOV-101A) is powered from the MCCs 1J1-2N and 2S and will be unable to close. Containment isolation will rely on the operability of 1-RS-MOV-100A.

The conditions described in Note 7 require: (1) entry into LCO 3.6.7 for the pump, with a CT of 7 days, and (2) entry into both LCO 3.6.7 and LCO 3.0.6 for the valves, with a CT of 72 hours. As was the case in Note 3, LCO 3.0.6 will be entered in lieu of LCO 3.6.3 because the safety function of this train (containment isolation) is maintained by the redundant valve (1-RS-MOV-100A).

6.0 Components with No Affect on Operability

Table 2 lists all potentially de-energized components that will not affect TS or TRM operability. Where applicable, a redundant component and its power supply have also been listed.

Table 3: List of components that have no affect on Tech Spec operability.

Mark Number	Description	Redundant Component	Power Source for Redundant Component
----------------	Future	N/A	N/A
1-CC-P-1B	Motor heater	N/A	N/A
1-CH-EHR-6B	Boric acid tank heater	1-CH-EHR-6A	1-EE-MCC-1H1-2N
1-CH-EHR-7B	Boric acid tank heater	1-CH-EHR-7A	1-EE-MCC-1H1-2N
1-CH-MOV-1269A	"B" Charging pump normal suction	N/A	N/A
1-CH-MOV-1269B	"B" Charging pump alternate suction	N/A	N/A
1-CH-MOV-1270B	"C" Charging pump alternate suction	N/A	N/A
1-CH-MOV-1286B	"B" Charging pump discharge to normal header	N/A	N/A
1-CH-MOV-1287B	"B" Charging pump discharge to alternate header	N/A	N/A
1-CH-MOV-1287C	"C" Charging pump discharge to alternate header	N/A	N/A
1-CH-MOV-1373	Charging pump common recirc	1-CH-MOV-1275A,B,C	1-EE-MCC-1H1-2S,-2N
1-CH-P-1B1	Aux oil pump for "B" charging pump	N/A	N/A
1-CH-P-2B	Boric acid transfer pump	1-CH-P-2A	1-EE-MCC-1H1-2S
1-CV-P-3B	Containment vacuum pump	1-CV-P-3A	1-EE-MCC-1H1-2N
1-DA-P-1B	Safeguards sump pump	1-DA-P-1A	1-EE-MCC-1H1-2S
1-EE-BKR-1J1-2S-A4R	480 Volt receptacle #35	N/A	N/A
1-EG-P-1JB	"1J" diesel fuel oil transfer pump	1-EG-P-1JA	1-EE-MCC-1J1-1

1-EP-CB-13R	Heat trace transformer 13R	1-EP-CB-13N	1-EE-MCC-1H1-2N
1-EP-CB-14AR	Heat trace transformer 14R	N/A	N/A
1-EP-CB-14BR	Heat tracing distribution panel	N/A	N/A
1-EP-CB-19B	Alt feed to 1-EP-CB-19B (SOV panel)	Primary Power Source	1-EP-CB-4B
1-EP-CB-41AR	Heat trace transformer 41R	1-EP-CB-41AN	1-EE-MCC-1H1-2S
1-EP-CB-41BR	Heat tracing distribution panel	1-EP-CB-41BN	1-EE-MCC-1H1-2S
1-EP-CB-80C	Alt feed to 1-EP-CB-80C (Inst panel)	Primary Power Source	1-EP-CB-4B
1-EP-CB-84B	Motor heater cabinet 1-EP-CB-84B	N/A	N/A
1-EP-CB-84G	Motor htr cabinet 1-EP-CB-84G	1-EP-CB-84F	1-EE-MCC-1H1-2N
1-FW-MOV-100A	MOV header to feed "A" steam generator	1-FW-HCV-100A	1-EI-CB-23E
1-FW-MOV-100C	MOV header feed to "C" steam generator	1-FW-HCV-100C	1-EI-CB-23E
1-HV-F-1B	Motor heaters	N/A	N/A
1-HV-F-37D	Control rod cooling fan	N/A	N/A
1-HV-F-37E	Control rod cooling fan	N/A	N/A
1-HV-F-37F	Control Rod Drive cooling fan	1-HV-F-37C	1-EE-MCC-1H1-2S
1-HV-F-68B	Transformer cooling fan for 1J 480 Volt	N/A	N/A
1-HV-F-70B	Aux feed pump house exhaust fan	1-HV-F-70A	1-EE-MCC-1H1-2N
1-HV-F-71B	Safeguard's emergency vent fan	1-HV-F-71A	1-EE-MCC-1H1-2N
1-HV-F-8B	Aux building central exhaust fan	1-HV-F-8A, 8C	1-EE-MCC-1H1-2N,-2J1-2N
1-HV-MOD-1230	1J substation fresh air supply damper	N/A	N/A
1-IA-C-2B	Containment Instrument Air compressor	1-IA-C-2A	1-EE-MCC-1H1-2N
1-IC-DRIV-1E	Incore instrumentation drive assembly "E" feeder	N/A	N/A
1-PG-P-2B	PG pump	1-PG-P-2A	1-EE-MCC-1H1-2N
1-QS-P-1B	Motor heaters	N/A	N/A
1-RH-MOV-1720B	RHR discharge to "C" RCS loop	1-RH-MOV-1720A	1-EE-MCC-1H1-2S
1-RH-P-1B	Motor heaters	N/A	N/A
1-RS-P-1B	Motor heaters	N/A	N/A
1-SI-MOV-1865C	"C" Accumulator outlet	Passive system	Passive System
1-SW-MOV-102B	Recirc Spray heat exchanger supply header crosstie	N/A	N/A
1-SW-MOV-106B	Recirc Spray heat exchanger return header crosstie	N/A	N/A

1-SW-MOV-110B	"A" Service Water header supply to air recirc fan	1-SW-MOV-110A	1-EE-MCC-1H1-2N
1-SW-MOV-114B	Air recirc fan return to "A" Service Water header	1-SW-MOV-114A	1-EE-MCC-1H1-2N
1-SW-P-6	Rad monitor sample pump	N/A	N/A
1-SW-P-7	Rad monitor sample pump	N/A	N/A
1-SW-P-9B	Rad monitor sample pump	N/A	N/A
2-CC-P-1B	Motor heater	N/A	N/A
Transformer 128	Alt feed to 1-EP-CB-19B and 1-EP-CB-80C	N/A	N/A
Transformer 65	Transformer 65	Transformer 64	1-EE-MCC-1H1-2N

7.0 Conclusion

Components associated with the MCCs 1J1-2N and 2S support the following emergency safety functions: Emergency Core Cooling, Auxiliary Feedwater, Containment Depressurization, and Containment Isolation. Unless otherwise stated, the safety functions of the inoperable equipment will be accomplished by redundant trains, none of which rely on the MCCs 1J1-2N and 2S for power. In certain instances, actions will be taken either prior to de-energizing the MCCs or following a design basis accident in order to ensure the successful actuation of these systems, as needed.

Rigorous examination of the system by station personnel has led to implementing measures focused on maintaining electric load to “H” trains of safety-related equipment and instructing operators on how to safely operate the plant under these conditions. North Anna Power Station will be able to accomplish this under any design basis accident scenarios that may arise during the 72-hour Completion Time.

In summary, a combination of factors will be utilized to ensure the success of this One Time Only evolution. These factors include, but are not limited to, actions to be performed prior to de-energizing the MCCs 1J1-2N and 2S and adherence to all commitments being made to the NRC by Virginia Electric and Power Company in this License Amendment Request, Serial Number 09-301.

Appendix A

Detailed descriptions of the Emergency Core Cooling System (ECCS) are provided in the form of excerpts from the North Anna Technical Specification Bases.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS—Operating

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;
b. Rupture of a control rod drive mechanism-control rod assembly ejection accident;
c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and
d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the MSLB where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. Within approximately 5 hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

The ECCS consists of two separate subsystems: High Head Safety Injection (HHSI) and Low Head Safety Injection (LHSI). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

The ECCS flow paths consist of piping, valves, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the HHSI pumps and the LHSI pumps. Each of the two subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from opposite trains to achieve the required 100% flow to the core.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps. Water from the supply header enters the LHSI pumps through parallel, normally open, motor operated valves. Water to the HHSI pumps is supplied via parallel motor operated valves to ensure that at least one valve opens on receipt of a safety injection actuation signal. The supply header then branches to the three HHSI pumps through normally open, motor operated valves. The discharge from the HHSI pumps combines prior to entering the boron injection tank (BIT) and then divides again into three supply lines, each of which feeds the injection line to one RCS cold leg. The discharge from the LHSI pumps combine and then divide into three supply lines, each of which feeds the injection line to one RCS cold leg. Control valves in the HHSI lines are set to balance the flow to the RCS. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs and preclude pump runout.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the LHSI pumps, the HHSI pumps supply water until the RCS pressure decreases below the LHSI pump shutoff head. During this period, the steam generators are used to provide part of the core cooling function. During the recirculation phase of LOCA recovery, LHSI pump suction is transferred to the containment sump. The LHSI pumps then supply the HHSI pumps. Initially, recirculation is through the same paths as the injection phase. Subsequently, recirculation alternates injection between the hot and cold legs.

The HHSI subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as an MSLB. The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

Appendix B

Detailed descriptions of the Auxiliary Feedwater (AFW) System are provided in the form of excerpts from the North Anna Technical Specification Bases.

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction through separate and independent suction lines from the emergency condensate storage tank (ECST) (LCO 3.7.6) and pump to the steam generator secondary side via separate and independent connections to the main feedwater (MFW) piping outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or steam generator power operated relief valves (SG PORVs) (LCO 3.7.4). If the main condenser is available, steam may be released via the steam dump valves and recirculated to the condenser hotwell.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. Each pump is aligned to one steam generator, and the capacity of each pump is sufficient to provide the designated flow assumed in the accident analysis. The pumps are equipped with recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class 1E power supply and normally feeds one steam generator, although each pump has the capability to be realigned to feed other steam generators. The steam turbine driven AFW pump receives steam from three main steam lines upstream of the main steam trip valves (MSTVs). The steam supply lines combine into a header which is isolated from the steam driven auxiliary feedwater pump by two parallel valves. Main steam trip valves, MS-TV-111A and MS-TV-111B (Unit 1), MS-TV-211A and MS-TV-211B (Unit 2) are powered from separate 125 V DC trains and actuated by the Engineered Safety Features Actuation System (ESFAS). Opening of either trip valve will provide sufficient steam to the steam driven pump to produce the design flow rate from the ECST to the steam generator(s).

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

The AFW pumps may be aligned and supply a common header capable of feeding all steam generators. One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure associated with the lowest setpoint MSSV. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry

Appendix C

Detailed descriptions of the Quench Spray (QS) System are provided in the form of excerpts from the North Anna Technical Specification Bases.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Quench Spray (QS) System

BACKGROUND

The QS System is designed to provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. The QS System, operating in conjunction with the Recirculation Spray (RS) System, is designed to cool and depressurize the containment structure to less than 2.0 psig in 1 hour and to subatmospheric pressure within 6 hours following a Design Basis Accident (DBA). Reduction of containment pressure and the iodine removal capability of the spray limit the release of fission product radioactivity from containment to the environment in the event of a DBA.

The QS System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a spray pump, a dedicated spray header, nozzles, valves, and piping. Each train is powered from a separate Engineered Safety Features (ESF) bus. The refueling water storage tank (RWST) supplies borated water to the QS System.

The QS System is actuated either automatically by a containment High-High pressure signal or manually. The QS System provides a spray of cold borated water into the upper regions of containment to reduce the containment pressure and temperature during a DBA. Each train of the QS System provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission product removal. The QS System also provides flow to the Inside RS pumps to improve the net positive suction head available.

The Chemical Addition System supplies a sodium hydroxide (NaOH) solution into the spray. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere. The NaOH added to the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The QS System is a containment ESF system. It is designed to ensure that the heat removal capability required during the post accident period can be attained. Operation of the QS System and RS System provides the required heat removal capability to limit post accident conditions to less than the containment design values and depressurize the containment structure to less than 2.0 psig in 1 hour and to subatmospheric pressure within 6 hours following a DBA.

The QS System limits the temperature and pressure that could be expected following a DBA and ensures that containment leakage is maintained consistent with the accident analysis.

Appendix D

Detailed descriptions of the Recirculation Spray (RS) System are provided in the form of excerpts from the North Anna Technical Specification Bases.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Recirculation Spray (RS) System

BACKGROUND

The RS System, operating in conjunction with the Quench Spray (QS) System, is designed to limit the post accident pressure and temperature in the containment to less than the design values and to depressurize the containment structure to less than 2.0 psig in 1 hour and to subatmospheric pressure within 6 hours following a Design Basis Accident (DBA). The reduction of containment pressure and the removal of iodine from the containment atmosphere by the spray limit the release of fission product radioactivity from containment to the environment in the event of a DBA.

The RS System consists of two separate trains of equal capacity, each capable of meeting the design and accident analysis bases. Each train includes one RS subsystem outside containment and one RS subsystem inside containment. Each subsystem consists of one approximately 50% capacity spray pump, one spray cooler, one 180° coverage spray header, nozzles, valves, piping, instrumentation, and controls. Each outside RS subsystem also includes a casing cooling pump with its own valves, piping, instrumentation, and controls.

The two outside RS subsystems' spray pumps are located outside containment and the two inside RS subsystems' spray pumps are located inside containment. Each RS train (one inside and one outside RS subsystem) is powered from a separate Engineered Safety Features (ESF) bus. Each train of the RS System provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission product removal. Two spray pumps are required to provide 360° of containment spray coverage assumed in the accident analysis. One train of RS or two outside RS subsystems will provide the containment spray coverage and required flow.

The two casing cooling pumps and common casing cooling tank are designed to increase the net positive suction head (NPSH) available to the outside RS pumps by injecting cold water into the suction of the spray pumps. They are also beneficial to the containment depressurization analysis. The casing cooling tank contains at least 116,500 gal of chilled and borated water. Each casing cooling pump supplies one outside spray pump with cold borated water from the casing of the outside RS subsystems. Each casing cooling pump is powered from a separate ESF bus.

The inside RS subsystem pump NPSH is increased by reducing the temperature of the water at the pump suction. Flow is diverted from the QS system to the suction of the inside RS pump on the same safety train as the quench spray pump supplying the water.

The RS System provides a spray of subcooled water into the upper regions of containment to reduce the containment pressure and temperature during a DBA. Upon receipt of a High-High containment pressure signal, the two casing cooling pumps start, the casing cooling discharge valves open, and the RS pump suction and discharge valves receive an open signal to assure the valves are open. Refueling water storage tank (RWST) Level–Low coincident with Containment Pressure–High High provides the automatic start signal for the inside RS and outside RS pumps. Once the coincidence logic is satisfied, the outside RS pumps start immediately and the inside RS pumps start after a 120-second delay. The delay time is sufficient to avoid simultaneous starting of the RS pumps on the same emergency diesel generator. The coincident trip ensures that adequate water inventory is present in the containment sump to meet the RS sump strainer functional requirements following a loss of coolant accident (LOCA). The RS system is not required for steam line break (SLB) mitigation. The RS pumps take suction from the containment sump and discharge through their respective spray coolers to the spray headers and into the containment atmosphere. Heat is transferred from the containment sump water to service water in the spray coolers.

The Chemical Addition System supplies a sodium hydroxide (NaOH) solution to the RWST water supplied to the suction of the QS System pumps. The NaOH added to the QS System spray ensures an alkaline pH for the solution recirculated in the containment sump. The resulting alkaline pH of the RS spray (pumped from the sump) enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The RS System is a containment ESF system. It is designed to ensure that the heat removal capability required during the post accident period can be attained. Operation of the QS and RS systems provides the required heat removal capability to limit post accident conditions to less than the containment design values and depressurize the containment structure to less than 2.0 psig in 1 hour and to subatmospheric pressure within 6 hours following a DBA.

The RS System limits the temperature and pressure that could be expected following a DBA and ensures that containment leakage is maintained consistent with the accident analysis.

Appendix E

Detailed descriptions of the Service Water (SW) System are provided in the form of excerpts from the North Anna Technical Specification Bases.

B 3.7 PLANT SYSTEMS

B 3.7.8 Service Water (SW) System

BACKGROUND

The SW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the SW System also provides this function for various safety-related and non-safety-related components. The safety-related function is covered by this LCO.

The SW System is common to Units 1 and 2 and is designed for the simultaneous operation of various subsystems and components of both units. The source of cooling water for the SW System is the Service Water Reservoir. The SW System consists of two loops and components that can be aligned to operate on either loop. There are four main SW pumps taking suction on the Service Water Reservoir, supplying various components through the supply headers, and then returning to the Service Water Reservoir through the return headers.

Eight spray arrays are available to provide cooling to the service water, as well as two winter bypass lines. The isolation valves on the spray array lines automatically open, and the isolation valves on the winter bypass lines automatically shut, following receipt of a Safety Injection signal. The main SW pumps are powered from the four emergency buses (two from each unit). There are also two auxiliary SW pumps which take suction on North Anna Reservoir and discharge to the supply header. When the auxiliary SW pumps are in service, the return header may be redirected to waste heat treatment facility if desired.

However, the auxiliary SW pumps are strictly a backup to the normal arrangement and are not credited in the analysis for a DBA. During a design basis loss of coolant accident (LOCA) concurrent with a loss of offsite power to both units, one SW loop will provide sufficient cooling to supply post-LOCA loads on one unit and shutdown and cooldown loads on the other unit. During a DBA, the two SW loops are cross-connected at the recirculation spray (RS) heat exchanger supply and return headers of the accident unit.

On a Safety Injection (SI) signal on either unit, all four main SW pumps start and the system is aligned for Service Water Reservoir spray operation. On a containment high-high pressure signal the accident unit's Component Cooling (CC) heat exchangers are isolated from the SW System and its RS heat exchangers are placed into service. All safety-related systems or components requiring cooling during an accident are cooled by the SW System, including the RS heat exchangers, main control room air conditioning condensers, and charging pump lubricating oil and gearbox coolers.

The SW System also provides cooling to the instrument air compressors, which are not safety-related, and the non-accident unit's CC heat exchangers, and serves as a backup water supply to the Auxiliary Feedwater System, the spent fuel pool coolers, and the containment recirculation air cooling coils. The SW System has sufficient redundancy to withstand a single failure, including the failure of an emergency diesel generator on the affected unit.

Additional information about the design and operation of the SW System, along with a list of the components served, is presented in the UFSAR, Section 9.2.1 (Ref. 1). The principal safety related function of the SW System is the removal of decay heat from the reactor following a DBA via the RS System.

Appendix F

Detailed descriptions of the Power Operated Relief Valves (PORVs) are provided in the form of excerpts from the North Anna Technical Specification Bases.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are air or nitrogen operated valves that are controlled to open at a set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by unit operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORVs, their block valves, and their controls are powered from the emergency buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. The PORVs are air operated valves and normally are provided motive force by the Instrument Air System. A backup, nitrogen supply for the PORVs is also available. Two PORVs and their associated block valves are powered from two separate safety trains.

The unit has two PORVs, each having a relief capacity of 210,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure—High reactor trip setpoint following a step reduction of 50% of full load with steam dump. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

Appendix G

Detailed descriptions of the Residual Heat Removal (RHR) are provided in the form of excerpts from the North Anna Technical Specification Bases.

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS) to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

Attachment 5

Supporting Documentation
(Probabilistic Risk Analysis)

North Anna Power Station Unit 1
Virginia Electric and Power Company
(Dominion)

Table of Contents

SUMMARY OF CHANGES

Revision	Author	Summary
0		Initial issuance.
1	A. Afzali	In this revision the fire impact assessment has been updated and a list of all Reg. Guide 1.200 Rev. 1 GAPs is provided. Also, a number of additional clarifications have been made to the entire document. Note that significant changes have been made but no revision bars are used.
2	A. Afzali	In this revision, a couple of clarifications (in particular to the Tier 2 section) were made.

1.0 PURPOSE

This PRA notebook documents the required analyses to support a proposed *Technical Specifications Change Request* (TSCR) for the North Anna Power Station (NAPS). NAPS Licensing has requested PRA support to justify a one-time 72-hour *Completion Time* (CT) for the 480 VAC *Motor Control Centers* (MCC) 1-EE-MCC-1J1-2N and -2S. These MCCs will be denoted as 1-EE-MCC-1J1 in this document. The proposed CT will be used for damage inspection following the failure of a breaker that is powered by the 1J1 bus.

For the internal events and internal flooding hazards, the current North Anna PRA model N105A [NB 01] with a couple of minor modifications is used to evaluate the impact on the pertinent regulatory figures of merit for the proposed Technical Specifications Change Request. Attachment B provides additional information about modeling changes that were made to the base N105A model to perform the risk calculations. The N105A model contains complete and updated logic for the risk assessment of internal events and flooding. An assessment of the external events, including fire, seismic and tornado risk, is also included in this notebook.

2.0 INTRODUCTION

This notebook documents the analyses in accordance with the requirements of Regulatory Guides 1.174 [RG 01] and 1.177 [RG 02]. The RG 1.174 evaluation estimates an average annual increase in CDF/LERF risk due to this single *Completion Time* (CT) entry of 72 hours for the MCC. As a conservative measure, although it is expected that the actual 1J1 MCC's out of service time to be less than 72 hours, for the CDF/LERF calculations, the entire 72 hour potential exposure time is used. The RG 1.177 analysis estimates the *Incremental Conditional Core Damage Probability* and *Incremental Conditional Large Early Release Probability* (ICCDP/ICLERP) figures of merit for a single CT entry.

As required by [RG 01] and [RG 02], a three-tiered approach has been used to evaluate the risk associated with the proposed Technical Specifications (TS) *Completion Time* (CT). These tiers evaluate the risk impact of the MCC outage, the risk impact of other equipment unavailability concurrent with the MCC outage, and the availability of a *Configuration Risk Management Program.* These requirements are addressed in detail in the following sections.

3.0 ANALYSIS

3.1 Inputs

This analysis uses the North Anna PRA internal events and flooding model, N105A that was developed in [NB 01]. The analysis also includes an evaluation of the fire, seismic and tornado hazards. The fire and seismic analyses have made use of the *Independent Plant Examination-External Events* (IPEEE) information.

The proposed TSCR is being developed for an at-power emergency bus outage, eliminating the need for evaluation of other operating modes.

3.2 Assumptions

1. All the assumptions and assertions made in developing the N105A model, other than the maintenance-induced unavailability of the 1J1 MCC, apply to this evaluation.
2. All assumptions and assertions made in developing the IPEEE fire PRA model, other than the ones explicitly modified in this calculation, apply to this evaluation.

3.3 Method of Tier 1 Analysis

Regulatory Guide 1.174 [RG 01] and Regulatory Guide 1.177 [RG 02] are the applicable regulatory guides for preparation of the risk assessment.

Regulatory Guide 1.174 provides guidance on developing a risk-informed licensing submittal. This document classifies potential increases in *Core Damage Frequency* (ΔCDF) and *Large, Early Release Frequency* (ΔLERF).

Regulatory Guide 1.177 provides additional guidance specific to risk-informed Technical Specification changes. This Regulatory Guide provides guidance on the acceptable *Incremental Conditional Core Damage Probability* (ICCDP) and *Incremental Conditional Large, Early Release Probability* (ICLERP). These risk metrics are the result when a risk increase, defined as the frequency of core damage or large radionuclide release per year, are integrated over the time of the proposed Technical Specifications *Completion Time* (CT). The thresholds for ICCDP and ICLERP in Regulatory Guide 1.177 are 5E-7 and 5E-8, respectively.

The effect on risk of the proposed increase in *Completion Time* (CT) for restoration of the 1J1 MCC has been evaluated using the NRC's three-tier approach suggested in RG 1.177:

Tier 1 – PRA Capability and Insights,
Tier 2 - Avoidance of Risk-Significant Plant Configurations, and
Tier 3 - Risk-Informed Configuration Risk Management

Although RG 1.177 requires the evaluation of the proposed change on the total risk (i.e., on-line and shutdown risk), this evaluation only quantifies the on-line risk. This is appropriate since the shutdown risk will not be impacted as a result of the proposed change. For this one time extension, the maintenance will take place while the unit is online and not during shutdown.

In Tier 1, the impact of the proposed TS change on the figures of merit (CDF, ICCDP, LERF, and ICLERP) are assessed by considering (1) the validity of the model and (2) the PRA insights and findings.

3.4 PRA Model Applicability and Quality

The latest PRA model, N105A, has been used to analyze the risk of the proposed TSCR. The N105A model, which evaluates internal events and flooding, was released in March, 2007. The internal events (including the internal flooding hazard) PRA model is maintained and updated under the PRA configuration control program in accordance with Dominion procedures. Plant changes, including physical and procedural modifications as well as changes in performance data, are reviewed for applicability and the PRA is updated to reflect such changes on a regular schedule by qualified personnel, with independent reviews and approvals.

In order to verify and improve the quality of the North Anna PRA model, an independent review of the NAPS internal events at power was performed in 2001 by the *Westinghouse Owner's Group* (WOG). The peer review is documented in the Westinghouse PRA peer review report [REPORT 05]. All of the "A" and "B" *Findings and Observations* (F&Os) have either been resolved or found to have no impact upon the proposed Technical Specification (TS) change. Documentation of the resolution of the B-significance F&Os is provided in PRA Notebook volume [NB 03].

Additionally, the NAPS PRA model was also subjected to a self assessment against Capability Category II requirements of the ASME Standard for PRA, including Addendum B [STD 03] and Regulatory Guide (RG) 1.200 Revision 1 [RG 03]. The review was conducted by a team of experts with experience in performing NEI PRA Certifications and ASME PRA Standard Reviews. The scope of this assessment was to compare the current PRA model against ASME standard RA-S-2002 (including RA-Sa-2003 and RA-Sb-2005) to determine if each of the requirements of Capability Category II had been met and sufficiently documented. The approach of the assessment was to develop a comprehensive list of all potential areas for improvement and to be aggressive in pursuing model enhancement by conservatively characterizing a Supporting Requirement (SR) as "Not Met" if one or more areas for improvement were identified. This conservative philosophy is different than that which is used for PRA model peer reviews that are performed in accordance with NEI 05-04, Revision 2, where "findings" and "suggestions" are used to characterize such observations. Using this conservative philosophy, although the preponderance of evidence point to meeting the applicable SR at Category II level, the assessment characterized a number of SRs as not meeting Capability Category II requirements. Based on a review of the findings and suggestions listed in the assessment many of the instances where a SR was indicated as "Not Met" could be characterized as a "suggestion" using the guidance in NEI 05-04.

To support the Bus 1J1 CT extension, consistent with the industry practices, the "Not Met" SRs were reviewed to:

- Identify those unmet SRs that do not have an impact on the risk insights provided in support of this application (e.g., documentation-only issues).
- Identify potential sensitivity studies that can be performed to ensure that the risk insights are not significantly affected by the "Not Met" findings.

As a result of this review, the following conclusions were reached:

i. A significant number of unmet SR issues pertained to documentation only. Enhancements to the documentation would not change the model and, therefore, would have no impact on the analysis performed in support of this application.

ii. A number of unmet SRs related to initiating event identification. For example, events related to the process used to identify plant systems that have the potential to cause an initiating event. However, although new initiating events may be identified, based on the experience with dealing with this comment, it is judged that 1) the accident progression for these potential initiating events is similar to the progression for initiating events already included in the model and 2) the frequency of these newly identified potential initiating events is lower than the existing initiating event frequencies. Therefore, the impact on this analysis is negligible.

iii. A number of additional unmet SRs pertained to the Accident Sequence (AS) element. One issue that resulted in characterizing an AS related SR as not meeting Capability Category II is that the basis for some system success criteria is not documented and that, as a result of developing the documentation, changes could occur. No expected changes or outliers were identified, so resolution of this item likely would not impact the analysis results. A few items related to the completeness of accident sequence modeling, but these items were for insignificant sequences, e.g., ATWS after a LOCA. The last item was that sources of uncertainty were not documented. Based on the discussion above, it is not expected that resolving the unmet SRs for the AS element with the potential for model changes would alter the findings of this analysis.

iv. The overall quality of a High Level Requirement (HLR) was found to be more than adequate for this one time application. For example, out of the 22 SRs related to AS HLR, one was found not to be applicable, 9 SRs were found to meet capability Category II, and 9 out of the 12 unmet SRs were found to be documentation related. One of the three unmet SR pertained to the uncertainty evaluation, and the other two unmet SRs pertain to specific examples of area of improvements and not necessarily indicated a systemic problem.

v. A few unmet SRs were assessed to have no impact on the CDF/LERF estimate. For example, SC-B1 SR is characterized as "Not-Met" because, in reviewer's opinion, data used to develop the success criteria for seal LOCA and Offsite Power recovery appears to be dated. The "Not Met" characterization seems to be overly conservative because the reviewers also found that "the resulting success criteria seem to be reasonable as compared to those used in other plants." Additionally, the reviewer's also stated that, "it is not certain if the fault tree models themselves may have more up-to-date models than the documentation indicates." Another example is DA-C12, where although unavailability data is based on plant-specific data and is documented in the appropriate notebook, the SR is characterized as unmet because it was not clear to the reviewers that both units' data was being used to compute unavailability. Also, the reviewers concluded that using a floor value of 1.0E-6 that was used for components that are not expected to be taken out of service and which had no observed unavailability is too low. This particular issue does not have any impact on this one time CT extension since maintenance activities on all other risk significant components is prohibited.

vi. A number of SRs were characterized as not met due the same apparent cause. For example, although the reviewers found that the intent of SY-A17 and SY-A19 were generally met, these SRs were characterized as not meeting capability category II because load sequencing for the diesels is not included in the model, nor are there any assumptions or referenced calculations pertaining to load sequencing. The impact of this potential omission on this one time extension of CT for 1J1 MCC is judged to be negligible. Similar discussion also applies to SRs SY-A11 and SY-A13, where inadvertent SI actuation was judged to be inadequately modeled.

vii. Certain unmet SRs related to identification, screening, and modeling of pre-initiator operator errors. Numerous pre-initiator operator errors are included in the PRA model. Although a rigorous analysis of such events could result in the identification of additional items, pre-initiator operator errors are typically not important to the overall PRA results so it is not expected that resolving the unmet SRs for the pre-initiator HR element with the potential for model changes would alter the findings of this one time CT extension. Additionally, any change in the risk estimation would impact the base case as well as the 1J1 case. Therefore, the overall risk-insights with respect to the change in risk for this one time extension of the CT are judged to be unchanged.

viii. A number of unmet SRs related to post-initiator operator actions. None of these items noted any major weaknesses, so it is not expected that resolving the unmet SRs for the post-initiator HR element with the potential for model changes would alter the findings of this analysis. Additionally, similar to the pre-initiator SRs, any change on the risk estimation would impact the base case as well as the 1J1 case. Therefore, the overall risk-insights with respect to the change in risk for this one time extension of the CT are judged to remain unchanged.

ix. Unmet items related to internal flooding are either due to documentation or have the potential to equally impact the base case and the 1J1 LAR case, with no impact on the delta calculations. Therefore, they are not expected to impact the risk insights for the proposed one time extension of CT for 1J1 MCC.

Based on the discussion presented above, and the considerable effort made to incorporate the latest industry insights into the PRA as well as results of self-assessments and Peer Reviews, Dominion is confident that the current NAPS internal events PRA model meets the expectations for PRA technical adequacy for this one-time CT extension.

As stated in section 3.1, this analysis uses the NAPS IPEEE study to provide an estimate of the change in the external events risk due to the proposed one-time CT extension. The IPEEE external events models (specifically, models for the Fire, Seismic, and Tornado hazards) have not been updated since 1994. Although, the NRC found that the NAPS fire submittal met the intent of the IPEEE process, for this analysis a conservative approach is adopted to compensate for usage of an outdated external events PRA model as well as potential optimisms and uncertainties in the analysis. As a result of using these conservatisms for the fire analysis, the estimated increase in fire CDF, due to the unavailability of the 1J1 MCC, is almost a factor 2 greater than the IPEEE base case fire CDF. As stated in section 3.7.2 of this notebook, this significant increase in fire risk is highly conservative on the basis that:

- In most cases (from the quantification point of view), other than use of the no-maintenance model in the screening stage, no credit is taken for the fact that this is a one-time extension when the work will be performed with plant configuration known and certain compensatory measures (such a use of fire watch) in place.
- From the fire hazard point of view and postulated initiators, the availability/reliability of the 2nd heat removal function is an important factor. The availability/reliability of the cooling function provided by components supported by the 1J1 MCC (i.e., the TDAFW or MDAFW pump 3B) are not significantly impacted by the unavailability of the 1J1 MCC because the MOVs on the flow path are normally open, kept open, and will be open

and de-energized. Also, these valves can manually be operated locally, thus providing or isolating flow to the SGs.

- Similar to the internal flooding hazard, due to the plant physical configuration and location of Safe Shutdown equipment and their associated cables, the significant contributors to the internal fire risk include those scenarios that have the potential to disable redundant components performing the same function. Therefore, the impact of unavailability of one train is not as consequential.

Therefore, Dominion is confident that, due to the process used in this evaluation, the use of the out-dated IPEEE external model has not resulted in an underestimation of the delta risk increase.

3.5 Internal Events and Flooding Analysis

This analysis used the zero-maintenance N105A model (referred to here as N105A-TM0), with 1-RC-MOV-1536 failed closed to reflect the plant condition as of Rev. 0 of this notebook, as its base case. Attachment B provides additional information about modeling changes that were made to the base N105A model to perform the risk calculations. The results are documented in Section 4.0. Note that, since the completion of the Rev. 0 of this notebook, 1-RC-MOV-1536 is returned to the normal at power configuration (i.e., open) and is available to perform its function. However, the model used for estimating the pertinent figures of merit (e.g., CCDPs and CDFs values), still assumes that the valve is closed. Using this slightly conservative model does not result in a change in the risk insights for this application since the impact of the valve closure on the results is negligible.

The dominant accident sequences were reviewed for the case with the 1J1 480 VAC MCC unavailable. The results are as follows.

- The *Steam Generator Tube Rupture* (SGTR) initiating event contributes 64% (1.0E-6/yr) of the *Large Early Release Frequency* (LERF) with the 1J1 MCC unavailable. This configuration is LERF-limiting, rather than CDF-limiting, because the SGTR is a containment bypass event.
  - The most limiting SGTR sequences include a failure to cool down and depressurize the RCS. This sequence contributes 21% (3.3E-7/yr) to LERF and is independent of the MCC outage.
  - The next two most limiting SGTR sequences include failures of the High Head Safety Injection (HHSI) system and failure of feedwater isolation. These sequences total 21% (3.3E-7/yr) of LERF. The HHSI failures occurred due to the MCC outage on one train and coincident, random failures on the opposite train.
  - Several SGTR sequences include failure of the Residual Heat Removal (RHR) system due to the tagout of the 1J1 MCC. This bus powers the normally-closed isolation valve 1-RH-MOV-1701 from the Reactor Coolant System to the RHR system inside containment. If recovery of the MOV is unsuccessful, then decay heat is removed via the secondary system and any secondary faults will result in an offsite release. These sequences total 13% (2.0E-7/yr) of LERF.

- The most limiting LERF sequence which is not a SGTR is the vessel rupture, contributing 18% (2.7E-7/yr) to LERF. This sequence is independent of the proposed MCC *Completion Time*.

There are no other sequences involving the MCC tagout that contribute more than 5% to the overall LERF.

3.6 Common Cause Issues

No common cause analysis has been performed. The original breaker failure has to date not revealed any characteristics of common cause concerns. The proposed CT is strictly a precautionary measure to inspect for potential damage to surrounding equipment. There is no evidence of potential common cause vulnerability and thus none has been modeled.

The current PRA model does not include any common cause faults in the emergency electrical power distribution system, other than those associated with the diesel generators.

3.7 Fire Analysis

[REPORT 02] documented the original IPEEE fire analysis for North Anna. It screened out all but four areas as insignificant contributors to core damage risk. The NAPS fire PRA model was developed using the following approach:

Fire areas of potential risk significance were identified using the initial qualitative and quantitative screening steps defined in the FIVE methodology [REPORT 07] document.

Those fire areas which did not screen out were subject to detailed modeling described in various procedure guides such as NUREG-2300 [REPORT 08], NUREG-2815 [REPORT 09] or NSAC-181 [REPORT 10]. The COMPBRN IIIe code [REPORT 11] was used for all deterministic modeling of intra-area fire propagation. Inter-area fire propagation analysis was not required based on the review of the fire area boundaries performed to address the Fire Risk Scoping Study, NUREG/CR-5088 [REPORT 12] issues.

Fire frequencies in particular locations accounted for both generic experience (US plant experience obtained from the EPRI Fire Event Data Base) and area specific fixed ignition sources. The contribution of transient fuels and sources was accounted for by addressing plant specific procedures for the control of combustibles and ignition sources, as well as for periodic inspections for transients.

No credit was taken in the analysis for the detection and suppression of fires (i.e., fires were allowed to burn until they self extinguished).

Fire Risk Scoping Study Issues were addressed through specifically tailored walkdowns as defined in the FIVE methodology, including seismic fire interactions, effects of fire suppressant on safety related equipment, fire barrier effectiveness and control systems interactions.

3.7.1 Approach for Assessing Change in Fire-Hazard-Induced Risk due to the proposed CT Extension

The current NAPS fire PRA model, which was developed in support of the Individual Plant Examination for External Events (IPEEE) study [REPORT 02], is a vulnerability fire PRA model. Therefore, it contains a number of assumptions and assertions that are meant to effectively, yet, quickly identify areas of vulnerability. Also, the quantification part of the IPEEE fire PRA model used the appropriately modified average internal events PRA model to quantify the risk of the postulated damage. In this application of the fire PRA model, to reflect the plant configuration during the proposed application, the assumptions and assertions of the IPEEE fire model are reviewed and adjusted, if necessary, and the case specific internal events PRA model (i.e., no-maintenance model with MCC 1J1 unavailable "N105A-TM0") is ran to quantify the risk. The approach used in this evaluation is sometimes over conservative. This conservative approach was used to ensure that any optimism that may have crept in due to the use of the semi-qualitative nature of this assessment is well compensated for by the conservatisms included in other sections. Since this is not a risk-ranking study (i.e., the relative risk is not important), the use of conservatisms in some areas to compensate for potential non-conservatism in other parts is appropriate.

The following steps were followed to assess the potential change in the fire-hazard-induced risk due to the proposed CT extension:

a) The qualitatively and quantitatively screened fire areas were reviewed to assess whether the basis for screening would change due to unavailability of the 1J1 MCC.
b) Those screened fire areas that were found to have higher CCDP value were re-analyzed to determine their contribution to the CDF figure of merit, given the proposed plant configuration (i.e., the average base model was not used since this is a one-time extension).
c) The contribution for those areas that were retained in the base case analysis for detailed analysis, were re-evaluated to assess any potential increase in CDF.
d) The delta Fire CDF so calculated was added to the internal events delta CDF to calculate the impact of 1J1 MCC being out-of-service on the CDF figure of merit.
e) The delta fire CDF so calculated, in combination with a mixture of qualitative and quantitative evaluation of the containment performance, are used to quantify the increase in the LERF figure of merit.

3.7.1.1 Qualitative Screened Fire Areas

In the IPEEE study, this screening was performed in several steps. In the first step all plant areas which did not contain any susceptible Appendix R shutdown equipment (in any of their compartments) were screened out. Next, for the areas remaining, the requirement, or not, for a plant shutdown was determined assuming that 1) all Appendix R safe shutdown equipment and cables in a given area (including all compartments) are damaged **and** 2) the normal alternate shutdown path (as defined within the Appendix R framework) is unavailable. **A demand for shutdown was assumed** unless it could be shown with confidence that the fire would not cause an automatic trip or plant operating conditions or Technical Specifications would not require a shutdown within 8 hours.

If the fire does not create a demand for safe shutdown using the equipment assumed to be unavailable or damaged by the fire, then the fire area and all its compartments were screened out. (Note: It was not necessary to assume a loss of offsite power as is the case in Appendix R studies, unless there is some potential for the postulated fire inducing such an event as identified in step 3).

For this analysis, if appropriate and desirable, the second criterion (i.e., the criterion that the normal alternate shutdown path (as defined within the Appendix R framework) is unavailable) is adjusted as follows:

a) given the train supported by 1J1 MCC is unavailable and
b) the train supported by the H bus is available

Crediting condition b (i.e., crediting availability of the train supported by the H bus) is judged to be appropriate for this application because prior to performing the work on the 1J1 MCC, the availability of the redundant train (i.e., the train supported by the 1H bus) will be verified.

It is also noted that in a later revision of the FIVE methodology [REPORT 13] fire areas or compartments should not be screened out unless it can be shown that Appendix R equipment is not damaged <u>and</u> there is no demand for shutdown. This revision was issued several months after the NAPS screening analysis was completed. For the original NAPS analysis the impact of the changes is that several fire areas eliminated in the qualitative screening analysis would require further evaluation. The IPEEE study, however, concluded that these fire areas would have been eliminated in the quantitative screening analysis phase. As a result, the number of areas requiring detailed analysis would not change. This conclusion was substantially verified in the fire analysis which was performed for Surry under the revised rules. For this analysis, since all the screened fire areas were re-evaluated, the original NAPS IPEEE qualitative screening criteria did not have a significant impact.

Based on a review of the qualitative analysis [NB 05], the following evaluation has been performed:

- A number of fire areas were qualitatively screened out because they did not contain any safe shutdown equipment AND fire in these areas would not result in a forced plant shutdown. Such areas included Turbine Building Lube Oil Room (Fire compartment TB-LOR), Service Building East (Fire zone Z-34), Service Building West (Z-35), Service Building Stairwell (Zone 54), Technical Support Center (Zone 46A), and Auxiliary Building Boiler Room (Zone 22). For this application, it is concluded that the contribution of these areas to the fire-induced risk would not change since the safe shutdown components supported by 1J1 MCC would not be challenged.
- A number of areas were qualitatively screened out because they did not contain any Safe Shutdown components. However, fire in these areas could result in a forced shutdown (manual/automotive trip or forced shutdown within 8 hours). These areas included Unit 1 Normal Switchgear Room (Fire area 5-1, fire compartment NSR-1) and Unit 1 Motor Generator Set house

(Fire Area 3-1, Z-27-1, MGSH-1), where a loss of offsite power and a reactor trip initiating events can be postulated, respectively. For such areas, to calculate the change in the fire induced CDF, the fire frequency for the major fixed ignition sources in the area (e.g., fire frequency for electrical cabinets in case of fire area 5-1 and fire frequency for MG Set for fire area 3-1) is multiplied by the CCDP estimated for the postulated initiating event with the failure probabilities for the 1J1 MCC and any fire-impacted component set to 1.0. Note that, due to relatively low impact of the 1J1 MCC on the fire safe shutdown capability, this approach for estimating the delta CDF is highly conservative because the base case fire-induced CDF is assumed to be negligible. Also, disregarding the fire frequency from the other potential ignition sources in each area is justified since, the duration of the 1J1 MCC unavailability is limited and no significant maintenance activities will be performed.

- A number of areas were screened out because, although the area contained one or more safe shutdown components, the potential for a plant trip could be ruled out. In this application, the change in the fire risk due to the proposed CT extension is calculated as follows:

  - For areas where the affected safe shutdown equipment of concern are supported by the 1J bus, the change in risk is considered to be negligible since the 1J1 MCC outage would not add any additional risk (that is the equipment in these areas are assumed damaged by fire and availability or unavailability of the 1J1 MCC is considered to be irrelevant). In some cases (e.g., fire area 9B-1 (EDG 1J compartment), this consideration may be an oversimplification. However, this oversimplification is judged to be justified on the basis that the risk from fires in such areas is dominated by the fire-induced loss of safe shutdown components in the area. Additionally, based on a review of the final rescreening results, this potential oversimplification did not result in a change in the results, since the fire areas containing the H-supported train components remained screened out (e.g., the 1H EDG compartment remained screened out).
  - For areas where the affected safe shutdown equipment of concern is supported by the 1H bus, the risk is calculated by multiplying the fire frequency by the appropriate CCDP, unless it could be clearly shown that the impact of the 1J1 MCC unavailability on the fire-induced risk for the area is negligible. If quantification was deemed to be necessary, then the fire frequency is set to be equal to the ignition frequency for the major fixed ignition sources in the area. The CCDP is calculated using the "no maintenance" model with the following modifications:
    - guaranteed failure of the 1J1 MCC and other fire vulnerable components,
    - the frequency for the postulated initiating event set to 1.0, and
    - the frequencies for the other initiating events in the model set to zero.

- A number of areas were qualitatively screened out on the basis that at least two alternate shutdown paths were available following a fire in such areas. For example, Emergency Diesel Generator (EDG) Room 1H was screened out since the offsite power and 1J EDG were unaffected by the fire. Another example is that a major part of the Turbine Building was screened out because all EDGs and at least two Motor Driven Auxiliary Feed Water pumps were unaffected by a postulated fire in this area. In this application, the change in the contribution of fires in such area is evaluated as follows:

- For areas where the affected safe shutdown of concern are supported by the 1J bus, the change in risk is considered to be negligible since the 1J1 MCC outage would not add any additional risk (that is the equipment in these areas are assumed damaged by fire and availability or unavailability of the 1J1 MCC is considered to be irrelevant). For example, the change in fire risk for the 1J EDG room is considered to be negligible. Again, this oversimplification is judged to be justified on the basis that the risk from fires in such areas is dominated by the fire-induced loss of safe shutdown components in the area. Additionally, based on a review of the final rescreening results, this potential oversimplification did not result in a change in the results, since the fire areas containing the H-supported train components remained screened out (e.g., the 1H EDG compartment remained screened out). For areas where the affected safe shutdown equipment of concern are supported by the 1H bus, the risk is calculated by multiplying the fire frequency for the major fixed ignition sources in the area with the CCDP for an appropriate surrogate initiating event. Note that the CCDP is estimated using the "no maintenance" model with the failure probabilities for the 1J1 MCC and fire vulnerable/impacted components in the area set to 1.0.

Note that for performing the above calculations, the internal events model is used (i.e., a new Fire PRA model is not developed). The major limitation of using the internal events model is that the human error probability (HEP) estimates used in the internal events model may not be appropriate for the fire scenario of concern. The major concerns with the appropriateness of HEP estimates, and the way these concerns are addressed in this evaluation, are as follows:

1) The fire event may present additional stress/distraction for Operators that may not be present for a similar initiating event in the internal events model. To address this concern, the CCDPs values for the fire scenarios are increased by a factor of 2 or 5, depending on the severity of the postulated damage. Increasing the CCDP by a factor of 2 or 5 is conservative since an increase in failure probability of one Operator action for one initiating event is not usually expected to increase the CCDP significantly.
2) The fire event may present plant physical conditions that may prevent Operators from performing the credited operator action. To address this concern, a list of Operator actions that in the internal events analysis are credited to be performed outside of the areas that were retained in the original fire PRA analysis for detailed evaluation (i.e., the control room, the emergency switchgear room, general portion of the Auxiliary Building, and the cable/vault tunnel) is obtained. The success of these operator actions would be set to zero (i.e., the failure probability will be set to 1.0), if the postulated fire event is evaluated to prevent the action from taking place.

It should be noted that in the IPEEE study, the circuits for the automatic actuation of components were not traced and as a result operator actions were credited to manipulate those valves that required to change state in response to an initiating event.

Table 3.7.1-1 is a summary of the analysis performed to address areas/compartments that were qualitatively screened from further evaluation in the IPEEE. It includes all fire areas and fire compartments chosen during the qualitative analysis. This table uses the following information for evaluating the potential contribution of fires in each area to core damage frequency.

Turbine Trip Initiating Event

From run MCC 1J1-TNORM.EQP CCDP contribution given guaranteed failure of 1J1 MCC = 4.291E-7
It is recognized that the above CCDP is based on the internal events model. As such failure probabilities of some operator actions (HEPs), credited in the internal events model, may be higher for a similar fire induced initiating event. However, since the postulated initiating event would not present a significantly different challenge to the operators, independent of its cause of occurrence, the HEP estimates would not increase significantly. Nevertheless, as a conservative measure, the CCDP value is increased by a factor of 2 to account for a potential additional fire-event-induced stress. Also, note that the fire-induced initiating event frequency assigned to each area is the total fire frequency for the significant fixed ignition sources in the area and no credit for the severity factor is applied. This is an additional conservatism.

Therefore, CCDP for fires is conservatively estimated as 8.6E-7 which is slightly greater than twice the estimated CCDP from the internal events model.

LOOP initiating Event

From run MCC1J1-T1.EQP, CCDP given a LOOP initiating event and guaranteed failure of 1J1 MCC = 1.881E-5
Similar to the Turbine Trip case above, it is recognized that the CCDP is based on the internal events model. As such failure probabilities of some operator actions (HEPs), credited in the internal events model, may be higher for a similar fire induced initiating event. Similarly, however, since the postulated initiating event would not present a significantly different challenge to the operators, independent of its cause of occurrence, the HEP estimates would not increase significantly. Nevertheless, as a conservative measure, the CCDP value is increased by a factor of 5 in this case to account for a potential additional fire-event-induced stress in combination with a LOOP event. Also, note that the fire-induced initiating event frequency assigned to each area is the total fire frequency for the significant fixed ignition sources in the area and no credit for the severity factor is applied. This is an additional conservatism.

Therefore, CCDP for fires is conservatively estimated as = 6.0E-5

Results

Based on the approach shown above (which is a combination of the qualitative and quantitative screening), with the following plant configuration all but one of the IPEEE screened fire areas/compartments would screen out:

1. All accident mitigating systems and functions, other than those supported by the 1J1 MCC, are operable (i.e., the no-maintenance model is used)
2. 1J1 MCC is out of service

This is not unexpected since the contribution of 1J1 MCC being unavailable is well compensated for by restricting unavailability on all other accident mitigating components.

The general area of the Turbine Building is the only compartment that could not be screened out. The change in the contribution of this compartment to the fire risk due to 1J1 MCC being unavailable is estimated in the detailed analysis section of this notebook.

Note that based on a comparison of the local manual Operator actions credited in the internal events model (Attachment F) and the postulated damage in each area, it was concluded that none of the postulated scenarios would require the failure probability of the local manual operator action to be set to 1.0.

3.7.1.2 Quantitatively Screened Fire Areas

Based on a review of NAPS calculation file [NB 06], none of the fire areas retained after the qualitative analysis were quantitatively screened out.

3.7.1.3 Determination of Impact in the IPEEE Non-Screened Fire Areas

In the IPEEE study, detailed analysis was performed for four potentially significant fire areas which could not be eliminated as part of the qualitative and quantitative screening process. For these areas, section 6 of NAPS calculation file 5T45.NF/08.X (where X is 1, 2, 3, etc) series, which include documentation of the changes that were made to reflect the postulated damage in each one of the defined compartments in each unscreened fire area, was reviewed. The information available in section 6 includes the postulated initiating event and a list of basic events that were assigned certain failure probabilities (mostly 1.0). Based on a review of the basic event lists, the following criteria were used:

1. If the basic event representing the 1J1 MCC was assigned to fail as a consequence of a fire in a compartment, then the impact of the fire in that compartment was considered to be non-consequential on the proposed increase in the CT for 1J1 MCC.
2. If a basic event representing a component that supports the 1J1 MCC (e.g., 1J Bus) was assigned to fail as a consequence of a fire in a compartment, then the impact of the fire in that area was

considered to be non-consequential on the proposed increase in the CT for 1J1 MCC.

3. If neither of the above 2 conditions applied, then ideally the IPEEE fire PRA model would be solved twice. Once with all the basic events representing the maintenance-induced unavailability of the component supported by the H bus set to zero and the second time the above generated model with 1J1 MCC failure probability set to zero. The first run would be made to ensure that a more representative delta is calculated, since for this evaluation, the operability of the redundant train will be verified prior to performing work on the 1J1 MCC. However, based on a review of the IPEEE documentation and the state of the NAPS fire PRA model, it is judged that the best practical approach is to use the results of the IPEEE's detailed analysis to gain a conservative estimate of the risk increase.

Also, note that for the 1J1 MCC case, one additional area (general section of the Turbine Building) survived the qualitative screening. The contribution from this area is also further analyzed in this section of this notebook but in this case, since the postulated damage is limited, the current internal events model is used to calculate the delta.

3.7.1.3-1 Determination of Delta CDF in the Cable Vault &Tunnel

The IPEEE study separated the CV&T into three compartments: (1) Tunnel which includes the area outside of the ESGR labeled "cable vault" on most diagrams, (2) the Electrical Penetration Room (Elec Pen), which has also been called the cable vault area, and (3) Rod Drive Room.

Estimation of delta CDF

Based on a review of Table 4-1.3-1of [NB 07], "Fire Area/Compartment Safe Shutdown Equipment Detail Worksheet" for CV&T fire area, for an App. R fire, a number of Unit 2 components (e.g., Unit 2 Charging pumps) are relied upon to mitigate the consequences of a fire in this area. In the IPEEE fire PRA, this fire area was divided into a number of compartments and each compartment was separately analyzed. The analyses of impact of the 1J1 MCC unavailability on the risk for each compartment are discussed below.

Delta CDF for the Service Building CV&T Compartment

Based on a review of page 61 of [NB 07], this sub compartment does not contain significant number of components supported by the 1J1 MCC. Therefore, condition 3 of the above set of criteria applies. Two runs are made. The CCDP estimates for the base case, with 1J1 MCC available, and the 1J1 case, where 1J1 MCC is assumed unavailable, are estimated to be 2.26E-7 and 4.29E-7, respectively (See runs MCC1J1(A)-TNORM-HEPs_ETC.EQP and MCC1J1-TNORM-HEPs_ETC.EQP, respectively. Again, since the internal events PRA model is used for this evaluation, the CCDP values are multiplied by a factor of 5 to account for potential additional stress by the fire in this compartment. Therefore, the base case and 1J1 case CCDPs are 1.13E-6 and 2.15E-5, respectively and the delta CCDP is 1.01E-5. Based on a review of page 62, the initiating event frequency for this fire scenario is 1.56E-4 per year. Therefore, the change in CDF is 1.58E-9 per year.

Delta CDF for the Electrical Penetration Compartment

Based on a review of page 69 on [NB 07], condition 1 of the above set of rules applies (i.e., 1J1 MCC would be affected in this sub compartment). Therefore, there is no delta CDF contribution from this sub compartment.

Delta CDF for the Rod Drive Room Compartment

This compartment is physically located above the electrical penetration area. The compartment contains two 480V electrical buses 1H1 and 1J1. For the IPEEE study, this compartment was further subdivided into Control Rod Drive Room General Area (GA), Control Rod Drive Room 1H1 Bus area (1H1), Control Rod Drive Room 1J1 Bus area (1J1). The evaluation for each sub compartment is provided below:

- In the IPEEE study, the fires initiating in the general area do not contribute significantly to the CDF due to limited impact of the postulated fires on accident mitigating functions. For this evaluation, it is also concluded that the fires in the GA will not have a significant impact on the proposed CT extension on the basis of low fire frequency (1.9E-3 per year per page 81 of [NB 07]) and limited damage.
- In the IPEEE study, fires originating in the 1J1subcompartment were the most significant contributors to the CDF. For this analysis, the unavailability of the 1J1 MCC would not have an impact on the delta CDF.
- In the IPEEE study, fires originating in the 1H1 sub compartment were the 2nd most significant contributors to the CDF. Based on a review of the event trees for this area ([NB 07], page 86), the fire frequency for this sub compartment is 2.86E-4 per year and its contribution to CDF is 2.41E-7, which means the IPEEE CCDP for this area is 2.41E-7/2.86E-4 or 8.6E-4. Based on a review of page 85 of [NB 07], 1-EE-1J1-2 is amongst the components that are affected by a fire in this sub compartment. Also, based on a review of Table 6.4-1 and 6.4-2 of [NB 07], as well as Table 4-1.3-2 of [NB 05], the following observations are made
    - The major non-fire-induced failures which contribute to the CDF are loss of AFW and MFW functions.
    - Random failures or maintenance-induced unavailability of the TDAFW pump (AFW-P-2) and the 3B MDAFW pump are included amongst the top cutsets for fires originating in this sub compartment.
    - The Appendix R credits Unit 2 high head safety injection function in this area and this function is not affected by a fire in this sub compartment.

Based on these observations, it is concluded that the unavailability of the 1J1 MCC would not result in a significant increase in the conditional probability of core damage sequences in this area on the following bases:

- The availability/reliability of the cooling function provided by the TDAFW or MDAFW pump 3B are not significantly impacted by the unavailability of the 1J1 MCC because neither the pumps nor the flow path are affected. The pumps are not powered by the MCC. The MOVs on the flow path are normally open, kept open, and will be open and de-energized. Also, these valves can manually be operated locally, thus providing or isolating flow to the SGs ([NB 12], Table 1, note 2). Also, note that an Appendix R fire in the CV&T area is postulated to result in a loss of 1J1 MCC. Therefore, procedures are in place for this area to manually operate these valves in an event of fire in the CV&T area.
- The availability/reliability of the high head safety injection function is not significantly impacted because for fires in this sub compartment, the Unit 2 HHSI is credited. Again, as noted in the first bullet, an Appendix R fire in the CV&T area is postulated to result in a loss of 1J1 MCC. Therefore, the safe shutdown for this area already includes an evaluation of the HHSI flow path due to the unavailability of the 1J1 MCC. Although the fire PRA scenarios may assume a more limited damage and as such the 1J1 MCC unavailability may have an impact on the CDF estimate, it is judged that such scenarios are a subset of all events. Therefore, the CCDP multiplier and the verification of the availability of the alternate path compensates for any optimistic consideration with respect to the reliability of any manual action that may be credited for the HHSI flow path.

Based on the above observations, discussions and conclusions, it is asserted that the additional impact of the 1J1 MCC unavailability on the consequences of a fire in this sub compartment is limited. However, as a bounding analysis, it is conservatively assumed that the consequences can be represented by increasing the IPEEE CCDP estimate (calculated above as 8.6E-4) by a factor of 5. Therefore, the CCDP, CDF, and delta CDF for this sub compartment, with 1J1 MCC unavailable are estimated as 4.3E-3, 1.23E-6 per year, and 9.9E-7 per year, respectively.

Total Delta CDF

The total delta CDF is dominated by the delta CDF in the Control Rod Drive Room 1H1 Bus area (1H1), which is conservatively estimated as 9.9E-7 per year. It should be additionally noted that due to the work being done (which results in the 1J1 MCC series being de-energized) as well as fire watch compensatory measures which will be deployed in this area, the fire initiation frequency is smaller than that used in the IPEEE study.

3.7.1.3-2 Determination of Delta CDF in the ESGR

In the IPEEE study the switchgear room was divided into the following four compartments:

The 1H Switchgear Room (1H SWGR)- This room contains Emergency Switchgear, transformers, motor control centers, and cable race ways associated with Train H; battery Room 1-II is also located in this room. In addition there are minimal set of cable race ways (conduits only) associated with Train J, which are referred to as "crossover" raceways.

The 1J Switchgear Room (1J SWGR)- This compartment contains Emergency Switchgear, transformers, motor control centers, and cable race ways associated with Train J; battery Room 1-IV is also located in this room. In addition there are minimal set of cable race ways (conduits only) associated with Train H, which are referred to as "crossover" raceways.

The Instrument and Relay Rack Room (IRR)- This compartment contains solid state protection and relay logic cabinets associated with both Train H and Train J. The area also contains overhead train A and B cable trays and BOP under floor cable chases. The IRR also contains all process instrumentation cables which are used to monitor plant parameters and trip or ESF conditions.

The AC Room- This compartment contains the ESGR air conditioning units and related cables.

Based on the above descriptions, it is concluded that the proposed 1J1 MCC CT extension request has no impact on the risk originating from the fires originating in the 1J SWGR compartment.

Estimation of Delta CDF

Based on a review of [NB 08], page 266, in the IPEEE study it was estimated that the Emergency Switchgear Room has a fire-induced core damage frequency of 3.26E-6 per year. This represented 84% of the total Unit 1 fire CDF. The IRR alone contributes 63% of the total Unit 1 fire CDF. This result is not unexpected since almost all of the power station's equipment has a cable inside of this fire area. It should be noted that the Appendix R relies on the alternate safe shutdown capability using equipment outside this area. Therefore, although the fire risk in this area is relatively high, the 1J1 MCC availability does not have a significant impact on the fire-induced CCDP since most fires would either disable both trains or the 1J bus related power or cables.

Also, the IPEEE study concluded that reliability of the secondary heat removal function was the most important function in the ESWGR CDF sequences. Recovery of AFW or MFW included actions such as removing control circuit fuses from the MFW or AFW 4160 V breakers cubicles.

Based on a review of the top sequences for the ESGR fires ([NB 08] pages 267-279), description of these sequences provided on pages 280-291, and the credited components provided in Table 4-1.6-1 of [NB 05], page 42-45, the following observations are made:

1. Consistent with the IPEEE conclusions, fire induced reduction in the reliability of the AFW function is a major contributor to the CDF.
2. In sequences ranked 1st through 7th, and the 9th ranked sequence, either both MDAFW pumps or the 3B AFW pump are assumed to be damaged and core damage results. These sequences (all postulated as a result of a fire in the IRR sub compartment) contribute 47.2% of the total fire CDF from the ESWGR fire area.
3. The feed and bleed function, using HHSI system (which may be credited as a backup to the loss of AFW function) can be provided by the Unit 2 HHSI system.
4. Lower ranked sequences in the IRR room (e.g., Sequences 68 and 69) also include fire damage to both MDAFW pumps or damage to the B MDAFW pump.

5. A number of lower ranked sequences (e.g., 70th, 66th, 65th) are postulated to occur in the 1J SWGR sub compartment.
6. The availability/reliability of the cooling function provided by the TDAFW or MDAFW pump 3B is not significantly impacted by the unavailability of the 1J1 MCC because neither the pumps nor the flow path are affected. The pumps are not powered by the MCC. The MOVs on the flow path are normally open, kept open, and will be open and de-energized. Also, these valves can manually be operated locally, thus providing or isolating flow to the SGs (See [NB 12], Table 1, note 2).

Based on the above, the following conclusions are made:

1. Based on observations 1-3, and 6, the frequencies of the high ranked sequences from the IPEEE study will not significantly increase due to the unavailability of the 1J1 MCC because the 1J1 MCC would not have an impact on reliability of the AFW function or HHSI function as modeled for fires in the ESWGR fire area.
2. Based on observations 4, 5, and 6, the frequencies of the lower ranked sequences from the IPEEE study will also not significantly increase due to the unavailability of the 1J1 MCC because the 1J1 MCC would not have an impact on reliability of the AFW function or HHSI function as modeled for fires in the ESWGR fire area.

Therefore, it is judged that the increase in the ESWGR fire risk, due to the unavailability of the 1J1 MCC, is minimal because either the postulated fire damage in the area would disable the function provided by the MCC, or the unavailability of the MCC does not change reliability of the functions credited to mitigate the consequences of a fire in this fire area.

Total Delta CDF

To ensure that any potential impact is included, a delta CDF is conservatively estimated as follows:

The total base case fire CDF from the ESWGR room = 3.26E-6 per year (Page 266 of calculation file [NB 08])
Total fire frequency is 1.27E-2 per year (page 138 [NB 08])
An average CCDP can be approximated as 3.26E-6/1.27E-2= 2.57E-4 (it is noted that this is not really a CCDP but is used as a surrogate for an average of all the CCDPs)

As a conservative measure, due to the unavailability of 1J1 MCC, the average CCDP is increased by a factor of 2 (this is highly conservative because fires initiating in the 1J1 MCC are the highest contributors to the fire risk in this area). Therefore, delta CDF for this sub compartment, with 1J1 MCC unavailable is estimated as 3.26E-6 per year.

3.7.1.3-3 Determination of Delta CDF in the General Aux Building Fire Compartment

The Auxiliary Building is a four-level structure constructed of reinforced concrete with metal siding used for the upper levels. This fire area houses both normal operating and emergency components, in particular cable and equipment for the Component Cooling and Chemical Volume Control Systems.

In the IPEEE study, the potential fires and equipment damage in the Auxiliary Building were analyzed. The majority of the fire scenarios were determined to cause only limited damage, except for fires originating from the CCW pumps.

In the base fire PRA model, the Auxiliary Building contributes less than 1% to the Unit 1 total fire CDF. The CDF for this area is not higher because, the IPEEE model credited Operator action for manipulating MOVs and manual valves, after the postulated fire was extinguished. Human error probabilities (HEPs) for manual local operation of valves were included in the fire PRA model. Since the availability of the 1J1 MCC only impacts remote capability function of supported components and, as stated here, remote operation of valves was not credited in the fire PRA model, the unavailability of the 1J1 MCC is not expected to result in a change in the estimated risk for this area.

Also, based on a review of the top sequences for the Auxiliary Building (page 48-29, [NB 09]), it is noted that the reliability of the components credited for the mitigation of the postulated fire scenarios (for example, Unit 2 Charging system, Unit 2 CC, Unit 1 CC, AFW system) is not impacted by the 1J1 unavailability.

Therefore, based on the above evaluation and the low contribution of this area to the total Unit 1 fire CDF, it is concluded that the proposed extension of the 1J1 MCC CT will not impact the fire risk in this area.

3.7.1.3-4 Determination of Delta CDF in the Control Room

Fire scenarios which were developed in the IPEEE study for the Control Room mainly postulated to result in loss of control or indication provided by the control room cabinets. Typically, for most functions, the control and indications for the redundant trains are provided in close proximity of each other and are postulated to be impacted by the same fire scenario. Also, for more severe fires, the control room is expected to be evacuated and the plant to be shutdown from the Auxiliary shutdown panel, located in the Emergency Switchgear Room. An Auxiliary Monitoring Panel is also provided in the Fuel Handling Building. Emergency Diesel Generators are provided with local control panels and Component Cooling, Service Water, and RHR pumps can be operated from the Switchgear. Additionally, a number of local operator actions, such as manipulation of MOVs are required.

In the IPEEE fire PRA study, the Main Control Room fire contributes 4% to the Unit 1 total fire CDF

Based on a review of fire scenarios delineated on pages 11 through 21 of calculation file [NB 10] as well as a review of fire PRA results provided on pages 119 through 129 of the same calculation file, it is concluded that the impact of the 1J1 MCC unavailability on the conditional probability of core damage for the control room fire scenarios is negligible on the basis that:

1. The control and indication for the redundant components are mostly affected by the same fire.
2. The fire PRA model credits manual local operation of a number of valves, including those that are powered by the 1J1 MCC (e.g., See a list of operator actions that are required in Auxiliary Building per Appendix A of [NB 10]). This list was generated based on plant procedure 0-FCA-1.
3. The cooling provided by the AFW system is important in most fire scenarios. Due to the configuration of the AFW valves which are supported by the 1J1 MCC (as discussed previously), the reliability of this function is not significantly affected by 1J1 MCC unavailability.

Therefore, no delta CDF is calculated for the fires originating in the Control Room.

3.7.1.3-5 Determination of Delta CDF in the General Area of Turbine Building (TB)

The IPEEE study screened out this area based on the FIVE methodology screening rules. In this analysis, based on the analysis described in section 3.7.1.1, this fire compartment is retained for further evaluation. The delta CDF for this fire area is conservatively calculated as follows:

Initiating Event = T1 (Loss of Offsite Power)
Fire Frequency = 2.6E-2 per year (similar to the 1J1 case)
CCDP for a LOOP given guaranteed failure of MDAFW 3A = 1.22E-4 (From run MCC1J1(A)-T1-AFW-3A-FS.EQP) * 5 = 6.1E-4
Therefore, CDF (base) = 2.6E-2 * 6.1E-4 = 1.6E-5 per year
Delta CDF = 1.8E-5 (from the screening table) – 1.6E-5 = 2.0E-6 per year

This is conservative due to the use of a high fire frequency, postulation of loss of offsite power for the given frequency, and due to the postulated damage for the assigned fire frequency.

3.7.2 Total increase in Fire CDF Due to the Unavailability of 1J1 MCC

Based on the results of the assessment described in sections 3.7.1.1, 3.7.1.2, and 3.7.1.3, the total increase in CDF is:

Total increase = Delta CDF from the Turbine Building Fires + Delta CDF from the CV&T Fire + Delta CDF from the ESWGR Fires = 2.0E-6 + 9.9E-7 +3.26E-6 = 6.25E-6 per year. As shown in Table 4.1-1, the base case fire CDF is 3.91E-6 per year. This evaluation presented here conservatively estimates an increase of 6.25E-6 per year, resulting in a total fire CDF of 1.0E-5 per year.

Note that, for most fire areas, the qualitative assessment of the unscreened fire areas (see sections 3.7.1.1, 3.7.1.2, etc of this document), indicates that the impact of the 1J1 MCC being out of service on the fire-induced CDF for a particular compartment is negligible. However, conservatively, the risk calculation, intentionally, uses a very conservative approach to provide a quantitative estimate of the CDF increase. As a result, the estimated CDF increase is about factor of two greater than the base case fire CDF (That is, the fire CDF with 1J1 being out of service is more than two times higher than the base case fire CDF). Given the components that would be impacted by 1J1 MCC being out of service, this is highly conservative. This conservative approach was used to ensure that any optimism that may have crept in due to the use of the internal events model or the qualitative nature of this assessment is well compensated for by the conservatism included in other sections. Since this is not a vulnerability study (i.e., the relative risk is not important), then the use of conservatisms in some area to compensate for potential non-conservatism in other parts is appropriate. Some of the conservatisms in the modeling include:

- In most cases (from the quantification point of view), other than use of the no-maintenance model in the screening stage, no credit is taken for the fact that this is a one-time extension when the work will be performed with plant configuration known and certain compensatory measures in place.
- From the fire hazard point of view and postulated initiators, the availability/reliability of the 2nd heat removal function is an important factor. The availability/reliability of the cooling function provided by components-supported by the 1J1-MCC (i.e., the TDAFW or MDAFW pump 3B) are not significantly impacted by the unavailability of the 1J1 MCC because the MOVs on the flow path are normally open, kept open, and will be open and de-energized. Also, these valves can manually be operated locally, thus providing or isolating flow to the SGs.
- Similar to the internal flooding hazard, due to the plant physical configuration and location of Safe Shutdown equipment and their associated cables, the significant contributors to the internal fire risk include those scenarios that have the potential to disable redundant components performing the same function. Therefore, the impact of unavailability of one train is not as consequential.

3.7.3 Impact on Large Early Release Frequency (LERF) Figure of Merit

The IPEEE study did not include a qualitative estimate of the LERF figure of merit. Instead, it performs a qualitative evaluation of the containment to address three areas of concern: bypass, containment isolation and heat removal. From this LAR point of view, the IPEEE analysis was reviewed to determine if anyone of the three IPEEE-evaluated concerns would be impacted by the 1J1 MCC unavailability.

Note that induced SGTR was included in the conditional probabilities of the end states, as described in [NB 13].

Impact on Containment Bypass-

The two significant bypass mechanisms identified were the interfacing system LOCA (V-sequence) and the steam generator tube rupture (SGTR). Mechanical failure was the cause of both of these events. The IPEEE study concluded that the components involved are piping and check valves, a fire would not be expected to initiate one of the events. This conclusion is also applicable to the plant configuration when 1J1 MCC is out of service and, therefore, there is no impact on any release frequency, including the LERF.

Containment Isolation Reliability

Loss of containment isolation can occur in a number of different ways and in theory could happen before, during or after an initiating event. The IPEEE study concluded that since North Anna is operated at subatmospheric pressure, there is no possibility of a large hole or other leakage path prior to operation. However, a leak could reasonably be postulated to occur as a result of an accident. The likelihood of such a leak would be a function of the peak containment pressure and to a lesser extent the time at pressure, as well as the reliability of the containment isolation function (such as reliability of valve closure).
Based on a review of calculation file for containment isolation modeling ([NB 14], Table 4-2 (North Anna Containment Penetration List and Screening) and Figure 4-1 (North Anna Unit 1 Containment Isolation fault Tree)), MOVs are not credited for containment isolation and as a result their reliability, is not included for estimating the reliability of containment isolation function. Therefore, the containment isolation reliability is not affected by the 1J1 MCC being out of service (note that the potential impact of the 1J1 MCC being OOS, on the screening criteria used in [NB 14], was assessed. Based on this evaluation it was concluded that the screening results would not change)

Containment Heat Removal

In the IPEEE study, detailed analysis of fires resulted in the selection of appropriate internal events trees which could be used to determine the contribution to core damage frequency from fires in an area or compartment. The trees selected for the various areas included: Transients with Main Feedwater Available (T3), Loss of All Seal Cooling (T4), Loss of Switchgear Room Cooling (T8), and Loss of an Electrical Bus (T9A,T9B). Based on a review of [REPORT 14] which includes the Level 2 trees for these Level 1 event trees, it is judged that a change in the reliability of Containment Heat Removal (CHR) function could impact the LERF estimate.

Based on a review of [NB 15] and [DRW 01], a number of Quench Spray and Recirculation Spray systems' components (mostly MOVs (or MOVs on the supporting systems such as Service Water)), are powered by the 1J1 MCC. Some of these valves (e.g., 1-QS-MOV-101B) are normally closed valves that have to open to support the CHR function.

Fire-Induced LERF Calculations

Based on the above evaluations, the change in the LERF figure of merit is calculated as follows:

$\Delta LERF = \Delta_{LERF}^{CDF} + \Delta_{LERF}^{CHR}$

Where Δ_{LERF}^{CDF} is the increase in LERF due to increase in CDF and
Δ_{LERF}^{CHR} is the increase in LERF due to decrease in the reliability of CHR function

The Δ_{LERF}^{CDF} is calculated as follows:

Δ_{LERF}^{CDF} = increase in CDF * random probability of containment isolation failure probability

From section 3.7.2, the increase in fire-induced CDF = 6.25E-6 per year.
From [NB 16], all potential containment pathways, which their failure may result in a large early release event, have been screened out. However, as a conservative measure it is decided to assign a probability estimate to the containment failure likelihood. Per [NB 14], Attachment B, Event Trees (e.g., the Event Tree for the T8A (Loss of Switchgear Room Cooling due to loss of Unit 1 AHUs)), a failure probability of 1.48E-3, was estimated for Surry's containment isolation. This estimate is used in this analysis for random failure probability of containment isolation for NAPS.

The Δ_{LERF}^{CHR} is calculated as follows:

$\Delta_{LERF}^{CHR} = F_{PDS}^{LERF} * P_{LERF}^{PDS}$
Where
F_{PDS}^{LERF} is the increase in the frequency of fire-induced Plant Damage State (PDS) that are mapped to the LERF figure of merit
And
P_{LERF}^{PDS} is the conditional probability of LERF for a given PDS

To calculate F_{PDS}^{LERF} first the pertinent PDSs must be identified. A review of the quantified event trees, that represents the fire-induced core damage events, (REPORT 14), shows that core damage sequences that include failure of systems that support containment heat removal function (i.e. Quench Spray (QS) and Recirc Spray (RS)), are mapped to PDSs 23, 22, 11, and 10 (excluding the PDSs that follow containment structural failure. These PDSs have a higher conditional probability of going to LERF but the probability of containment structural failure (about 2.0E-2) is bounded by the value (6.425E-2) which is used here). By conservatively assuming a failure probability of 1.0 for all the functions which are credited for the containment heat removal function (i.e., assuming that for the configuration of concern, the QS, RS, functions would fail with probability of 1.0), the F_{PDS}^{LERF} is bounded by the increase in the core damage frequency.

To calculate P_{LERF}^{PDS} the bed file that is used for the quantification of the current NAPS PRA model is reviewed. Based on this review, the following conditional LERF fractions are obtained:

PDS	Conditional LERF Prob
PDS 23	6.425E-2
PDS 22	1.555E-3
PDS 11	4.059E-2
PDS 10	2.417E-2

Since for this application, the change in CDF is estimated based on a qualitative evaluation of the IPEEE core damage results, the frequency for each fire damage state is not available. Therefore, conservatively, the following assumptions are made:

1. Unavailability of the 1J1 MCC would result in failure of the CHR function for all fire events that are modeled to contribute to the increase in CDF.
2. All core damage sequences that contribute to the increase in CDF (due to unavailability of 1J1 MCC) are assumed to go to the PDS with the highest conditional LERF probability (i.e., they all mapped to PDS 23).

Therefore,

$\Delta LERF = \Delta_{LERF}^{CDF} + \Delta_{LERF}^{CHR} = 6.25E\text{-}6 * 1.48E\text{-}3 + 6.25E\text{-}6*6.42E\text{-}2 = 4.11E\text{-}7$ per year

Since the IPEEE study did not calculate contribution of the fire hazard to LERF, in this analysis, only the delta LERF is estimated.

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
1-1	RC-1	Unit 1 Containment	Yes	Yes	This area was screened out per FIVE-	This area was screened out in the IPEEE study because, per FIVE, Containment fires can be screened out without conducting an analysis if, (a) plant experience indicates that fires in containment have not occurred on a re-occurring basis during plant operation, (b) redundant trains of critical equipment cannot be exposed to the same fire plume or are located within a confined space which might be susceptible to hot gases. This logic provided for the IPEEE study applies to this assessment. Therefore, the area remains screened out.
2	MCR	Main Control Room	Yes	Yes	Detailed scenario was developed	Detailed evaluation recalculated
3-1	CV&T-1	Unit 1 Cable Vault and Tunnel	Yes	Yes	Detailed scenario was developed	Detailed evaluation recalculated
3-1	Z-27-1, MGSH-1	Unit 1 Motor Generator Set House	No	*	Screened	Quantified for this analysis as a turbine trip/loss of MFW (T23) with 1J1 MCC unavailable. From discussion of qualitative screening CCDP = 8.6E-7 Fire frequency = MG Set is the major fixed fire

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
	.					ignition source in this compartment. From [REPORT 15] fire frequency = 3.4E-3 per year Fire CDF = 3.4E-3 * 8.6E-7 = 2.9E-9 per year. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
4-1	CSR-1	Unit 1 Cable Tray Spreading Room	No	*	Screened.	Quantified for this analysis as a turbine trip/loss of MFW (T23) with 1J1 MCC unavailable. From discussion of qualitative screening CCDP = 8.6E-7 Fire frequency = Other than cables, no other major fixed ignition source located in this area. In the IPEEE study, a Fire Ignition Source Data Sheet was not generated for this area. Therefore, as an approximate, fire frequency for this compartment is estimated based on the cable-induced fire frequency for the Cable Vault and Tunnel fire area. From Table 3.3-1 of [NB 06], fire frequency =

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
						3.52E-4 per year. Clearly the CDF is significantly less than 1.0E-6 per year. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
5-1	NSR-1	Unit 1 Normal Switchgear Room	No	*	Screened	Quantified for this analysis as a LOOP with 1J1 MCC unavailable. From the above discussion of qualitative screening, the CCDP = 6.0E-5 Fire frequency = Electrical cabinets are the major fixed fire ignition source in this compartment. From NAPS calculation file [NB 06] (NAPS quantitative screening calculation file) frequency = 7.5E-3 per year (based on fire frequency for the emergency Switchgear Room) Fire CDF = 7.5E-3 * 6.0E-5 = 4.5E-7 per year. Although the value is very close to the screening

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
						value of 1.0E-6, the area is assessed to have a negligible impact on the duration of the proposed increase in 1J1 CT on the basis that 1) the base case CDF contribution is not subtracted from the value and 2) no credit for severity factor is given.
6-1	ESGR-1	Unit 1 Emergency Switchgear Room	Yes	Yes	Detailed evaluation	Detailed evaluation recalculated
7A-1	BR1-I	Unit 1 Battery Room 1-I	No	*	Screened.	Quantified for this analysis as a turbine trip with 1J1 MCC and Unit 1 Battery unavailable. CCDP = 4.29E-7 (from run MCC1J1-TNORM-BAT1-I.EQP) * 2 = 8.6E-7. The fire frequency is significantly less than 1.0 E-1 per year, therefore, the CDF is significantly lower than 1.0E-6 per year. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
7B-1	BR1-II	Unit 1 Battery Room 1-II	No	*	Screened	Quantified for this analysis as a turbine trip with 1J1 MCC and Unit 1 Battery unavailable. CCDP = 4.29E-7 (from run MCC1J1-TNORM-

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
						BAT1-II.EQP) * 2 = 8.6E-7. The fire frequency is significantly less than 1.0 E-1 per year, therefore, the CDF is significantly lower than 1.0E-6 per year. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
7C-1	BR1-III	Unit 1 Battery Room 1-III	No	*	Screened for IPEEE	This area is assessed to have a negligible impact on the duration of the proposed increase in 1J1 CT on the basis that components in the area are in the same train as the 1J1 MCC.
7D-1	BR1-IV	Unit 1 Battery Room 1-IV	No	*	Screened for IPEEE but quantified for this analysis as a reactor trip with 1J1 MCC unavailable.	This area is assessed to have a negligible impact on the duration of the proposed increase in 1J1 CT on the basis that components in the area are in the same train as the 1J1 MCC.
8	TB	Turbine Building	Yes	Yes	For IPEEE this compartment screened out *Explanation*: A fire may result in a loss of station service and a partial loss of transfer bus potentially causing a loss of one motor driven AFW pump.	The contribution from this area is calculated as follows: Initiating Event = T1 (Loss of Offsite power) Initiating event frequency =This area contains a

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
					The normal alternate means of shutdown would be to use the remaining motor driven or turbine driven auxiliary feedwater pump. In the event of either of the undamaged AFW pumps being unavailable (due to unrelated random failures) an independent shutdown path would remain intact. Turbine building fires can therefore be screened from further consideration.	number of main ignition sources (e.g., MFW pump, electrical cabinets). Based on a review of [NB 06], "Electrical cabinets" class of fixed ignition sources has the highest fire frequency. Therefore, the fire frequency is 2.6E-2 per year. CCDP is calculated by running the N105A-TM0 model and setting failure probabilities of 1J1 MCC and the MDAFW 3A to 1.0 (MDAFW pump 3A was selected since it is supported by the H train). Based on a run, CCDP for this configuration for the internal events is 1.37E-4 (See Run MCC1J1-T1-AFW-3A-FS-EQP). Fire-induced version of this scenario is a relatively complicated event; therefore, the CCDP is increased by a factor of 5. That is, the CCDP for the fire event is 6.9E-4 and CDF= 2.6E-2 *6.9E-4 = 1.8E-5 per year. It is conservatively concluded fires initiated in this compartment can impact duration of the proposed extension to the 1J1 MCC CT.

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
8	8B, TB-LOR	Turbine Building, Lube Oil Room	No	*		A fire in this compartment will not result in a plant trip or loss of any SSD components. Therefore, the contribution of this compartment to the fire-induced risk, as a result of the proposed CT extension, would not change since the safe shutdown components supported by 1J1 MCC would not be challenged.
8	ESGR/CR-1	Unit 1 MCR/ESGR Chiller Room	Yes	No		In the IPEEE study, based on a review of TS 3.7.11. Action D, it was concluded that a fire in this compartment would not result in a plant trip or a forced shutdown within 8 hours. For this analysis, based on a review of TS 3.7.11. Action E. it is judged that a fire in this area would result in a plant shutdown. Also, based on a review of IPEEE data, it is concluded that the safe shutdown credited equipment which are potentially affected by a fire in this compartment are HVAC related components. Such potential damage would impact both trains of the safe shutdown path. As a

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compartment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
						result, the contribution of the proposed Tech Spec change on the fire risk in this area is negligible.
8	Z-34, SB-E	Service Building East	No	*		No initiating event or damage to safe shutdown components is postulated. For this application, it is concluded that the contribution of these areas to the fire-induced risk would not change since the safe shutdown components supported by 1J1 MCC would not be challenged.
8	Z-35, SB-W	Service Building West	No	*		No initiating event or damage to safe shutdown components is postulated. For this application, it is concluded that the contribution of these areas to the fire-induced risk would not change since the safe shutdown components supported by 1J1 MCC would not be challenged.
8	50, SB-STAIR	Service Building Stairwell	No	*		No initiating event or damage to safe shutdown components is postulated. For this application, it is concluded that the contribution of these areas to the fire-induced risk would not change since the safe shutdown components supported by 1J1 MCC

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
						would not be challenged.
8	46A, TSC	Technical Support Center	No	*		No initiating event or damage to safe shutdown components is postulated. For this application, it is concluded that the contribution of these areas to the fire-induced risk would not change since the safe shutdown components supported by 1J1 MCC would not be challenged.
8	Z-46B, TSCBR	Technical Support Center Battery Room	No	*		No initiating event or damage to safe shutdown components is postulated. For this application, it is concluded that the contribution of these areas to the fire-induced risk would not change since the safe shutdown components supported by 1J1 MCC would not be challenged.
8	Z-22, AHBR	Auxiliary Heating Boiler Room	No	*		No initiating event or damage to safe shutdown components is postulated. For this application, it is concluded that the contribution of these areas to the fire-induced risk would not change since the safe shutdown components supported by 1J1 MCC would not be challenged.

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
8	Z-36, OB	Office Building	No	*		No initiating event or damage to safe shutdown components is postulated. For this application, it is concluded that the contribution of these areas to the fire-induced risk would not change since the safe shutdown components supported by 1J1 MCC would not be challenged.
8	Z-21B, RR-OB	Records Room Office Building	No	*		No initiating event or damage to safe shutdown components is postulated. For this application, it is concluded that the contribution of these areas to the fire-induced risk would not change since the safe shutdown components supported by 1J1 MCC would not be challenged.
8	Z-21C, RR-NOB	Records Room New Office Building	No	*		No initiating event or damage to safe shutdown components is postulated. For this application, it is concluded that the contribution of these areas to the fire-induced risk would not change since the safe shutdown components supported by 1J1 MCC would not be challenged.
9A-1	EDG-1H	Unit 1 Emergency Diesel Generator 1H	Yes	Yes	Area screened out	Quantified for this analysis as a turbine trip/loss of

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
					Explanation: A fire may cause the loss of the 1H EDG but will not cause automatic plant trip. However such a fire in combination with an unrelated failure of another EDG unit would require a controlled plant shutdown within 6 hours. Since there is no fire impact on offsite power supplies to the normal or emergency buses) a shut down path is available which is independent of the equipment assumed to be damaged or unavailable. The area can therefore be screened out.	MFW (T23) with 1J1 MCC unavailable and 1H EDG unavailable. CCDP = 4.29E-7 (See run MCC1J1-TNORM-EDG1H-FS) (note that there is not much change in the CCDP given 1H EDG is out of service) Fire frequency = Diesel is the major fixed fire ignition source in this compartment. From EPRI-TR-1000894 (Fire Events Database for U.S. Nuclear Power Plants, Oct. 2000) fire frequency = 3.9E-2 per year Fire CDF = 3.9E-2 *4.29E-7*2 = 3.4E-8 per year. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
9B-1	EDG-1J	Unit 1 Emergency Diesel Generator 1J	Yes	Yes	Area screened out *Explanation*: See 1H EDG Room	This area is assessed to have a negligible impact on the duration of the proposed increase in 1J1 CT on the basis that components in the area are in the same train as the 1J1 MCC.

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
10A	FOPR-A	Fuel Oil Pump House Room A	Yes	Yes	Area screened out *Explanation:* A fire in the EDG fuel oil pump room may result in the loss of the A fuel oil pump for Unit 1 and Unit 2 but will not cause automatic plant trip. However such a fire in combination with an unrelated failure of a B fuel oil pump would require a controlled shutdown within 6 hours. Since there is no impact on offsite power a shut down path is available which is independent of the equipment assumed to be damaged or unavailable. The fuel oil pump room can therefore be screened out.	There is a total of 4 pumps. 1HA and 1HB support EDG H and 1JA and 1JB support EDG J. In an event of a fire in this compartment, the A train for both H and J EDGs is impacted. However, the B train remains operable. Given that a fire in this area would not result in a loss of offsite power nor cause an initiating event, the contribution of fires in this compartment to the fire risk, independent of the availability of 1J1 MCC, is considered to be below 1.0E-6 and can be screened out from further evaluation.
10B	FOPR-B	Fuel Oil Pump House Room B	Yes	Yes	Area screened out *Explanation:* See A Fuel Oil Pump Room	There is a total of 4 pumps. 1HA and 1HB support EDG H and 1JA and 1JB support EDG J. In an event of a fire in this compartment, the B train for both H and J EDGs is impacted. However, the A train remains operable. Given that a fire in this area would not result in a loss of offsite power nor cause an initiating event, the contribution of fires

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
						in this compartment to the fire risk, independent of the availability of 1J1 MCC, is considered to be below 1.0E-6 and can be screened out from further evaluation.
10C	MCC	Fuel Oil Pump House Motor Control Center Room	No	*		Quantified for this analysis as a turbine trip/loss of MFW (T23) with 1J1 MCC unavailable. From discussion of qualitative screening CCDP = 8.6E-7 Fire frequency = The major ignition sources in this compartment are MCCs. Based on a review of [REPORT 15] fire frequency for this area can be shown to be less than 0.1 per year. Therefore, the CDF contribution is well below the 1.0E-6 threshold. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
11	AB	Auxiliary Building	Yes	Yes	Area NOT screened out	See detailed evaluation.

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
11	11A, CPC-1A	Charging Pump Cubicle 1A	Yes	No		This is a rather small fire area. A fire in this compartment will not result in a plant trip. Even if a reactor trip were assumed, loss of 1A charging pump in combination with the unavailability of 1J1 MCC would not have a significant impact on the pertinent redundant safe shutdown components and systems (e.g., Charging pumps B and C will be available and the MOVs in the flow path that are powered by the 1J1 MCC will be in their required operating position).
11	11B, CPC-1B	Charging Pump Cubicle 1B	Yes	No		This is a rather small fire area. A fire in this compartment will not result in a plant trip. Even if a reactor trip were assumed, loss of 1B charging pump in combination with the unavailability of 1J1 MCC would not have a significant impact on the pertinent redundant safe shutdown components and systems (e.g., Charging pumps A and C will be available and redundant MOVs in the flow path which are powered by the H bus will be available).

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
						Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
11	11C, CPC-1C	Charging Pump Cubicle 1C	Yes	No		This is a rather small fire area. A fire in this compartment will not result in a plant trip. Even if a reactor trip were assumed, loss of 1B charging pump in combination with the unavailability of 1J1 MCC would not have a significant impact on the pertinent redundant safe shutdown components and systems (e.g., Charging pumps A and B will be available and redundant MOVs in the flow path which are powered by the H bus, will be available). Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
11	Z-20, DB	Decontamination Building	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components.

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
						Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
11	Z-18, FB	Fuel Building	Yes	No		Based on a review of [NB 02], Table 4-1.11, the IPEEE study assumes that, from the safe shutdown point of view, only the Unit 2 Aux shutdown panel may be affected by a fire in this compartment. Based on a review of this analysis by a former SRO, it was not clear whether the IPEEE assumption was correct. However, a fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
11	51, AB-STAIR	Auxiliary Building Stairwell	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
						compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
12	SWPH	Service Water Pump House	Yes	No		A fire in this compartment may result in the loss of all 4 normal service water pumps (2 per Unit), leaving the two aux service water pumps available. Based on a review of 480 Volt one-line diagram for 1J1 MCC ([DRW 01]), none of the Aux SW pumps related components are powered by the 1J1 MCC. Also, the IPEEE study concluded that a fire in this area would not result in a plant shutdown. Even if the plant were to be forced to shutdown, the availability of the other safe shutdown components which are required for a forced shutdown (i.e., a non-trip shutdown) is not significantly affected by the unavailability of 1J1 MCC. Therefore, **the change** in risk would be minimal and the fire contribution from this compartment is considered to have a negligible

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
						impact on the duration of the proposed increase in 1J1 CT.
13	ASWPH	Auxiliary Service Water Pump House	Yes	No		A fire in this area would not result in a plant trip and would not impact the availability of the 4 normal SW pumps. Since during the proposed unavailability of the 1J1 MCC, all other components important to safety (e.g., all normal SW pumps) are available to perform their function, the impact on accident mitigation capability would be negligible. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
14A-1	TDAFW-1	Unit 1 Turbine Driven Auxiliary Feedwater Pump Room	Yes	Yes	Area screened out *Explanation*: A loss of the turbine driven auxiliary feedwater in combination with a random failure of one of the motor driven pumps will result in the requirement for a controlled shutdown within 6 hrs. However since one motor driven and the main	A fire in this compartment is assumed to result in a T23 initiating event and a loss of TDAFW pump, in combination with unavailability of the 1J1 MCC. It is noted that this is conservative, since the postulated initiating event assumes main feedwater to be unavailable whereas the IPEEE

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
					feedwater/condensate is unaffected by the fire, at least one shutdown path is available which is independent of the equipment assumed to be damaged by the fire or unavailable. The area can therefore be screened out.	had concluded that MFW system would be available. Initiating event frequency =The major ignition source in this area is the Turbine Driven AFW pump. The IPEEE study qualitatively screened out this compartment and did not develop fire ignition source datasheet for this compartment. It is judged that using the Main Feed Water fire frequency bounds the TDAFW fire frequency. Based on a review of [NB 05], page 38, the generic fire frequency for the MFW pump in the Turbine Building is 4.0E-3 per year. Since there is only one TDAFW pump in this room but there are typically 3 MFW pumps in the Turbine Building, the fire frequency for the TDAFW pump is assigned as 4.0E-3/3= 1.3E-3 per year. CCDP is calculated by running the N105A-TM0 model and setting failure probabilities of 1J1 MCC

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
						and the TDAFW pump to 1.0. Based on a run (Run MCC1J1-TNORM-TDAFW-FS.EQP), CCDP for this configuration for the internal events is 3.03E-6. As stated in the general discussion of this analysis, the CCDP is adjusted to reflect the potential impact on operator actions. The postulated scenario in this compartment is not considered to be a significantly challenging event. Therefore, the CCDP is increased by a factor of 2 to account for a potential higher stress on the operators. Therefore, the CCDP for the fire event is estimated as 6.1E-6. It is clear that the CDF (i.e., 1.3E-3 * 6.1E-6) is significantly less than 1.0E-6/yr screening threshold. Therefore, it is concluded fires initiated in this compartment do not have an impact on duration of the proposed extension to the 1J1 MCC CT.
14B-1	MDAFW-1	Unit 1 Motor Driven Auxiliary Feedwater Pump Room	Yes	Yes	Area screened out: *Explanation:* Loss of both motor driven Auxiliary	A fire in this compartment is assumed to result in a T23 initiating event and a loss of both MDAFW

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
					Feedwater pumps may result from a fire which will require a controlled plant shutdown within 6 hours. However even with an assumed coincident random failure of the turbine driven pump, the normal method of plant shutdown (main feedwater/ condensate) remains available. The area can therefore be screened out.	pumps, in combination with unavailability of the 1J1 MCC. It is noted that this is conservative, since the postulated initiating event assumes main feedwater to be unavailable whereas the IPEEE study had concluded that MFW system would be available. Initiating event frequency =The major ignition source in this area is the two Motor Driven AFW (MDAFW) pumps. The NAPS IPEEE study qualitatively screened out this compartment and did not develop fire ignition source datasheet for this compartment. In the Surry Power Station's (SPS) IPEEE study, a fire frequency of 1.6E-3/yr is estimated for fire area 19-1 (Auxiliary Feedwater Pump house, Page 4-30 of [REPORT 16]). Also, based on the above evaluation for fire area 14A-1, the fire frequency for the TDAFW pump is estimated as1.3E-3 per year. Based on

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
						these evaluations, the fire frequency for this compartment is estimated as 2.0E-3 per year. CCDP is calculated by running the N105A-TM0 model and setting failure probabilities of 1J1 MCC and the 2 MDAFW pumps to 1.0. Based on a run, CCDP for this configuration for the internal events is 2.46E-4 (Run MCC1J1-TNORM-AFWPA&B-FS.EQP). As stated in the general discussion of this analysis, the CCDP is adjusted to reflect the potential impact on operator actions. The postulated scenario in this compartment is not considered to be a significantly challenging event. Therefore, the CCDP is increased by a factor of 2 to account for a potential higher stress on the operators. Therefore, the CCDP for the fire event is 4.92E-4 and CDF= 4.92E-4 * 2.0E-3 = 9.844E-7 per year. This estimate is close to the screening threshold of

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
						1.0E-6 per year. Therefore, for this case, the base case CDF is also important. Based on a run for this area with both MDAFW failure probabilities set to 1.0, the base case CCDP = 2.456E-4 (Run MCC1J1(A)-TNORM-AFWPA&B-FS.EQP). Using the same approach for the above case (i.e., multiplying the run CCDP by a factor of 2) and calculating the CDF, the resulting CDF is = 2.456E-4*2*2.0E-3 = 9.824E-7 per year. That is the increase in CDF is about 2E-9 and, therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT. This is to be expected, since the reliability of the TDAFW pump is not significantly impacted by the 1J1 MCC unavailability (pertinent valves are either in the accident mitigation configuration or can be manually manipulated.)
15-1	QSPH-1	Unit 1 Quench Spray Pump House and	Yes	Yes	Compartment screened out	Power cables for both MDAFW pumps are routed

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
		Safeguards Area			*Explanation*: A fire in this area may disable Quench Spray trains requiring a controlled plant shutdown. However even with an assumed coincident random failure of the turbine driven pump, the normal method of plant shutdown (main feedwater/condensate) remains available. The area can therefore be screened out.	in this area (Table 4-1.15-1 of[NB 05]). Therefore, similar to fire area 14-B (MDAFW pump room), a fire in this compartment is assumed to result in a T23 initiating event and a loss of both MDAFW pumps, in combination with unavailability of the 1J1 MCC. It is noted that this is conservative, since the postulated initiating event assumes main feedwater to be unavailable whereas the IPEEE study had concluded that MFW system would be available. Initiating event frequency =The major ignition source in this area is the two quench spray pumps. The NAPS IPEEE study qualitatively screened out this compartment and did not develop fire ignition source datasheet for this compartment. In the Surry Power Station's (SPS) IPEEE study, a fire frequency of 1.1E-3/yr is estimated for fire area 19-2 (Containment Spray Pump Room) (Page 4-31

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
						of [REPORT 16]). Therefore, the fire frequency for this compartment is estimated as 1.1E-3 per year. CCDP is calculated by running the N105A-TM0 model and setting failure probabilities of 1J1 MCC and the 2 MDAFW pumps to 1.0 (Run MCC1J1-TNORM-AFWPA&B-FS.EQP).. The impact of the unavailability of the quench spray pumps on the reliability of functions credited for mitigation of the postulated initiating event in this compartment is negligible. Since the potential fire-induced damage for fire area 14-B and this compartment is the same and the fire frequency for this compartment is less, the conclusions of the fire compartment 14-B is applicable to this model. That is, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT. This is to be expected, since the reliability

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
						of the TDAFW pump is not significantly impacted by the 1J1 MCC unavailability (pertinent valves are either in the accident mitigation configuration or can be manually manipulated.)
15-1	Z-16-1, SA-1	Unit 1 Safeguards Area	Yes	Yes	Compartment screened out *Explanation*: A fire in this area may disable all ORS and LHSI pumps requiring a controlled plant shutdown. However even with an assumed coincident random failure of the turbine driven pump, the normal method of plant shutdown (main feedwater/condensate) remains available. The area can therefore be screened out.	It is noted that the consequences of a fire in this area is very similar to theconsequences of a fire in the QSPH-1 compartment. Indeed comparing the results of a run for this area where LHSI and ORS are failed in addition to those components failed for the QSPH-1 compartment (ie., comparing the results for runs Run MCC1J1-NORM-MDAFW-LHSI-CRS.EQP and Run MCC1J1-TNORM-AFWPA&B-FS.EQP), it is notes that the CCDP only increases by (2.92E-4- 2.46E-4) 4.6E-5. Given the fire frequency for this area is about the same as the QSPH-1, the conclusions of the QSPH-1 are applicable to this area.
17-1	MSVH-1	Unit 1 Main Steam Valve House	Yes	No		A fire in this compartment may result in a loss of SG level indication/pressure transmitters.

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
						Although, the IPEEE study had concluded that a fire in this area will not result in a plant trip, for this analysis, conservatively, it can be postulated that a fire in this compartment may result in a plant trip. However, there are no major ignitions sources in this area. Therefore, due to low initiating event frequency and low CCDP, it is concluded that the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
19	WDB	Waste Disposal Building	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
19	Z-19B, BRB	Boron Recovery Building	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components.

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
						Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
26	FPB	Motor Driven Fire Pump Building	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
32	ISCH	Intake Structure Control House	Yes	No		Based on a re view of [NB 05], a fire in this area would not result in a plant trip and would not impact the availability of the 4 normal SW pumps. Since during the proposed unavailability of the 1J1 MCC, all other components important to safety (e.g., all normal SW pumps) are available to perform their function, the impact on accident mitigation capability would be negligible. Therefore, the fire contribution from this

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
						compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
42	YVP	Yard Valve Pit	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
48	SWVH	Service Water Valve House	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
49	DAW	Dry Active Waste Building	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
						impact on the duration of the proposed increase in 1J1 CT.
YARD	Z-23, SB	Security Building	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
YARD	Z-24, SCC	Security Control Center	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
YARD	Z-8C, XFMRS	Main and Station Service Transformers	No	*		This is an outdoor fire area. A fire in this area would not damage any safe shutdown credited components but may result in a LOOP event. This area can be modeled similar to fire area 5-1 (Normal Switchgear Room). Therefore, the CCDP

Table 3.7.1-1 Qualitative Screening Summary North Anna Power Station						
IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
						for this area can be estimated as 6.0E-5. Transformers are the major ignition sources in this area. Per EPRI fire events database, Table 5-3,the fire frequency for this area can be estimated as 5.1E0-3/yr and the change in CDF can be roughly estimated as 5.1E-3 * 6.0E-5 = 3.06E-7 per year. Therefore, the area is assessed to have a negligible impact on the duration of the proposed increase in 1J1 CT and no further evaluation is performed.
YARD	CTFS	Condensate Truck Fill Station	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
YARD	Z-37, FOST	Fuel Oil Tanks	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment: IPEEE	Comment: 1J1 MCC LAR
						impact on the duration of the proposed increase in 1J1 CT.
YARD	Z-39, APSB	Security Auxiliary Power Supply Building	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
YARD	Z-41-1, CCT&PH-1	Unit 1 Casing Cooling Tank & Pump House	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
YARD	Z-43, RSB	Records Storage Building	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
						1J1 CT.
YARD	Z-44, MSF	Maintenance Shop Facility	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
YARD	Z-45, PBXB	PBX Building	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
YARD	DINERS	Rail Dining Cars	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
OUTSIDE	Z-25, WTB	Water Treatment Building	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
OUTSIDE	Z-29, WH-2	Warehouse No. 2	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
OUTSIDE	Z-30, VPH	Vacuum Priming House	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
OUTSIDE	Z-31,	Vacuum Priming Tank Enclosure	No	*		A fire in this area is not expected to cause a plant

Table 3.7.1-1

Qualitative Screening Summary
North Anna Power Station

IPEEE Fire Area Number	IPEEE Fire Compart-ment	Description	Is Unit 1 Appendix R Equipment in the Area	Will a Fire Initiate a Unit 1 Shutdown?	Comment	
					IPEEE	1J1 MCC LAR
	VPTE					shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
OUTSIDE	Z-33, SGCH	Spillway Gate Control House	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.
OUTSIDE	Z-40, BCWCT	Bearing Cooling Water, Cooling Towers	No	*		A fire in this area is not expected to cause a plant shutdown nor impact safe shutdown components. Therefore, the fire contribution from this compartment is considered to have a negligible impact on the duration of the proposed increase in 1J1 CT.

3.8 Seismic Analysis

The North Anna IPEEE did not perform seismic risk calculations. The North Anna IPEEE used a seismic margins evaluation based upon a *Review Level Earthquake* (RLE) with a peak ground acceleration of 0.3g [REPORT 03, p. 2-1]. The frequency of this event may be estimated from the *EPRI Mean Seismic Hazard Curve* for Surry.

The seismic CDF associated with a 1J1 bus tagout may be evaluated as follows. A severe seismic event is likely to produce a long-term LOOP that is not readily recovered, due to potential severe damage in the switchyard. Therefore it is similar to a LOOP event due to other causes, with little or no opportunity for recovery. Based on a review of the internal events LOOP cutsets' and results' for the base case configuration (MCCJ1(A)-T1.EQP) and the 1J1 configuration (MCC1J1-T1.EQP), it is noted that:

- The unavailability of the 1J1 MCC increases the LOOP's CCDP by 7.5E-6. This is a very small increase in the CCDP.
- The new cutsets appearing in the top 10 cutsets are where a component in the opposite train fails to perform its function. This is conservative because for the purpose of this one-time extension, the operability of the opposite train will be verified.

Since the seismic frequency is only 4.9E-5/yr (REPORT 04, p. A-10, *mean* value at 0.3 g = 300 cm/sec^2), it is concluded that the combination of potential seismic events, seismically-induced loss of other components, and a concurrent 1J1 bus outage will not contribute significantly to the cutsets in this analysis for either CDF or LERF. It is recognized that during a seismic event other failures may occur, resulting in higher LOOP CCDP. However, given the current state-of-the-art in seismic analysis, a seismic event is postulated to disable redundant components. As a result, although the CCDP value may increase, the delta CCDP is not expected to increase significantly.

Also, similar to the fire hazard, it is judged that the risk insights for this one-time extension are not impacted by this qualitative assessment of the seismic hazard, on the basis that:

1) There is a significant margin in the calculated figures of merit.
2) The expected out of service time of the MCC is about 52 hours.
3) The most significant contributor to the risk (e.g., the reduction in the capability to use RHR for the decay heat removal), is not required for most seismic induced plant transients.
4) Given the current state of knowledge in seismic PRA, where redundant components are assumed failed for a given seismic level, the risk estimate due to the unavailability of one redundant MCC would not be appreciably higher.

3.9 Tornado Analysis

A tornado strike is likely to produce a long-term LOOP that is not readily recovered, due to potential severe damage in the switchyard; therefore it is similar to a LOOP event due to other causes, with little or no opportunity for recovery. As stated in section 3.8, the unavailability of the 1J1 MCC is estimated to result in an increase of 7.5E-6 in the LOOP's CCDP. Since the tornado frequency is only 1.94E-4/yr (IPEEE, p. 5-8) and the fact that the unavailability of the 1J1 MCC would not have a significant impact on the operability of the functions credited for mitigating consequences of a Tornado event (e.g., AFW function and the emergency power function (e.g., Emergency Diesel generators), it is concluded that the change in risk will be insignificant.

3.10 RG 1.177 Tier 2: Avoidance of Risk Significant Plant Configurations

There is reasonable assurance that risk-significant plant equipment configurations will not occur when 1J1 MCC is out of service using the proposed Technical Specification changes based on the following:

<u>Technical Specifications and Safety Function Determination Program</u>

Adhering to the current Technical Specification requirements will prevent many of the more risk significant configurations from being entered into. Specifically, there are requirements concerning the operability/availability of emergency buses as specified in TS LCO 3.8.9. Potential configurations that should be avoided while the MCC is out of service are the unavailability of the redundant emergency buses and components supported by the H bus.

The Safety Function Determination Program (SFDP) requires provisions for cross-division checks to ensure a loss of the capability to perform a safety function assumed in the accident analysis does not go undetected. TS LCO 3.0.6 establishes requirements regarding supported systems when support systems are found inoperable. Upon entry into TS LCO 3.0.6 an evaluation is required to determine whether there has been a loss of safety function. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of TS LCO 3.0.6. Procedure O-GOP-9.4 implements the SFDP.

<u>Risk Management and Compensatory Actions</u>

The risk associated with having 1J1 MCC will be managed by adhering to the requirements for online risk assessment and management as described in the Dominion procedure NF-AA-PRA-370. In addition to the risk directly associated with the MCC unavailability, the procedure requires that potentially risk significant configurations during the period of its unavailability are assessed and managed. Based on a review of dominant cutsets these additional risk management actions and restrictions which will be used include:

- AFW pump 3A will be verified operable and signs will be posted.
- The MOVs on the TDAFW and MDAFW 3B's flow path will be verified to be open, kept open, and de-energized.

- An Operator will be designated as the AFW valve Operator and will be deployed to manually operate the de-energized AFW MOVs, if needed.
- A sound-powered phone system will be installed for communication between the AFW pump house and the Control Room.
- Unit 2 charging pumps and both Auxiliary Service Water pumps will be verified operable.
- Risk awareness briefings for maintenance and operations personnel will be performed prior to the work.
- Maintenance will be performed around-the-clock to minimize the time spent with equipment unavailable.
- Verification of redundant equipment operability and posting of signs.
- Disallow elective work that may cause a trip hazard or that may result in the unavailability of other equipment within the scope of (a)(4) program. For example, a Maintenance Operating Procedure (MOP) will be utilized to control the removal of MCCs 1J1-2S and 2N electrical loads, to ensure compliance with Technical Specifications, and to verify that safety function is maintained throughout the extended 72 hour Completion Time. This includes posting of signs for protected equipment.
- Fire watches will be established in the Cable Vault and Tunnel area and the Service Water Pump House (Fire Area 12).
- Pressurizer Operated Relief Valve (PORV) 1-RC-PCV-1456 will be placed in manual operation to prevent the possibility of a Small Break LOCA should the PORV cycle automatically and fail to re-seat while the PORV block valve (1-RC-MOV-1535) is de-energized open.
- A contingency plan will be added to applicable Electrical Maintenance procedures to expedite re-energizing the MCCs, if needed.

3.11 Tier 3 Risk-Informed Plant Configuration Control and Management

Dominion's 10 CFR 50.65(a)(4) program fully satisfies the requirements of Regulatory Guide 1.177 Tier 3. RG 1.177 Section 2.3 states that "The licensee should develop a program that ensures that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity. A viable program would be one that is able to uncover risk-significant plant equipment outage configurations in a timely manner during normal plant operation."

The Dominion (a)(4) program performs full PRA analyses of all planned maintenance configurations in advance. Configurations that approach or exceed the NUMARC 93-01 risk limits (e.g., 1.0E-6 for CDP) are avoided or addressed by compensatory measures. Historically, North Anna rarely approaches these limits. Emergent configurations are identified and analyzed by the on-shift staff for prompt determination of whether risk management actions are needed. The configuration analysis and risk management processes are fully proceduralized in compliance with the requirements of (a)(4).

North Anna's 10 CFR 50.65(a)(4) compliance program requires analysis and management of all configuration risks. The emergency power system is included in the (a)(4) scope and any component unavailability is monitored, analyzed and managed. When a configuration approaches the (a)(4) risk limits, plant procedures direct the implementation of risk management actions in compliance with the regulation. If the configuration is planned, these steps are taken in advance.

The proposed 1J1 bus outage is not expected to approach the required risk management thresholds of the (a)(4) regulation. While combinations of unavailable equipment and/or evolutions, including a 1J1 bus outage, may approach the limits and even require risk management actions, the risks arising from these configurations will be dominated by factors other than the MCC. As a result, the risk significance of the proposed 1-EE-MCC-1J1 tagout does not warrant limitations upon other equipment. Nevertheless, this analysis has assumed that no concurrent, planned maintenance will be performed and the analysis will be invalidated otherwise. This limitation is applicable to <u>planned</u> maintenance only, and not emergent. (This analysis <u>has</u> accounted for the closed pressurizer PORV block valve.)

Regulatory Guide 1.177 also refers to the Tier 3 program as a *Configuration Risk Management Program.*

4.0 RESULTS

4.1 Summary of the Risk Measures (RG 1.174 & 1.177 Tier 1)

The baseline *Core Damage Frequency* (CDF) and *Large Early Release Frequency* (LERF), from the N105A internal events and flooding, zero-maintenance model, are CDF = 4.49E-6/year and LERF = 7.83E-7/year, from the North Anna model notebook QU.2, Rev. 3, page 4.

The North Anna N105A model includes both internal events and flooding. The fire risks have been analyzed as well. The seismic and tornado risk contributions were evaluated and found to be negligible. These tables provide the combined CDFs and LERFs in detail.

Table 4.1-1
CDF and LERF Summary for Tier 1 Analysis

	CDF (yr^{-1})				LERF (yr^{-1})			
	Internal Events & flooding	**Fire**	**Seismic & Tornado**	**Total**	**Internal Events & flooding**	**Fire**	**Seismic & Tornado**	**Total**
Baseline (zero maintenance model)	4.49E-6	3.91E-6	Not quantified	8.40E-6	7.83E-7	Not quantified	Not quantified	7.83E-7
1-EE-MCC-1J1 out of service	1.07E-5	1.0E-5	Not quantified	2.07E-5	1.57E-6	Base case + 4.11E-7	Not quantified	1.57E-6 + Base case fire + 4.11E-7

The base case fire risk was taken from the NAPS non-seismic IPEE (p. 1-7). Fire LERF was not quantified. No seismic IPEEE was performed for North Anna and thus no numbers can be reported. The annual average tornado risk was screened out because of its low value.

Table 4.1-2
Results of Tier 1 Analyses

	Core Damage Risk	**Large Early Release Risk**
Internal Events Risk	CDF = 8.40E-6/yr***	LERF = 7.83E-7/yr*** + Fire LERF
Risk with 1-EE-MCC-1J1 OOS	CDF = 2.07E-5/yr	LERF = 1.57E-6/yr+Base case LERF + 4.11E-7/yr
Risk increase	ΔCDF = 1.23E-5/yr	ΔLERF = 7.9E-7/yr + 4.11E-7/yr = 1.20E-6/yr
Risk increase per CT entry (72 hours)	ICCDP = 1.01E-7	ICLERP = 9.86E-9
Annual Average Risk Increase with 1J1 OOS	CDF = 1.01E-7/yr	LERF = 9.86E-9/yr
* RG 1.177 classifies a change as "Small" when the ICCDP is < 5E-7 and the ICLERP is < 5E-8.		
** RG 1.174 classifies these changes as "Small" due to their combined baseline risk and expected risk increase.		
*** CDF values are Internal Events, Internal Flooding and Fire hazards. The LERF value is for the Internal Events, Internal Flooding, and Fire Hazards.		

There have been several previously approved, risk-informed Technical Specifications changes at North Anna. These changes and their cumulative risk impacts are tabled below, in addition to the currently proposed TS change.

Table 4.2-3
Summary of Approved or Pending NAPS Risk-Informed Technical Specifications Changes

Risk-informed TS Change	**Reference**	**Annual CDF Increase**	**Annual LERF Increase**
1J1 MCC 72-hour CT (one time only)	TSCR #N-075 (NAPS.RA.LI.8)	No permanent impact	No permanent impact
14-day underground fuel oil storage tank CT (one time only)	N-058 (NAPS.RA.LI.6)	No permanent impact	No permanent impact
RPS and ESF actuation system analog channel surveillance test internal extensions from monthly to quarterly and Completion Time extensions	TSCR #N-038 (ET NAF 98-0200, Rev. 0)	3E-07/yr (1% of baseline risk)	Not quantified
Supplemental RPS/ESFAS functions	TSCR #N-038 supplemental (SM-1317 Table 1 & SM-1290,	3E-09/yr	3E-10/yr

Table 4.2-3
Summary of Approved or Pending NAPS Risk-Informed Technical Specifications Changes

Risk-informed TS Change	Reference	Annual CDF Increase	Annual LERF Increase
	Rev. 0)		
7-day inverter Completion Time extension	TSCR #N-012 (SM-1360)	8.1E-08/yr	4.6E-10/yr
14-day Emergency Diesel Generator (EDG) CT	TSCR #318B (SM-0969, Rev. 0)	1.3E-06/yr	1.3E-07/yr *
14-day N2 backup supply for PORVs	TSCR #323 (ET NAF 95-0018, Rev. 0 & ET NAF 98-0202, Rev. 0)	Not quantified	Not quantified
Total		1.7E-6/yr	1.3E-07/yr

* Not quantified. LERF was conservatively estimated as 10% of the CDF change.

The cumulative ΔCDF of the proposed and previously approved TS changes is 1.7E-6/yr and the cumulative ΔLERF is 1.3E-7/yr. According to RG 1.177, the cumulative annual increases in CDF and LERF are still "small."

4.2 Assessment of Key Uncertainties

This section evaluates (qualitatively or quantitatively) the key areas of the modeling uncertainty, termed "epistemic uncertainty" in [RG 01], in the emergency power system analyses.

<u>Emergency Power Distribution System Model</u>

The North Anna PRA model has been peer reviewed. All of the "A" and "B" *Findings and Observations* (F&Os) were resolved or found to have no impact upon the proposed Technical Specification (TS) change. Documentation of the resolution of the B-significance F&Os is provided in PRA Notebook volume [NB 03]. The potential impact of Regulatory Guide 1.200 quality issues on this application are addressed in this notebook.

In addition, the model has already been successfully used in several risk-informed Technical Specifications CT extensions, three of which were specifically applicable to the emergency power distribution system. Further, the two-unit model has been used for 10 CFR 50.65(a)(4) configuration risk analysis since 2000, and its results have been scrutinized by the operations staff on a daily basis. This feedback has been incorporated into the PRA model and contributed to its steady improvement. As a result, modeling uncertainties have been conservatively addressed or are negligible.

External Events

While the uncertainty associated with the external events is larger than internal events, the results of the external event calculations indicate significant increases in the external events risk could occur without any impact on the risk metrics. For example, from the seismic analysis point of view, this conclusion is reached on the basis that, as stated in section 3.8, given the state-of-the-art for seismic analysis, all redundant components are failed for a given seismic event. Therefore, the potential failure of a portion of one train to function automatically due to the unavailability of a set of MOVs is not going to be impacted by the uncertainties.

Overall Uncertainty

The analyses performed in support of this application contain significant conservatisms (e.g., conservatisms in the Fire PRA model to compensate for the potential optimisms in the model.. Despite these conservatisms, the combined risk impact of all initiating events was below the applicable RG 1.177 threshold for classification as "small." The LERP and the CDP calculations have a factor of ~5 margin. Therefore, any modeling uncertainties have to increase these figures of merit by a factor of 5 to reach the LERP and CDP limits. It is judged that, given that

- this is a one-time extension,
- the modeling uncertainties are highly likely to impact the base case as well as the 1J1 case,
- and the restrictions put on the maintenance activities,

the impact of all uncertainties on the final results is negligible for this proposed Technical Specifications change.

5.0 CONCLUSIONS

This risk evaluation supports a one-time 72-hour *Completion Time* for the North Anna 480 VAC 1J1 Motor Control Center. The increases in annual *Core Damage* and *Large Early Release Frequencies* associated with the proposed extension of the Technical Specification Completion Time are characterized as "small changes" by Regulatory Guide 1.174. The *Incremental Conditional Core Damage* and *Large, Early Release Probabilities* associated with the proposed Technical Specification CT meet the acceptance criteria in Regulatory Guide 1.177.

Sensitivity calculations were not performed because the analysis includes significant conservatism and still demonstrated substantial margin to the limits of Regulatory Guides 1.174 and 1.177. No further assessment of modeling uncertainty is required.

The Regulatory Guide 1.174 requirement to "Track Cumulative Impacts" for "small changes" is satisfied by the Dominion model maintenance program procedures.

6.0 REFERENCES

[CODE 01] WinNUPRA, Version 3.0, Software Document File

[NB 01] North Anna Power Station Units 1 and 2 Probabilistic Risk Assessment Model Notebook Part III, PRA Model Development Category QU - Quantification Volume QU.2, "Model Quantification Results," Revision 2, North Anna N105A Model, March 2007

[NB 02] North Anna Power Station Units 1 and 2 Probabilistic Risk Assessment Model Notebook Part IV, Appendix B, "PRA Quality Summary Notebook," Revision 0 for N105A model, August 2007

[NB 03] North Anna Power Station Units 1 and 2 Probabilistic Risk Assessment Model Notebook Part IV, Appendix A, "PRA Model Reviews," March 2006

[NB 04] North Anna Power Station Units 1 and 2 Probabilistic Risk Assessment Model Notebook Part IV, Appendix A.1, "Internal Events Model Independent Assessment," August 2007

[NB 05] Calculation 5T45.NF02, North Anna Fire Qualitative Screening, 12/18/94

[NB 06] Calculation 5T45.NF05, NAPS Fire Quantitative Screening, 05/25/93

[NB 07] Calculation 5T45.NF08.1, CV&T Detailed Analysis, approved 12/94

[NB 08] Calculation 5T45.NF08.2, Detailed Fire Scenario PRA ESGR, 07/94

[NB 09] Calculation 5T45.NF08.3, Detailed Fire Scenario PRA AB (aux building), 06/94

[NB 10] Calculation 5T45.NF08.4, Detailed Fire Scenario PRA Main Control Room, 12/94

[NB 11] Calculation 5T45.NF08.8, Detailed Fire Scenario PRA HEP Analysis, 06/20/94

[NB 12] Draft Deterministic Analysis which will be included as an attachment to the LAR

[NB 13] Appendix F.3 of the NAPS IPE (Mode of Induced Primary System Failure DET)

[NB 14] Calculation SM-1237, Surry and North Anna Containment Isolation Modeling, April 2001

[NB 15] NAPS PRA Notebook SY.3.CD, Containment Cooling and Sprays System Analysis Model, Rev. 1

[NB 16] Calculation SM-1237, Surry and North Anna Containment Isolation Modeling, Addendum 1

[REPORT 01] WinNUPRA User's Manual, Version 2.1, Scientech, Inc., April 2001

[REPORT 02] "Individual Plant Examination Of Non-Seismic External Events And Fires - North Anna Power Station Units 1 And 2," Virginia Electric And Power Company, 1994

[REPORT 03] "North Anna Power Station Units 1 and 2 Report on Individual Plant Examination of External Events (IPEEE) – Seismic Prepared in Response to USNRC Generic Letter 88-20 Supplements 4 and 5," May 1997

[REPORT 04] NUREG-1488, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," April 1994

[REPORT 05] North Anna Power Station Probabilistic Safety Assessment Peer Review Certification Report, July, 2001

[REPORT 06] WCAP-15674, Rev. 0, "Common Cause Failure Analysis Improvements Project," April, 2001

[REPORT 07] Electric Power Research Institute, EPRI TR-100370, Fire Induced Vulnerability Evaluation (FIVE), April

[REPORT 08] NUREG-2300, NRC (U.S. Nuclear Regulatory Commission), 1983. PRA Procedures Guide, NUREG/CR-2300, American Nuclear Society and Institute of Electrical and Electronic Engineers, January

[REPORT 09] NUREG-2815, NRC (U.S. Nuclear Regulatory Commission), 1985. Probabilistic Safety Analysis Procedures Guide. NURGE/CR-2815, Brookhaven National Laboratory, Vols. 1 and 2, August

[REPORT 10] NSAC-181, Nuclear Safety Analysis Center (NSAC), Electric Power Research Institute, Fire PRA Requantification Studies, Palo Alto, CA, January

[REPORT 11] COMPBRN, Electric Power Research Institute, "Oconee PRA, A Probabilistic Risk Assessment of Oconee Unit 3," NSAC-60, Palo Alto, California, 1994.

[REPORT 12] NUREG/CR-5088, NRC (U.S. Nuclear Regulatory Commission), 1989. Fire Risk Scoping Study, Sandia National Laboratory, January

[REPORT 13] NUMARC-1993 Nuclear Management and Resources Council, Letter from William H. Rasin, Revision 1 to EPRI Final Report, dated April 1992, TR-100370, Fire – Induced Vulnerability Evaluation Methodology, September 29th

[REPORT 14] NAPS Non-Seismic IPEEE report, Appendix A, List of Event Trees. 1994

[REPORT 15] EPRI-TR-1000894, Fire Events Database for U.S. Nuclear Power Plants, Oct. 2000

[REPORT 16] "Surry Power Station Units 1 and 2 Report on Individual Plant Examination of External Events (IPEEE)" – 1994

[RG 01] Regulatory Guide 1.174, Revision 1, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis," November 2002

[RG 02] Regulatory Guide 1.177, Revision 0, "An Approach For Plant-Specific, Risk-Informed Decision Making: Technical Specifications," August 1998

[RG 03] Regulatory Guide 1.200, Revision 2, "An Approach For Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities," March 2009

[STD 01] ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," April 2002

[STD 02] ASME RA-Sa-2003, "Addenda to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," December 2003

[STD 03] ASME RA-Sb-2005, "Addenda to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," December 2005

[DRW 01] FE drawing 11715_FE-1R

Attachment A – Justification of Volume Change

Based on comments from the NRC, a more detailed evaluation of fire risk and GAP analysis is included in Revision 1.

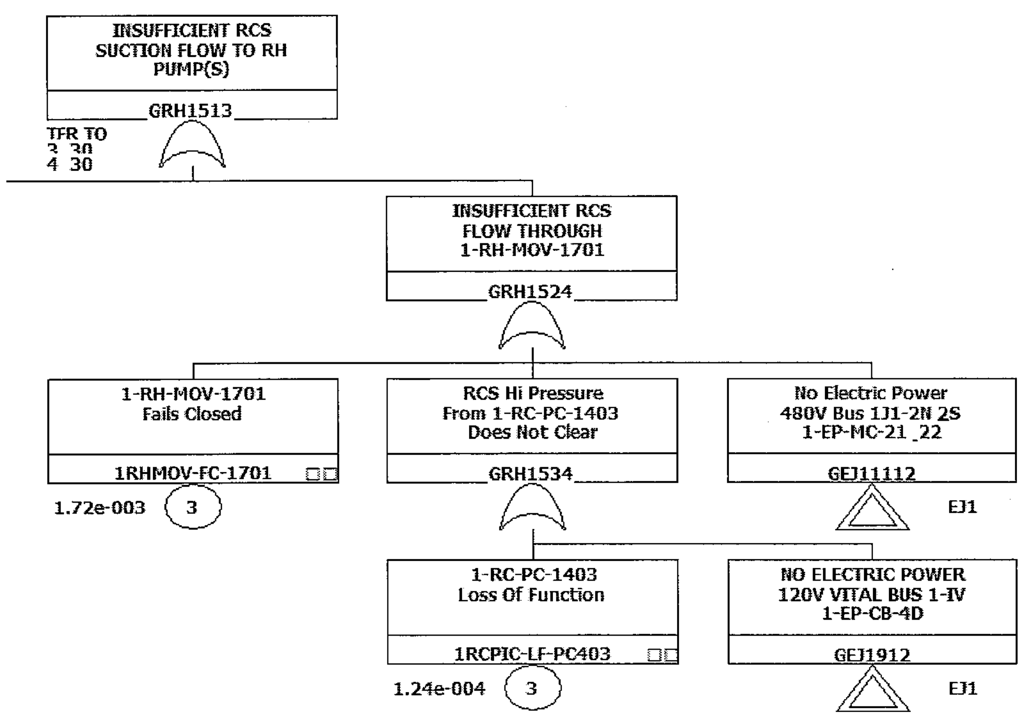
Revision 2 includes clarifications, especially in the Tier 2 section.

Attachment B – Description of Model Changes

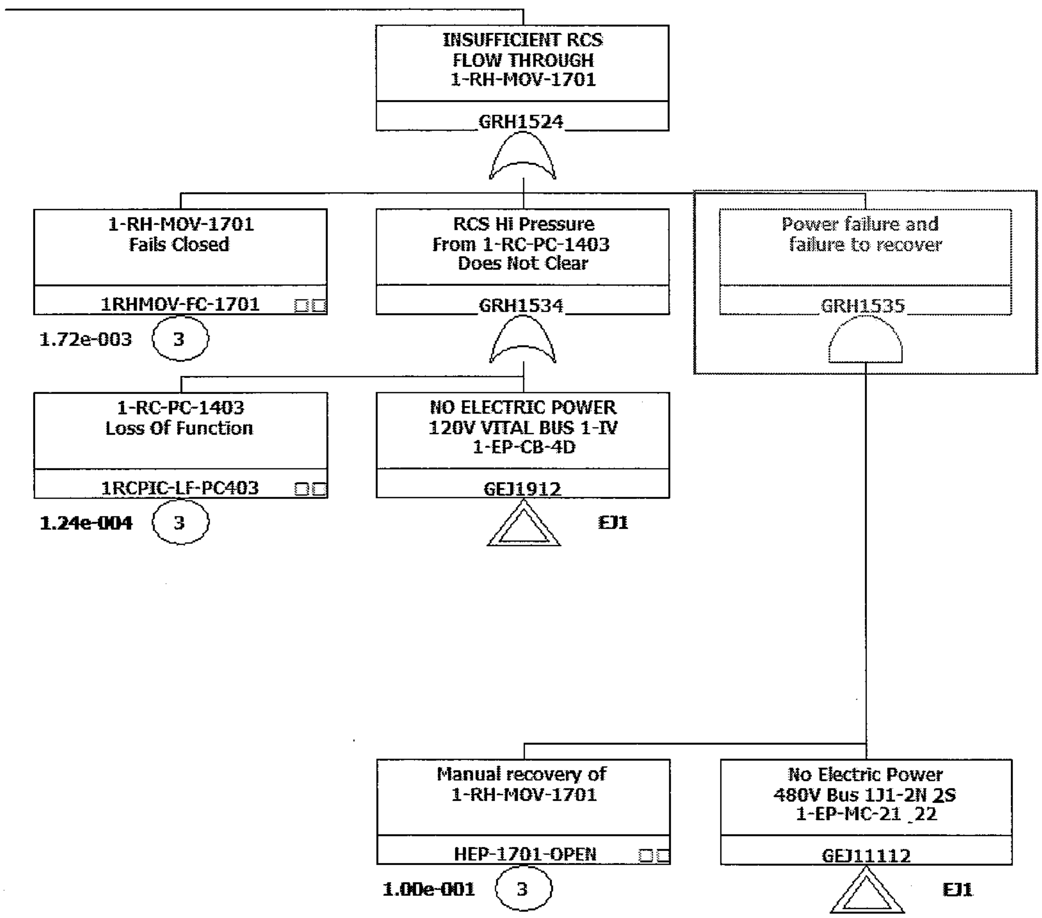
Several model changes were made in order to perform the risk calculation. The basic event 1EEBUS-LU-1J1-2 was changed from a "running" failure to a failure on demand (Type 3) variable, and its point estimate was reset to 1.0. It was not "logically" failed in order to permit evaluation of the cutsets in which it appeared. (A basic event set to LOGICAL failure does not appear in the WinNUPRA cutsets.)

In addition, a recovery was added to the RHR inlet Motor Operated Valve 1-RH-MOV-1701, which is powered off of the 1J1 bus (Reference 11715-FE-1R, Rev. 33). The current model logic is as follows:

NAPS UNIT 1 N0A/C MOD A		
Analyst: MGM	Creation Date: 11-11-1998	Revision: 10-18-2005



This logic was revised to add a recovery of the bus, in the absence of electrical power, by sending an operator to manually de-clutch the valve and open it by hand. The revised fault tree logic is as follows:



This MOV is in containment. In order to provide a high level of assurance that the recovery would occur, the station has committed to implement a standing order, for the duration of the one-time extended CT entry, with text similar to the following:

While the unit is in the extended Completion Time for the 1J1 Motor Control Center tagout, designate an operator with the responsibility to enter containment and open 1-RH-MOV-1701 manually if a Steam

Generator Tube Rupture occurs. This action should occur within 2 hours after the unit is cooled below 350 degrees F.

In order to estimate the associated *Human Error Probability*, the SPAR worksheets were filled out as shown in Attachment D. An operator error probability of 2E-4 was quantified; as a conservative measure, a value of 0.01 was used in the BASIC EVENT DATA (BED) file during the model solution. (The data file overwrites the 0.1 figure initially tested in the fault tree above as a screening value.)

In order to account for the presently-closed pressurizer PORV block valve 1-RC-PCV-1536, the failed-closed (FC) term for this valve was set to 1.0 and the failed-open term was set to 0.0.

Attachment C – Reviewer Comments / Resolutions

Comment Number	Section/Page	Review Comment	Response to Review Comment
1	Various	I have reviewed the changes and verified they are correct. The results are consistent with what would be expected due to the changes. I verified the calculations of the delta CDF and LERF and the ICCDPs and ICLERPs. All issues from the review were either incorporated or resolved by the Preparer.	All reviewer comments have been addressed.
2	3.7	Please incorporate editorial comments and include header/page numbers	Incorporated
3	Section 3.7 page 10	Based on our earlier discussion, please verify the following statement: "*Inter-area fire propagation analysis was not required based on the review of the fire area boundaries performed to address the Fire Risk Scoping Study, NUREG/CR-5088 (NRC, 1989) issues.*"	This is a statement which was made in the IPEEE study (Section 4, page 4-1, of the NAPS IPEEE report). In this analysis, unless there is evidence to the contrary, we are relying on the IPEEE report and we are not verifying the information provided in that report. Therefore, no verification is necessary.

Comment Number	Section/Page	Review Comment	Response to Review Comment
4	3.7	Please address JSS's comment on "*Why not use more recent data like NUREG/CR-6850 or Fire SDP 0609 App F?*"	The version of NAPS.RA.LI.8 (1J1 TC) which contains fire CCDP in table 3.7.1-1 based on the EPRI-TR-1000894 (Fire Events Database for U.S. Nuclear Power Plants, Oct. 2000) for the following four areas: 3-1, 9A-1, 10C and YARD. Since frequencies have been updated since the EPRI 2000 report, a sensitivity study was perform using the NRC SDP App. F fire ignition frequencies. As can be seen below, no fire compartments moved from "screened" to "unscreened" as determined by a threshold of 1E-6. **3-1** Z-27-1 Unit 1 Motor Generator Set House, Fire frequency 3.4E-3 per year increases to 1E-2 (assuming switchgear room from SDP App. F) -> Fire CDF = 1E-2 * 8.6E-7 = 8.6E-9 per year, which is **still below 1E-6 threshold** **9A-1** EDG-1H Unit 1 Emergency Diesel Generator 1H, Fire frequency = 3.9E-2 per year decreases to 3E-2 (SDP App. F) -> Fire CDF = 3E-2 *4.29E-7 = 1.3E-8, which is **still below 1E-6** **10C** MCC Fuel Oil Pump House Motor Control Center Room Fire frequency = less than 0.1. Note SDP App. F does not provide an Ignition Frequency for Fuel Oil Pump house so assume conservative 0.1. **No change** **YARD** Z-8C, XFMRS Main and Station Service Transformers fire freq

Comment Number	Section/Page	Review Comment	Response to Review Comment
5		Intentionally left blank.	
6	Section 3.4/ Pg 8	The text in item iv appears to duplicated the text at the end of item iii.	Agree. Deleted the text at the bottom of item iii.
7	Section 3.7.1.1/ Pg 12	In discussing adjustments to the criterion that the normal alternate shutdown path is unavailable, it is state 1J1 is unavailable, then 1H is available. Clarify if these conditions are combined with AND or OR. Also clarify why availability of bus 1H is included (e.g., is it for redundant train) since it is currently mentioned without context.	Added clarifications.
8	Section 3.7.1.1/ Pg 13	Several areas are listed as screened out because they contain no SSD equipment and this there would be no challenge to SSD equipment supported by 1J1. Is it also known that there are no related cables in these screened areas?	YES.
9	Section 3.7.1.1/ Pg 14/2nd bullet	States that risk for 1H EDG room calculated based on manual shutdown event, fire IEF for EDG and 1J1 MCC set to 1, it seems 1H EDG failure should also be set to 1.	Made editorial changes to better represent the approach used for the quantification of the 1H EDG room.
10	Section 3.7.1.1/ Pg 14/Item 2	States that operator actions from internal events analysis that are performed outside areas in fire evaluation would be set to zero if fire is evaluated to prevent action. It seems HEP should be set to 1 if fire prevents the action.	Depends on whether we talk in the failure space of success space. In the failure space the comment is correct but in the success space the wording is correct. The wording is changed to clarify the statement. Also note that for the screening stage, our analysis did not show any operator actions which will be performed in the area that is postulated to be affected by the fire scenario under the consideration.

Comment Number	Section/Page	Review Comment	Response to Review Comment
11	Section 3.7.1.1/ Pg 15	There seems to be missing text in calculations of the turbine trip IE and LOOP IE, following "(From run" and "(See run"	Yes. Will be adjusted prior to release.
12	Section 3.7.1.1/ Pg 16	In the first part of the results discussion, specify what the criterion/threshold was for quantitative screening fire areas/compartments. From Table 3.7.1-1 that threshold appears to be 1E-6. A basis for that value should also be provided.	To be consistent with the IPEEE study, the same threshold as the IPEEE was used (which is 1.0E-6). There is no need to restate the IPEEE criterion.
13	Section 3.7.1.3/ Page 16	Detailed analysis was performed for five (not four) areas, the four from the IPEEE and the one screened back in.	The sentence refers to the IPEEE study. Added clarification.

Comment Number	Section/Page	Review Comment	Response to Review Comment
14	Section 3.7.1.3/ Page 17	Condition 3 states that if a fire does not affect 1J1 MCC or a component that supports 1J1 MCC, then the PRA model would be solved twice. In the actual analysis however this is not consistently done. Specifically the rod drive room compartment meets this condition and the model should be rerun rather than simply assuming the IPEEE CCDP estimate bounds. The Aux Building and Control Room also meet this condition but it is simply concluded 1J1 OOS will have no impact on CDF.	Adjusted the wording to reflect the approach which was actually used. Ideally we would have run the applicable IPEEE fire PRA models. From the level of effort, defensibility, and consistency points of views, it would be the preferred approach. However, the IPEEE fire PRA model is not available for use and based on a discussion with Management, re-instating the fire PRA model would take considerable amount of time (using a similar effort for re-instating the SPS fire PRA model as a measuring stick). Arguably the second best approach is the approach stated under condition 3. However, based on a review of the available documentation, it was judged that a significant number of changes were made to the internal events model to reflect the impact of fire on the equipment and the operator action credited in the internal events model. Re creating all those changes seemed to be a lengthy process with significant opportunity to introduce mistakes. Therefore, it was judged that the best approach is to use the results of the IPEEE detailed analysis to gain a conservative estimate of the risk increase. It is judged that the final results show this approach has provided a conservative but reasonable estimate of risk increase.
15	Section 3.7.1.3-1/ Pg 17	There seems to be missing text in the section on Service Bldg CV&T, following "(See"	Yes.

Comment Number	Section/Page	Review Comment	Response to Review Comment
16	Section 3.7.1.3-1/ Pg 19	In the second bullet it is noted that the HHSI function is not affected by 1J1 OOS because Unit 2 HHSI is credited. (This assertion is also used throughout the analysis.) Need to clarify/note whether the Unit 2 HHSI provides support via any portion of the U1 HHSI flowpath that may include valves powered by 1J1 MCC.	As noted in the first bullet on the same page, an Appendix R fire in the CV&T area is postulated to result in a loss of 1J1 MCC. Therefore, the safe shutdown assessment for this area already includes an evaluation of the HHSI flow path due to the unavailability of the 1J1 MCC. Although the fire PRA scenarios may assume a more limited damage and as such the 1J1 MCC unavailability may have an impact on the CDF estimate, it is judged that such scenarios are a subset of all events. Therefore, the CCDP multiplier and the verification of the availability of the alternate path compensates for any optimistic consideration with respect to the reliability of any manual action that may be credited for the HHSI flow path.
17	Section 3.7.1.3-2/ Pg 21	Conclusions 1 and 2 listed are simply the same conclusion applied to two different groups of sequences.	OK
18	Section 3.7.1.3-4/ Pg 23	Item 2 notes the manual local operation of a number of valves is credited in the IPEEE and thus in this analysis as well. Some justification should be provided for continuing to credit all of these actions in light of current guidance to credit manual actions only where needed and justified.	This is not a revalidation of the IPEEE study. Although, for either a permanent LAR, or CT change for a risk significant component, the suggested action may be desirable, for this one time change, it is judged that the CCDP multiplier is adequate to address any optimism in the results.

Comment Number	Section/Page	Review Comment	Response to Review Comment
19	Section 3.7.1.3-5/ Pg 23	This appears to be a calculation of the baseline CDF, in which case CCDP should include only failure of MDAFW 3A (the value with 1J1 also OOS is from Table 3.7.1-1). It is unclear if just the text is incorrect, or if the model calculation was incorrect as well. In addition, there seems to be missing text in calculation of the turbine building delta CDF, following "(See run".	A text error. Corrected.
20	Section 3.7.2 / Pg 24	The items listed do not really seem to be conservatisms, they are more ground rules for the analysis which are not necessarily conservative. Some of the conservatisms that should be listed instead are: - In revisiting the screening of the fire areas the base-case fire CDF is typically neglected when evaluating delta CDF. - CCDP is increased by a factor of 2 or 5 to account for the impact of fire on HEPs - No credit for fire severity factor in assigning fire IEFs to an area for this analysis	The statement pertains to the qualitative assessment of the un-screened areas (section 3.7.1.3) not the assessment of the qualitatively screened areas. The comments here are true for section 3.7.1.1
21	Section 3.7.3 / Pg 26	The estimated failure probability for containment isolation at Surry is used for the NAPS fire-induced LERF calculation. Provide justification as to why the Surry value is applicable.	As stated in section 3.7.3, the probability of NAPS containment failure has been determined to be negligible (all paths were screened out). However, as a "conservative measure" I wanted to use a value. NAPS and SPS containment isolation are similar (if not identical) therefore, SPS's value is used. I am not sure it is necessary to add any additional justification than is already stated in the notebook.

Comment Number	Section/Page	Review Comment	Response to Review Comment
22	Table 3.7.1-1 / Pg 31	Compartment BR-I and BR-II. It is not clear if the calculated CCDP also includes failure of the battery, since it is the same value as a turbine trip.	It does. I think it (the result) means that our model calculates a negligible increase in CDF due to failure of a battery following a simple turbine trip IE.
23	Table 3.7.1-1 / Pg 35	Compartment ESGR/CR-1. The conclusion is made that the contribution of 1J1 OOS on fire risk is negligible, yet it is stated that a fire would require plant shutdown and the HVAC equipment affected by the fire would impact both SSD trains. It seems this conclusion can only be made if it is known that the equipment affected by 1J1 OOS is the same equipment affected by the HVAC failure, and this is not addressed.	The "impact analysis" reaches its "negligible impact" conclusion on the basis that the HVAC failure would concurrently impact both trains of safe shutdown. Therefore, the unavailability of 1J1 is inconsequential. No change is needed.
24	Table 3.7.1-1 / Pg 38	Compartment EDG-1H. It is not clear if the calculated CCDP also includes failure of the 1H EDG, since it is the same value as a turbine trip. In addition, applying a factor of 2 to account for fire impact on HEPs is not done as for the other areas evaluated.	This fact is noted in the Table. It seems to me that it means for a Loss of Feedwater IE, with maintenance on EDG 1J restricted, and MCC 1J1 unavailable, then the availability of the 1H EDG is not significant. I have not verified if the Internal events model is correct but our reviewers have not found the answer from the run to be incorrect. The CCDP value is corrected.
25	Table 3.7.1-1 / Pg 41-43	Compartments 11A – 11C. The analysis presented in the table (which simply concludes fire contribution with 1J1 OOS is negligible) does not appear to be consistent with the approach outlined in the text for these compartments (pg. 13 second sub-bullet)	Agree. The text on page 13 is modified to take care of this comment.

Comment Number	Section/Page	Review Comment	Response to Review Comment
26	Table 3.7.1-1 / Pg 51	Compartment MDAFW-1. The wrong numbers are given for CDF and delta CDF. Should be 9.82E-7 (instead of 4.982E-7) and 2E-9 (instead of 2E-7).	Corrected.
27	Table 3.7.1-1 / Pg 53	Compartment QPSH-1. Instead of concluding that impact of QS pumps on risk for this area is negligible would be more defensible to set QS pumps to 1.0 in calculation of CCDF (along with MDAFW pumps and 1J1).	Agree but will not change the results or insights. So, no change is made.
28	Table 3.7.1-1 / Pg 54	Compartment Z-16-1. This refers to the analysis for QPSH-1 but it is not clear if this is because MDAFW cables are routed here also. In any case, it seems a CCDP should be calculated for this compartment with MDAFW, ORS and LHSI pumps all set to 1, along with 1J1 OOS.	Agree.
29	Section 3.8 & 3.9 / Pg 65	The seismic and tornado analyses both note that cutsets with LOOP and 1J1 OOS are very rare in the internal events analysis. However, this could be due to the low average value used for 1J1 TM, and the prevalence of these cutsets could increase when 1J1 is set to 1 as for the new CT. Some discussion of this is needed to show that setting 1J1 to 1 would be offset by the low seismic/tornado IEF such that cutsets with 1J1 OOS remain nonsignificant.	Agree- Section Modified.
30	Section 4.1 / Pg 69	Following Table 4.2-3 it is stated that the proposed change is assumed to contribute to the annual average risk even though it is a one-time change. But the totals in Table 4.2.-3 do not include the delta CDF for the proposed change.	The statement is deleted.

Comment Number	Section/Page	Review Comment	Response to Review Comment
31	Section 4.3 / Pg 70	Clarify the statements regarding uncertainty in the external events analysis. Saying that significant increases in the external events risk could occur without any impact on the risk metrics does not make sense given that the base fire CDF is nearly the same as the base internal events CDF.	From the seismic analysis point of view, this conclusion is reached on the basis that, as stated in section 3.8, given the state-of-the-art for seismic analysis, all redundant components are failed for a given seismic event. Therefore, the potential failure of a portion of one train to function automatically due to the unavailability of a set of MOVs is not going to be impact by the uncertainties.
32	Section 4.3 / Pg 70	Clarify the paragraph on overall uncertainty, as it has no context and is difficult to understand – e.g., what is the assumption about RH MOV recovery and where do the stated margin factors come from? See comment 33 also.	Done.
33	Attachment B/ Pg. 74	This attachment is never mentioned in the body of the report, it would be helpful to discuss it in the actual analysis to give some of the calculations more context.	Done (see additions to section 3.5)
34	Attachment E	Items IE-C8 and IE-C12. It could also be noted that a change in IEF due to changes in CCF treatment in IE fault trees will not impact delta CDF for this analysis, since the IE change will be reflected in both the baseline CDF and CDF with 1J1 failed.	Agree. Done

Comment Number	Section/Page	Review Comment	Response to Review Comment
35	Attachment E	Item AS-B5A states that availability of the opposite train and the credited Unit 2 Charging pumps are not in question for this analysis. It is noted in this analysis that no planned maintenance will be concurrently scheduled during the one-time extended outage for 1J1. Clarify if this prohibition of planned maintenance will include Unit 2 components that are credited (i.e., the HHSI pumps and support systems).	Agree. It is covered in the Tier 2 discussion.
36	Attachment E	Item SY-A11 states that dual-unit initiating events are not addressed. Since this analysis credits the Unit 2 Charging pumps some discussion should be provided that dual-unit IEs would not be a concern since Unit 2 HHSI is credited in the fire analysis portion of this evaluation, and a postulated fire would not affect both units.	The gap analysis pertains to the internal events analysis not fire analysis. For fire analysis, the Unit 2 components are credited based on the App R and fire PRA where duel unit accidents are not postulated.
37	Attachment E	For several items the comments regarding applicability to the 1J1 AOT include a statement along the lines of "Based on the familiarity/ knowledge of the NAPS original model, xxxxx was considered/ generally addressed during model development." Suggest this statement be removed since it does not preclude applicability of the assessment comments to this AOT submittal, since the PRA assessment comments indicate such considerations were not documented, thus the reason for the comments.	Disagree. The statement (i.e., "Based on the familiarity…") is made for SRs that it appears the issue of concern was raised due to lack of documentation. The reason for the statement is to show that, based on knowledge of the model, we really believe this is a documentation issue not an issue that may become a modeling issue once the documentation is complete.

Comment Number	Section/Page	Review Comment	Response to Review Comment
38	Attachment E	Item IF-B2, omission of human-induced floods is a PRA completeness issue rather than a documentation issue. For this analysis however, it can be noted that the addition of new human-induced flood scenarios will not impact delta CDF for this analysis, since the IE change will be reflected in both the baseline CDF and CDF with 1J1 failed. This same comment could also apply to other IF items deemed documentation only.	Again, my recollection is that we did address human-induced (characterized as maintenance-induced) flooding. Therefore, I have tried to indicate that the SR comment was made because the reviewers did not find adequate documentation. Hence, the comment about the issue being a documentation issue. However, the suggested statement here appears to be included in the text already.

Comment Number	Section/Page	Review Comment	Response to Review Comment
39	Attachment E	For several items the comments regarding applicability to the 1J1 AOT include a statement along the lines of "Based on a review of the comments in "Assessment Comments column, it is not clear why this SR is characterized as "Not Met."" It is suggested that the Model/Documentation Enhancement Recommendations from the PRA assessment be included in Attachment E as these more specifically identify what is needed in order to fully meet each ST.	No change is needed. These statements are meant to highlight the fact that the issue raised in the SR seems to be an enhancement not a reason for characterizing the SR is not meeting the requirements. As stated in section 3.4 and verified with previous conversations with Ray and Tom as well as Ray's input for KPS's SAMA gaps, it is understood that the approach for the gap assessment (per Dominion instruction) was to "develop a comprehensive list of all potential areas for improvement and to be aggressive in pursuing model enhancement by conservatively characterizing a Supporting Requirement (SR) as "Not Met" if one or more areas for improvement were identified. This conservative philosophy is different than that which is used for PRA model peer reviews that are performed in accordance with NEI 05-04, Revision 2, where "findings" and "suggestions" are used to characterize such observations. Using this conservative philosophy, although the preponderance of evidence points to meeting the applicable SR at Category II level, the assessment characterized a number of SRs as not meeting Capability Category II requirements."

Comment Number	Section/Page	Review Comment	Response to Review Comment
40	Section 3.4, Page 9	Item ix states that all of the IF related gap items are documentation related. However, a number of the items in Attachment E for IF are described as non-documentation issues. Also, comment 38 also notes another IF item that is not documentation-related. The wording in item ix should be revised to more accurately reflect the analysis in Attachment E.	Agree. Done.
41	Various, including Section 5.0	As the fire risk has been estimated from the IPEEE information (which doesn't reflect current fire PRA methods and phenomena), it might be appropriate to acknowledge that the fire assessment is by necessity incomplete. However, even with the numerous conservatisms included in the fire assessment, the overall risk of the bus outage is still well below the RG 1.177 threshold. So, even if a more detailed fire assessment was performed, reflecting current methods, there is high confidence that the risk would still be below the acceptance threshold.	Agree. Indeed the last few paragraphs in section 3.7.2 are meant to make this case.

Attachment D – HEP Calculation for 1-RH-MOV-1701 Recovery

HRA Worksheets for At-Power

SPAR HUMAN ERROR WORKSHEET

Plant: NAPS Initiating Event: SGTR Basic Event: HEP-1701-OPEN Event Coder: ___________

Basic Event Description: Manually open 1-RH-MOV-1701 in containment

Does this task contain a significant amount of diagnosis activity?

YES ☐ (start with Part I - Diagnosis) NO ☒ (skip Part 1 - Diagnosis; start with Part II - Action) Why?

The operators are well trained on identifying a SGTR, and no other diagnosis is required.

3.1 PART I. EVALUATE EACH PSF FOR DIAGNOSIS

A. Evaluate PSFs for the Diagnosis Portion of the Task, If Any.

PSFs	PSF Levels	Multiplier for Diagnosis		Specific Reasons
Available Time	Inadequate time	P (failure)=1.0	☐	
	Barely adequate time (~2/3 x nominal)	10	☐	
	Nominal Time	1	☐	
	Extra time (between 1 and 2x nominal and > 30 min)	0.1	☐	
	Expansive time (> 2 x nominal and > 30 min)	0.01	☐	
	Insufficient Information	1	☐	
Stress/ Stressors	Extreme	5	☐	
	High	2	☐	
	Nominal	1	☐	
	Insufficient Information	1	☐	
Complexity	Highly complex	5	☐	
	Moderately complex	2	☐	
	Nominal	1	☐	
	Obvious diagnosis	0.1	☐	
	Insufficient Information	1	☐	
Experience / Training	Low	10	☐	
	Nominal	1	☐	
	High	0.5	☐	
	Insufficient Information	1	☐	
Procedures	Not available	50	☐	
	Incomplete	20	☐	
	Available, but poor	5	☐	
	Nominal	1	☐	
	Diagnosis/symptom oriented	0.5	☐	
	Insufficient Information	1	☐	
Ergonomic s/HMI	Missing/Misleading	50	☐	
	Poor	10	☐	
	Nominal	1	☐	
	Good	0.5	☐	
	Insufficient Information	1	☐	
Fitness for Duty	Unfit	P (failure)=1.0	☐	
	Degraded Fitness	5	☐	
	Nominal Time	1	☐	
	Insufficient Information	1	☐	
Work Processes	Poor	2	☐	
	Nominal	1	☐	
	Good	0.8	☐	
	Insufficient Information	1	☐	

Reviewer: ___________

B. Calculate the Diagnosis Failure Probability.

(1) If all PSF ratings are nominal, then the Diagnosis Failure Probability = 1.0E-2
(2) Otherwise, the Diagnosis Failure Probability is: 1.0E-2 x Time x Stress or Stressors x Complexity x Experience or Training x Procedures x Ergonomics or HMI x Fitness for Duty x Processes

Diagnosis: 1.0E-2 x ___ x ___ x ___ x ___ x ___ x ___ x ___ x ___ = _________

C. **Calculate the Adjustment**

When 3 or more negative PSF influences are present, in lieu of the equation above, you must compute a composite PSF score used in conjunction with the adjustment factor. Negative PSFs are present anytime a multiplier greater than 1 selected. The Nominal HEP (NHEP) is 1.0E-2 for Diagnosis. The composite PSF score is computed by multiplying all the assigned PSF values. Then the adjustment factor below is applied to compute the HEP:

$$HEP = \frac{NHEP \cdot PSF_{composite}}{NHEP \cdot \left(PSF_{composite} - 1\right) + 1}$$

Diagnosis HEP with Adjustment Factor = _________

D. Record Final Diagnosis HEP.

If no adjustment factor was applied, record the value from Part B as your final diagnosis HEP. If an adjustment factor was applied, record the value from Part C.

Final Diagnosis HEP = __n/a__

PART II. EVALUATE EACH PSF FOR ACTION

A. Evaluate PSFs for the Action Portion of the Task, If Any.

PSFs	PSF Levels	Multiplier for Action		Specific Reasons
Available Time	Inadequate time	P (failure)=1.0	☐	There is substantial time available to identify the task and complete it. RHR is not required until at least 4 hours after event initiation.
	Time available is ~ the time required	10	☐	
	Nominal Time	1	☐	
	Time available > 5x the time required	0.1	☒	
	Time available > 50x the time required	0.01	☐	
	Insufficient Information	1	☐	
Stress/ Stressors	Extreme	5	☐	Containment entry required, although adverse environment is not expected.
	High	2	☒	
	Nominal	1	☐	
	Insufficient Information	1	☐	
Complexity	Highly complex	5	☐	The evolution is simple.
	Moderately complex	2	☐	
	Nominal	1	☒	
	Insufficient Information	1	☐	
Experience/ Training	Low	3	☐	The operators at North Anna are trained at a high level and include many veterans.
	Nominal	1	☒	
	High	0.5	☐	
	Insufficient Information	1	☐	
Procedures	Not available	50	☐	The instructions are written, clear and concise.
	Incomplete	20	☐	
	Available, but poor	5	☐	
	Nominal	1	☒	
	Insufficient Information	1	☐	
Ergonomics /HMI	Missing/Misleading	50	☐	The valve location is known and the valve is clearly tagged.
	Poor	10	☐	
	Nominal	1	☒	
	Good	0.5	☐	
	Insufficient Information	1	☐	
Fitness for Duty	Unfit	P (failure)=1.0	☐	The operators all work routinely under a FFD program.
	Degraded Fitness	5	☐	
	Nominal	1	☒	
	Insufficient Information	1	☐	
Work Processes	Poor	5	☐	
	Nominal	1	☒	
	Good	0.8	☐	
	Insufficient Information	1	☐	

Reviewer:____________

B. Calculate the Action Failure Probability.

(1) If all PSF ratings are nominal, then the Action Failure Probability = 1.0E-3
(2) Otherwise, the Action Failure Probability is: 1.0E-3 x Time x Stress or Stressors x Complexity x Experience or Training x Procedures x Ergonomics or HMI x Fitness for Duty x Processes

Action: 1.0E-3 x _0.1_ x _2_ x _1_ x _1_ x _1_ x _1_ x _1_ x _1_ = __2E-4_

C. **Calculate the Adjustment Factor If Negative Multiple (≥ 3) PSFs are Present.**

When 3 or more negative PSF influences are present, in lieu of the equation above, you must compute a composite PSF score used in conjunction with the adjustment factor. Negative PSFs are present anytime a multiplier greater than 1 selected. The Nominal HEP (NHEP) is 1.0E-3 for Diagnosis. The composite PSF score is computed by multiplying all the assigned PSF values. Then the adjustment factor below is applied to compute the HEP:

$$HEP = \frac{NHEP \cdot PSF_{composite}}{NHEP \cdot (PSF_{composite} - 1) + 1}$$

Action HEP with Adjustment Factor = _n/a___

D. Record Final Diagnosis HEP.

If no adjustment factor was applied, record the value from Part B as your final diagnosis HEP. If an adjustment factor was applied, record the value from Part C.

Final Action HEP = __2E-4____

PART III. CALCULATE TASK FAILURE PROBABILITY WITHOUT FORMAL DEPENDENCE ($P_{W/OD}$)

Calculate the Task Failure Probability Without Formal Dependence ($P_{w/od}$) by adding the Diagnosis Failure Probability from Part I and the Action Failure Probability from Part II. In instances where an action is required without a diagnosis and there is no dependency, then this step is omitted.

$P_{w/od}$ = Diagnosis HEP ___n/a___ + Action HEP ___2E-4____ = _2E-4____

PART IV. DEPENDENCY

For all tasks, except the first task in the sequence, use the table and formulae below to calculate the Task Failure Probability With Formal Dependence ($P_{w/d}$).

If there is a reason why failure on previous tasks should not be considered, such as it is impossible to take the current action unless the previous action has been properly performed, explain here: ____________________

Dependency Condition Table

Condition Number	Crew (Same or different)	Time (close in time or not)	Location (Same or different)	Cues (additional or not)	Dependency	Number of Human Action Failures Rule ☐ - Not Applicable Why? ____________
1	S	C	S	NA	complete	When considering recovery in a series e.g., 2nd, 3rd, or 4th checker, If this error is the **3rd error in the sequence**, then the dependency is at least **moderate**. If this error is the **4th error in the sequence**, then the dependency is at least **high**.
2				A	complete	
3			D	NA	high	
4				A	high	
5		NC	S	NA	high	
6				A	moderate	
7			D	NA	moderate	
8				A	low	
9	D	C	S	NA	moderate	
10				A	moderate	
11			D	NA	moderate	
12				A	moderate	
13		NC	S	NA	low	
14				A	low	
15			D	NA	low	
16				A	low	
17					zero	

Using $P_{w/od}$ = Probability of Task Failure Without Formal Dependence (calculate in Part III):

For Complete Dependence the probability of failure is 1.
For High Dependence the probability of failure is $(1 + P_{w/od}) / 2$
For Moderate Dependence the probability of failure is $(1 + 6 \times P_{w/od}) / 7$
For Low Dependence the probability of failure is $(1 + 19 \times P_{w/od}) / 20$
For Zero Dependence the probability of failure is $P_{w/od}$

Calculate $P_{w/d}$ using the appropriate values:

$$P_{w/d} = (1 + (\_\_\_\_\_\_\_\_ + \_\_\_\_\_\_\_\_)) / \_\_\_\_\_\_\_\_ = \_\_\_\_\_\_\_\_$$

Reviewer: ________

Attachment E – GAP Analysis Results

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IE-A3	REVIEW the plant-specific initiating event experience of all initiators to ensure that the list of challenges accounts for plant experience. See also IE-A7	No	The NAPS PRA Model Notebook IE.2, "Initiating Event Quantification", (Revision 1, March 2006) reviewed all plant shutdown events documented in Licensee Event Reports during the life of the plant. The review led to the identification of the T2A initiator, a recoverable form of the T2 Loss of Main Feedwater (MFW) initiator. However, as noted in the comments for IE-A5 and IE-A7, the following events don't appear to be included in the evaluation: 1) those that have occurred during shutdown conditions, or have resulted in a controlled shutdown, or 2) initiating event precursors.	Although new initiating events may be identified, based on the experience with dealing with this comment, it is judged that 1) the accident progression for these potential initiating events is similar to the progression for initiating events already included in the model and 2) the frequency of these newly identified potential initiating events is lower than the existing initiating event frequencies. Therefore, the impact on this analysis is negligible.
IE-A4	PERFORM a systematic evaluation of each system, including support systems, to assess the possibility of an initiating event occurring due to a failure of the system. USE a structured approach (such as a system-by-system review of initiating event potential, or an FMEA [failure modes and effects analysis] or other systematic process) to assess and document the possibility of an initiating event resulting from individual systems or train failures.	No	A systematic evaluation of each NAPS system, including support systems, to assess the possibility of an initiating event occurring due to a failure of the system was performed and documented in Section 2.2 of the NAPS PRA Model Notebook IE.1, "Initiating Event Identification and Grouping," (Rev. 1, February 2006). However, as noted by the RG 1.200 clarification, the search for initiators should go down to the subsystem/train level where necessary. The analysis of electrical power is an area where the reviewer recommends the search for initiators be documented at the bus / panel level. Also, the potential for administrative shutdown should also be further reviewed. For example, the evaluation for the loss of component cooling water provides consideration for the loss of RCP seal cooling but doesn't discuss the possibility for an administrative shutdown due to technical specification requirements.	Although new initiating events may be identified, based on the experience with dealing with this comment, it is judged that 1) the accident progression for these potential initiating events is similar to the progression for initiating events already included in the model and 2) the frequency of these newly identified potential initiating events is lower than the existing initiating event frequencies. Therefore, the impact on this analysis is negligible.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IE-A4a	When performing the systematic evaluation required in IE-A4, INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause, and from routine system alignments.	No	For the systematic evaluation required in IE-A4, the examination of potential initiating events resulting from common cause failures for some systems is included, such as the complete loss of SW; however the evaluation is not complete. For example, considerations for common cause failure of more than one electrical bus have not been documented. Also, document the considerations for system alignments resulting from preventive and corrective maintenance.	Although new initiating events may be identified, based on the experience with dealing with this comment, it is judged that 1) the accident progression for these potential initiating events is similar to the progression for initiating events already included in the model and 2) the frequency of these newly identified potential initiating events is lower than the existing initiating event frequencies. Therefore, the impact on this analysis is negligible.
IE-A5	In the identification of the initiating events, INCORPORATE (a) events that have occurred at conditions other than at-power operation (i.e., during low-power or shutdown conditions), and for which it is determined that the event could also occur during at-power operation. (b) events resulting in a controlled shutdown that includes a scram prior to reaching low-power conditions, unless it is determined that an event is not applicable to at-power operation.	No	The NAPS PRA Notebook IE.1 incorporates initiating events that have occurred during full or low power operations into the IE evaluation. However, events that have occurred during shutdown conditions, or that have resulted in a controlled shutdown don't appear to have been included in the evaluation.	Although new initiating events may be identified, based on the experience with dealing with this comment, it is judged that 1) the accident progression for these potential initiating events is similar to the progression for initiating events already included in the model and 2) the frequency of these newly identified potential initiating events is lower than the existing initiating event frequencies. Therefore, the impact on this analysis is negligible.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IE-A6	INTERVIEW plant personnel (e.g., operations, maintenance, engineering, safety analysis) to determine if potential initiating events have been overlooked.	No	No documentation of plant personnel interviews to determine if potential initiating events have been overlooked was found in the PRA notebooks.	Although new initiating events may be identified, based on the experience with dealing with this comment, it is judged that 1) the accident progression for these potential initiating events is similar to the progression for initiating events already included in the model and 2) the frequency of these newly identified potential initiating events is lower than the existing initiating event frequencies. Therefore, the impact on this analysis is negligible.
IE-A7	REVIEW plant-specific operating experience for initiating event precursors, for the purpose of identifying additional initiating events. For example, plant specific experience with intake structure clogging might indicate that loss of intake structures should be identified as a potential initiating event.	No	No documentation of the review of plant-specific operating experience for initiating event precursors was found in the PRA notebooks.	Although new initiating events may be identified, based on the experience with dealing with this comment, it is judged that 1) the accident progression for these potential initiating events is similar to the progression for initiating events already included in the model and 2) the frequency of these newly identified potential initiating events is lower than the existing initiating event frequencies. Therefore, the impact on this analysis is negligible. Also, this could be a documentation issue only
IE-B1	COMBINE initiating events into groups to facilitate definition of accident sequences in the Accident Sequence Analysis element (para. 4.5.2) and to facilitate quantification in the Quantification element (para. 4.5.8).	No	Section 2.5 of IE.1 documents the grouping of initiating events. The NAPS initiating events grouping essentially complies with this SR, but the documentation should be clarified.	Documentation only
IE-C1	CALCULATE the initiating event frequency accounting for relevant generic and plant-specific data unless it is justified that there are adequate plant-specific data to characterize the parameter value and its uncertainty. (See also IE-C11 for requirements for rare and extremely rare events).	No	Based on a review of Sections 2.3 and 2.4 of NAPS PRA Model Notebook IE.2, "Initiating Event Data Analysis," (Rev. 2, December 2005), initiating event frequencies have been calculated using relevant generic and plant-specific data. Generic data is from NUREG/CR-5750 "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995", February 1999. It is recommended that generic frequencies be updated when revision to NUREG/CR-5750 becomes available. Also, as noted by this review and the WOG Peer Review, the ISLOCA frequency is based on methods that are not current.	This is a suggestion. The SR is basically met. No impact on the PRA model is expected.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IE-C4	USE as screening criteria no higher than the following characteristics (or more stringent characteristics as devised by the analyst) to eliminate initiating events or groups from further evaluation: (a) the frequency of the event is less than 1E-7 per reactor-year (/ry) and the event does not involve either an ISLOCA, containment bypass, or reactor pressure vessel rupture (b) the frequency of the event is less than 1E-6/ry and core damage could not occur unless at least two trains of mitigating systems are failed independent of the initiator, or (c) the resulting reactor shutdown is not an immediate occurrence. That is, the event does not require the plant to go to shutdown conditions until sufficient time has expired during which the initiating event conditions, with a high degree of certainty (based on supporting calculations), are detected and corrected before normal plant operation is curtailed (either administratively or automatically). If either criterion (a) or (b) above is used, then CONFIRM that the value specified in the criterion meets the applicable requirements in the Data Analysis section (para. 4.5.6) and the Level 1 Quantification section (para. 4.5.8).	No	NAPS initiating event screening does not employ the minimum criteria cited by this SR. For example, losses of systems that require administrative shutdown are screened in IE.1 without a basis. For screening purposes, such events must be demonstrated to not require the plant to go to shutdown conditions until sufficient time has expired during which the initiating event conditions, with a high degree of certainty (based on supporting calculations), are detected and corrected before normal plant operation is curtailed. As another example, it is not clear that the quantitative argument for screening the loss of CC to the RCP internal coolers meets the minimum criteria for this SR.	Although new initiating events may be identified, based on the experience with dealing with this comment, it is judged that 1) the accident progression for these potential initiating events is similar to the progression for initiating events already included in the model and 2) the frequency of these newly identified potential initiating events is lower than the existing initiating event frequencies. Therefore, the impact on this analysis is negligible. Also, this could be a documentation issue only

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IE-C8	If fault-tree modeling is used for initiating events, CAPTURE within the initiating event fault tree models all relevant combinations of events involving the annual frequency of one component failure combined with the unavailability (or failure during the repair time of the first component) of other components.	No	The fault tree models used to calculate initiating event frequencies appear to mod1el all relevant combinations of events involving the annual frequency of one component failure combined with the unavailability (or failure during the repair time of the first component) of other components. However, the fault trees apply a factor to CCF failure to run events that adjusts the CCF terms based on the mean time to repair of the first failed component. That is, when the first component fails, the mission time that the next components are exposed to the common cause term is the mean time to repair of the first component. This approach doesn't follow a fundamental PRA modeling assumption that CCFs occur relatively close in time. The database development for CCF terms has already made the assessment as to whether a particular historical event has involved mechanisms that could be expected to impact components within a relatively close time period (see quote from NUREG/CR-5485 below). Such a modeling basis ensures that common cause failure to run parameter estimates model mechanisms that could reasonably occur very close in time or at least within the 24-hour mission time. There is no sound logical basis to assume the mechanisms would not apply to a longer mission time (e.g., a 72-hour mean time to repair). NUREG/CR-5485: "For failures to become multiple failures from a shared cause, the conditions have to be conducive for the trigger event and/or the conditioning events to affect all components within the group simultaneously. The meaning of simultaneity in this context is that failures lead to inability of redundant components to perform their safety function within the appropriate mission time."	Although the frequency of an existing initiating events may be increased, based on the experience with dealing with this comment, it is judged that the increase in frequency will be minimal. Therefore, the impact on this analysis is negligible. Also, a change in IEF due to changes in CCF treatment in IE fault trees will not impact delta CDF for this analysis, since the IE change will be reflected in both the baseline CDF and CDF with 1J1 failed.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IE-C12	In the ISLOCA frequency analysis, INCLUDE the following features of plant and procedures that influence the ISLOCA frequency: (a) configuration of potential pathways including numbers and types of valves and their relevant failure modes, existence and positioning of relief valves (b) provision of protective interlocks (c) relevant surveillance test procedures (d) the capability of secondary system piping (e) isolation capabilities given high flow / differential pressure conditions that might exist following breach of the secondary system	No	The ISLOCA frequency calculation is based on methods that are not current as noted by this review and the WOG PRA Peer Review. An update to the ISLOCA frequency is planned.	The same comment was made about the ISLOCA frequency for the SPS model. The SPS model has since been updated to meet this SR (by the organization that made this comment). The new SPS ISLOCA frequency is not greater than the original ISLOCA frequency. Therefore, it is concluded that the impact of this comment on the 1J1 MCC one time extension is negligible. Also, a change in IEF due to changes in CCF treatment in IE fault trees will not impact delta CDF for this analysis, since the IE change will be reflected in both the baseline CDF and CDF with 1J1 failed.
IE-D1	DOCUMENT the initiating event analysis in a manner that facilitates PRA applications, upgrades, and peer review.	No	The initiating event analysis has been documented in a manner that facilitates PRA applications, upgrades, and peer review, with the exception of the areas where supporting requirements have not been met, as noted by this self assessment, as well as specific recommendations below.	Documentation only

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IE-D2	DOCUMENT the processes used to select, group, and screen the initiating events and to model and quantify the initiating event frequencies, including the inputs, methods, and results. For example, this documentation typically includes: (a) the functional categories considered and the specific initiating events included in each (b) the systematic search for plant-unique and plant-specific support system initiators (c) the systematic search for RCS pressure boundary failures and interfacing system LOCAs (d) the approach for assessing completeness and consistency of initiating events with plant-specific experience, industry experience, other comparable PRAs and FSAR initiating events (e) the basis for screening out initiating events (f) the basis for grouping and subsuming initiating events (g) the dismissal of any observed initiating events, including any credit for recovery (h) the derivation of the initiating event frequencies and the recoveries used (i) the approach to quantification of each initiating event frequency (j) the justification for exclusion of any data	No	The current PRA documentation satisfies this supporting requirement, with the exception of the enhancement recommendations for other IE supporting requirements.	Documentation only

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IE-D3	DOCUMENT the key assumptions and key sources uncertainty with the initiating event analysis.	No	While assumptions and sources of uncertainty are documented to some degree in the IE notebooks, a systematic review of uncertainty and assumptions that can impact the risk profile of the base PRA is not documented or referenced in the initiating events notebooks.	The internal events-based CDP calculation has margin of greater than 7 to the risk significance threshold. Therefore, the additional delta risk due to uncertainty would have to be greater than factor of 7 to reach the CDP limit. Also, the uncertainty affects both the base case and the 1J1 case. Therefore, from the delta point of view this "Not Met" SR may not result in a significant change. Therefore, it is concluded that the impact of this "Not Met" SR on this application is negligible.
AS-A2	For each modeled initiating event, IDENTIFY the key safety functions that are necessary to reach a safe, stable state and prevent core damage	No	Key safety functions that are necessary are identified in the Success Criteria Analysis, Volume SC.1. That notebook describes the safety functions for transients, ATWS, large, medium, and small LOCAs, and SGTR. Each specific transient event is not delineated and a clear tie from the Accident Sequence Analysis, Volume AS.1, is not provided. Although the interfacing systems LOCA and reactor vessel rupture are not modeled in detail, no discussion of safety functions for those events is provided.	This is basically a documentation issue. The SR is basically met. No impact on the PRA model is expected.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
AS-A3	For each modeled initiating event, using the success criteria defined for each key safety function (in accordance with SR SC-A4), IDENTIFY the systems that can be used to mitigate the initiator.	No	The systems used to meet each key safety function are delineated in Volume SC.1. No consideration of RCP seal integrity is provided in the systems needed to maintain the RCS integrity safety function even though RCP seal integrity is included on the event trees. No discussion is provided of how containment isolation valves impact the containment integrity safety function. No discussion of induced steam generator tube rupture is provided in either the RCS or containment integrity safety functions. The PRA models assume that bleed and feed cooling is failed when seal injection flow is not available. This assumption is conservative. However, additional justification should be provided on the quantitative impact of this assumption on risk-informed decision-making. Also, clarify whether this assumption applies to instances where seal cooling, i.e., thermal barrier cooling, is lost as opposed to seal injection.	This is a documentation issue. The reviewers have not identified any initiating event that its mitigation may be optimistically modeled. The only example provided here (feed and bleed) is judged to be conservatively modeled. Although the conservatism may mask the role played by the unavailability of the 1J1 MCC, the overall impact of conservatism is that the risk estimate will be higher. The higher delta risk will result in lower allowed CT. Therefore, the overall impact on the proposed CT extension is considered to be negligible.
AS-A4	For each modeled initiating event, using the success criteria defined for each key safety function (in accordance with SR SC-A4), IDENTIFY the necessary operator actions to achieve the defined success criteria.	No	The description of top event 1RC, feed and bleed initiation, in Volume AS.1 states that the operator opens two PORVs and starts one charging pump. These actions are different than described in FRH1 where first SI is initiated to start charging pumps and then PORVs are opened. The need for operator action to initiate high-head ECCS recirculation for bleed and feed is not clearly delineated in Volume AS.1 Also, the description of top event 1CHR mentions that there is a limit for sump water temperature to initiate recirculation. However, no discussion of the associated operator actions is provided. Operator actions are not included in Tables 5-1 through 5-6 of Volume SC.1 For SBO, the discussion of feedwater and safety injection use after power recovery states that automatic initiating signals would start the associated pumps. However, steps in ECA-0.0 direct that most pumps be placed in pull-to-lock so consideration of system restoration after power recovery should be given. The analysis for an induced SGTR following a steam or feedwater line break uses the same HEP as a random SGTR. No justification is provided as to why the cues and performance shaping factors are similar enough for both events. No discussion of the operator action needed to secure charging pumps following a steam line break is provided. Volume AS.1 does not clearly state that operator actions are required to isolate the faulted steam generator.	The SR requires that "for each key safety function (in accordance with SR SC-A4), IDENTIFY the necessary operator actions to achieve the defined success". The reviewers seem to challenge whether the IDENTIFIED operator actions for a few functions are correct or not. Therefore, it appears that this SR is met (i.e., the necessary operator actions are "IDENTIFIED"). This comment at best is an improvement and it does not show a systemic issue in the model. Therefore, there is no impact on the proposed extension of CT.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
AS-A7	DELINEATE the possible accident sequences for each modeled initiating event, unless the sequences can be shown to be a non-contribution using qualitative arguments.	No	The reason for not modeling sequence 10 of the T6 event tree states that the contribution would be negligible. However, the value of cutsets could be noticeable since success of node SL would be about 0.8. Also, there is no discussion of potential dependencies between the 1CH node and the nodes that follow SL. Similarly, TH sequence 22 is not modeled, but the failure probability of MFW of about 0.3 so the sequences not modeled could be noticeable. Several ATWS sequences are listed in assumption 23 of Volume AS.1 and are not developed because they have a low frequency. However, these sequences are excluded from core damage frequency. The potential for a random loss of offsite power following T9A, T9B, T23, and other events is not explicitly included in the event tree models. These failures may be excluded from core damage cutsets based on the DAM fault tree and, therefore, be underestimating CDF. Similarly, random losses of RCP seal cooling are not addressed in some transient event trees, e.g., T23. Although AS.1 indicates that this is because RCP seal failures are modeled as an separate initiating event, that justification is inconsistent with the treatment of switchgear room cooling failures that are considered as both random events and separate initiating events. The potential for a random opening of a safety relief valve is not addressed or included in the event tree models. Random failure of a SRV would transfer to a medium LOCA. Subsection 2.3.5.1 in AS.1 discusses seal LOCA modeling for the T6 event tree (SL node). No discussion is provided for the operator action that is modeled for the SL node (see functional equation 1SL-03). Also, if the seals do not “bind or pop”, a less significant seal LOCA will still occur according to the discussion, which could imply to a reviewer that LOCA mitigation is required. The T6 event tree takes credit for the installation of qualified O-rings, which appears to preclude the need for LOCA mitigation if the seals do not bind or pop (as discussed in AS.2). Subsection 2.3.9 states that "because of the low risk of MSLBs/MFLBs, no effort is made to separate the breaks into any of the three loops; instead, it is assumed that the break occurs on loop A; while this overstates the risk to some components and understates it to others, the effect is negligible because of low overall risk from MSLBs/MFLBs."	The example here related to the completeness of accident sequence modeling, but these sequences are for low frequency sequences, e.g., ATWS after a LOCA. Even if the exclusion of these low frequency sequences is a real issue, the 1J1 MCC unavailability is not expected to change the frequency of these sequences on the basis that the 1J1 MCC-supported components are mostly valves that have either redundant components or would be in the required accident mitigating configuration (e.g., AFW MOV), and can be manually locally operated. It should be noted that the frequency of the sequences discussed here will be even lower for this one time extension because, unlike the average model, the redundant components are available.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
AS-A9	USE realistic, applicable (i.e., from similar plants) thermal hydraulic analyses to determine the accident progression parameters (e.g., timing, temperature, pressure, steam) that could potentially affect the operability of the mitigating systems.	No	Volume AS.1 states that core damage is assumed if core exit thermocouple temperature exceeds 1200 degrees F. Volume SC.1 states that fuel integrity is assumed if fuel rod temperature is less than 2200 deg F. Both MAAP and RETRAN are used to analyze accident progression, however, the differing criteria for fuel integrity and core damage are not reconciled and no acceptance criteria for the codes are given.	Based on the experience in dealing with this issue, this is mostly a documentation issue and the results are not affected by this apparent gap. Also, it is noted that during MSPI, the NAPS success criteria were not identified as outliers.
AS-B1	For each modeled initiating event, IDENTIFY mitigating systems impacted by the occurrence of the initiator and the extent of the impact. INCLUDE the impact of initiating events on mitigating systems in the accident progression either in the accident sequence models or in the system models.	No	A limited discussion of systems that are impacted by the initiating event is provided in Volume AS.1.	Documentation only
AS-B3	For each accident sequence, IDENTIFY the phenomenological conditions created by the accident progression. Phenomenological impacts include generation of harsh environments affecting temperature, pressure, debris, water levels, humidity, etc. that could impact the success of the system or function under consideration [e.g., loss of pump net positive suction head (NPSH), clogging of flow paths]. INCLUDE the impact of the accident progression phenomena, either in the accident sequence models or in the system models.	No	Only a limited discussion of phenomenological conditions created by the accident progression is provided in Section 2.3 of Volume AS.1 For example, no discussion is provided on how a secondary line break outside containment affects the environmental conditions of equipment needed to mitigate the accident.	This is a documentation issue. The reviewers have not identified any initiating event that its mitigation may be optimistically modeled. The only example provided here

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
AS-B5a	If plant configurations and maintenance practices create dependencies among various system alignments, DEFINE and MODEL these configurations and alignments in a manner that reflects these dependencies, either in the accident sequence models or in the system models.	No	The NAPS models credit use of the opposite unit systems, e.g., charging system and diesel-generators, for accident mitigation. However, no documentation was identified that would show how opposite unit outages were considered. For example, during a refueling outage, a Train-A outage may make charging or CC cross-tie unavailable for a significant period of time. Such unavailability values could reach 5% overall. If unavailability during opposite-unit outages is included in the overall system unavailability, then that could be stated in the AS documentation.	This is a documentation issue only. In any case, there is not impact on this one time extension, when the availability of the opposite train and the credited Unit 2 Charging pumps are not in question.
AS-B6	MODEL time-phased dependencies (i.e., those that change as the accident progresses, due to such factors as depletion of resources, recovery of resources, and changes in loads) in the accident sequences. Examples are: (a) For SBO/LOOP sequences, key time-phased events, such as: (1) AC power recovery (2) DC battery adequacy (time-dependent discharge) (3) Environmental conditions (e.g., room cooling) for operating equipment and the control room (b) For ATWS/failure to scram events (for BWRs), key time-dependent actions such as: (1) SLCS initiation (2) RPV level control (3) ADS inhibit (c) Other events that may be subject to explicit time-dependent characterization include: (1) CRD as an adequate RPV injection source (2) Long term make-up to RWST	No	Only a limited discussion of the time-phased dependencies is provided in Volume AS.1. Loss of switchgear cooling is included as a top node in the NAPS event trees, however, failure of cooling does not result in an immediate loss of function. No discussion of the time available to cross-tie RCP seal cooling to the opposite unit is given. A brief discussion is provided of the need for cooling containment sump water if recirculation is needed. However, no discussion is provided of how this dependency is modeled. The time available to switch to ECCS recirculation is not discussed. The top event discussion for SBO does not include timing considerations for AC power recovery	Based on previous experience in dealing with this comment, this is a documentation issue. For example, the timing aspect of loss of SWGR cooling was analyzed and reanalyzed during IPE internal events and internal flooding development. No impact on the results of this LAR is expected. Also note that failure of the ESWGR cooling would disable the 1J1 MCC and therefore, there is no impact on the proposed extension for the 1J1 MCC CT.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
AS-C1	DOCUMENT the accident sequence analysis in a manner that facilitates PRA applications, upgrades, and peer review.	No	Documentation of the accident analysis is included in Volumes AS.1, AS.2, SC.1, and SC.2. The following assumption appears to be out of place in Subsection 2.3.1.2 of AS.1 and may be inaccurate. "3. It was assumed that a complete loss of power to an emergency bus would be a T9 initiator, whether the bus itself was lost, or if failure was due to loss of the normal power supply and failure of the emergency AC power supplies. It was assumed that if the normal power supply to the bus failed, but the emergency power was successful, there would be no initiating event." This assumption may be inaccurate from the perspective that Technical Specification LCOs may require plant shutdown prior to the time that normal power can be restored to the emergency AC buses with a high degree of certainty (based on supporting calculations). That is, a loss of the normal power supply to the emergency AC buses may need to be included as an initiating event to satisfy the Standard (IE-C4), which would invalidate this assumption. 2) The following assumption in Subsection 2.3.2.1 of AS.1 was confusing: "1RPS - Reactor Subcritical. (Formerly K) This top event is similar to the 1RPS function in Section 2.3.1, but is noted again here because of the unique house event BED file utilized to quantify the function. Because of the very short time frame involved for the SCRAM to occur, no credit is given to the AAC diesel, but also no failures are credible if they involve depletion of the batteries. Therefore, the house event BED files AAC and HOSNOBD are used to solve the function." The consideration of power availability does not seem to be applicable to 1RPS - Reactor Subcritical (RPS typically fails safe on power losses).	Documentation only

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
AS-C2	DOCUMENT the processes used to develop accident sequences and treat dependencies in accident sequences, including the inputs, methods, and results. For example, this documentation typically includes: (a) the linkage between the modeled initiating event in the Initiating Event Analysis section and the accident sequence model; (b) the success criteria established for each modeled initiating event including the bases for the criteria (i.e., the system capacities required to mitigate the accident and the necessary components required to achieve these capacities); (c) a description of the accident progression for each sequence or group of similar sequences (i.e., descriptions of the sequence timing, applicable procedural guidance, expected environmental or phenomenological impacts, dependencies between systems and operator actions, end states, and other pertinent information required to fully establish the sequence of events); (d) the operator actions reflected in the event trees, and the sequence-specific timing and dependencies that are traceable to the HRA for these actions; (e) the interface of the accident sequence models with plant damage states; (f) [when sequences are modeled using a single top event fault tree] the manner in which the requirements for accident sequence analysis have been satisfied.	No	A one-to-one correlation between each initiating event and the associated event tree is not clearly provided. The system success criteria and associated basis is not clearly provided. A discussion of the accident sequences will need to revised pending resolution of issues associated with other AS SRs. For example, the phenomenological conditions created by the accident. Operator actions needed are not clearly delineated along with any associated dependencies on system success or other operator actions (Refer to AS-B1, B3, and B6)	Documentation only

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
AS-C3	DOCUMENT the key assumptions and key sources of uncertainty associated with the accident sequence analysis.	No	No discussion of the sources of uncertainty associated with the accident progression is provided.	Documentation only

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SC-A2	SPECIFY the plant parameters (e.g., highest node temperature, core collapsed liquid level) and associated acceptance criteria (e.g., temperature limit) to be used in determining core damage. SELECT these parameters such that the determination of core damage is as realistic as practical, consistent with current best practice. DEFINE computer code-predicted acceptance criteria with sufficient margin on the code-calculated values to allow for limitations of the code, sophistication of the models, and uncertainties in the results, consistent with requirements specified under HLR-SC-B. Examples of measures for core damage suitable for Capability Category II / III, which have been used in PRAs, include: (a) collapsed liquid level less than 1/3 core height or code predicted peak core temperature >2,500°F (BWR) (b) collapsed liquid level below top of active fuel for a prolonged period; or code-predicted core peak node temperature >2,200°F using a code with detailed core modeling; or code-predicted core peak node temperature >1,800°F using a code with simplified (e.g., single-node core model, lumped parameter) core modeling; or code-predicted core exit temperature >1,200°F for 30 min using a code with simplified core modeling (PWR)	No	Core Damage is defined in the SC.1 notebook based on 10CFR50.46 LOCA criteria, which assumes a detailed core heatup evaluation is performed. However, many of the existing success criteria evaluations were done using MAAP, which is not capable of performing such evaluations. The SC notebooks need to present an alternate criteria that can be demonstrated for the MAAP runs or for any other evaluation codes/methods (e.g., for ATWS) that are used to determine if core damage has occurred. Also, the AS.1 notebook notes a different core damage criteria (>1200 degrees core exit temperature), which is different from what is noted in SC.1.	The comment from MAAP is at best a disputable comment. Also, it is noted that 1) at part of the MSPI model comparison, NAPS model was not identified as an outlier, 2) the success criteria would affect both the base case and the 1J1 case (as a result, the delta would not be affected as much). Therefore, it is judged that even if any changes were to be made, the impact will be minimal on the results supporting the proposed one time extension of 1J1 MCC CT.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SC-A4	SPECIFY success criteria for each of the key safety functions identified per SR AS-A2 for each modeled initiating event [Note (2)].	No	The success criteria discussion is inconsistent within the SC.1 notebook. Section 5.1 (acceptance criteria) discusses some but not all safety functions. For example, reactivity control is not discussed. The discussion of RCS integrity does not consider LOCAs through safety valves or RCP seal LOCA. In Section 5.2, a text description of the success criteria bases is provided for each initiating event, as is a tabular listing of the criteria used for each IE. The terminology used in this section differs somewhat from that in Section 5.1	This is a documentation enhancement.
SC-A4a	IDENTIFY mitigating systems that are shared between units, and the manner in which the sharing is performed should both units experience a common initiating event (e.g., LOOP).	No	The SC.1 notebook does not indicate if there are any shared mitigating systems, and if present, how they are considered in the SC development process. Unit-specific differences are noted, however, in the notebook	Based on the comment, this is, at best, a documentation issue. It appears that the reviewer expected that the SR to be met in the SC.1 note book as oppose to the system notebook. For example, based on previous experience, it is known that SW system is model as a shared system and the impact on the success criteria is addressed. It is judged that this SR is basically met.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SC-A5	SPECIFY an appropriate mission time for the modeled accident sequences. For sequences in which stable plant conditions have been achieved, USE a minimum mission time of 24 hr. Mission times for individual SSCs that function during the accident sequence may be less than 24 hr, as long as an appropriate set of SSCs and operator actions are modeled to support the full sequence mission time. For example, if following a LOCA, low pressure injection is available for 1 hr, after which recirculation is required, the mission time for LPSI may be 1 hr and the mission time for recirculation may be 23 hr. For sequences in which stable plant conditions would not be achieved by 24 hr using the modeled plant equipment and human actions, PERFORM additional evaluation or modeling by using an appropriate technique. Examples of appropriate techniques include: (a) assigning an appropriate plant damage state for the sequence; (b) extending the mission time, and adjusting the affected analyses, to the point at which conditions can be shown to reach acceptable values; or (c) modeling additional system recovery or operator actions for the sequence, in accordance with requirements stated in the Systems Analysis and Human Reliability sections of this Standard, to demonstrate that a successful outcome is achieved.	No	24 hours is used as the mission time for the PRA and discussed in the draft PRA Manual Success Criteria chapter. The PRA Manual notes that mission time will be extended if a stable operating state has not been achieved. From the information presented in the SC.1 notebook, it appears that 24 hours is sufficient to achieve a stable state (or Core Damage) for the initiating events. However, there is no explicit discussion of mission time within the SC.1 notebook itself.	This SR is basically met and the reason for characterizing this SR as "Not Met" is not clear. As the worst case, this may be a documentation issue only. No impact on the 1J1 LAR.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SC-A6	CONFIRM that the bases for the success criteria are consistent with the features, procedures, and operating philosophy of the plant.	No	The SC analyses discussed in SC.1 and SC.2 either are plant-specific analyses, or are Surry or vendor analyses that have been reviewed for applicability to NAPS. The SC documentation discusses primarily hardware features of the plant, and human actions do not appear to be discussed (e.g., No references to procedures, typical operator actions, etc,). Also see the discussion for SR AS-A4.	This SR is basically met and the reason for characterizing this SR as "Not Met" is not clear. As the worst case, this may be a documentation issue only. No impact on the 1J1 LAR.
SC-B1	USE appropriate realistic generic analyses/evaluations that are applicable to the plant for thermal/hydraulic, structural, and other supporting engineering bases in support of success criteria requiring detailed computer modeling. Realistic models or analyses may be supplemented with plant-specific/generic FSAR or other conservative analysis applicable to the plant, but only if such supplemental analyses do not affect the determination of which combinations of systems and trains of systems are required to respond to an initiating event.	No	The success criteria development (documented in notebooks SC.1 and SC.2) uses a combination of plant-specific and vendor analyses. In some cases UFSAR information is used, but the resulting success criteria seem to be reasonable as compared to those used in other plants. For seal LOCA and Offsite Power recovery, data used to develop these criteria appears to be dated and new information is available that might change results. However, it is not certain if the fault tree models themselves may have more up-to-date models than the documentation indicates.	This SR is basically met and the reason for characterizing this SR as "Not Met" is not clear. As the worst case, this may be a documentation issue only. No impact on the 1J1 LAR.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SC-B4	USE analysis models and computer codes that have sufficient capability to model the conditions of interest in the determination of success criteria for CDF, and that provide results representative of the plant. A qualitative evaluation of a relevant application of codes, models, or analyses that has been used for a similar class of plant (e.g., Owner's Group generic studies) may be used. USE computer codes and models only within known limits of applicability.	No	The current success criteria are based on various plant-specific and industry analyses. Some of these analyses are based on the MAAP code, which may not adequately model accident response. While the current success criteria seem to be reasonable, the basis for these criteria may not fully meet the requirements of Capability Category II. Also, the notebooks do not discuss how the currently-utilized codes were used within the limits of their applicability.	This SR is basically met and the reason for characterizing this SR as "Not Met" is not clear. As the worst case, this may be a documentation issue only. No impact on the 1J1 LAR. Also based on previous experience in dealing with similar comments, it is not expected that significant changes will be made when closing this comment.
SC-B5	CHECK the reasonableness and acceptability of the results of the thermal/hydraulic, structural, or other supporting engineering bases used to support the success criteria. Examples of methods to achieve this include: (a) comparison with results of the same analyses performed for similar plants, accounting for differences in unique plant features (b) comparison with results of similar analyses performed with other plant-specific codes (c) check by other means appropriate to the particular analysis	No	The SC notebooks do not contain any comparisons to other plants (other than references to Surry, on which many of the NAPS criteria are based) or other reasonableness or comparison checks. Since Surry does not constitute an independent source of data, this SR is not met.	Based on dealing with similar comment, no significant changes are expected. Also, it is noted that during PWROG MSPI model comparison, the NAPS model was not identified as an outlier. Therefore, it is concluded that this gap have has no impact on the 1J1 MCC LAR.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SC-C2	DOCUMENT the processes used to develop overall PRA success criteria and the supporting engineering bases, including the inputs, methods, and results. For example, this documentation typically includes: (a) the definition of core damage used in the PRA including the bases for any selected parameter value used in the definition (e.g., peak cladding temperature or reactor vessel level) (b) calculations (generic and plant-specific) or other references used to establish success criteria, and identification of cases for which they are used (c) identification of computer codes or other methods used to establish plant-specific success criteria (d) a description of the limitations (e.g., potential conservatisms or limitations that could challenge the applicability of computer models in certain cases) of the calculations or codes (e) the uses of expert judgment within the PRA, and rationale for such uses (f) a summary of success ceria for the available mitigating systems and human actions for each accident initiating group modeled in the PRA (g) the basis for establishing the time available for human actions (h) descriptions of processes used to define success criteria for grouped initiating events or accident sequences	No	The current SC.1 and SC.2 notebooks address some of the specific items noted in this SR. However, documentation needs to be added concerning how the timing for human actions was developed (if not discussed in the HR notebooks - the timing is presented from MAAP but not discussed further) and the limitations of the computer codes used on the analysis results.	Documentation only
SC-C3	DOCUMENT the key assumptions and key sources of uncertainty associated with the development of success criteria.	No	The SC.1 and SC.2 notebooks include listings of assumptions. However, the notebooks do not provide any discussion of the sources of uncertainties in the SC area	Documentation only

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SY-A2	COLLECT pertinent information to ensure that the systems analysis appropriately reflects the as-built and as-operated systems. Examples of such information include system P&IDs, one-line diagrams, instrumentation and control drawings, spatial layout drawings, system operating procedures, abnormal operating procedures, emergency procedures, success criteria calculations, the final or updated SAR, Technical Specifications, training information, system descriptions and related design documents, actual system operating experience, and interviews with system engineers and operators.	No	NAPS PRA generally meets this SR since most of the pertinent information is documented in the IPE, however the documentation specifically documented in the SY.3 notebooks. Drawings were referenced in SY.1. There was no reference or documentation found of any interviews with system engineers or operators.	The comments here reflect enhancements, if anything. The SR is basically met. No impact on the 1J1 LAR.
SY-A3	REVIEW plant information sources to define or establish (a) system components and boundaries (b) dependencies on other systems (c) instrumentation and control requirements (d) testing and maintenance requirements and practices (e) operating limitations such as those imposed by Technical Specifications (f) component operability and design limits (g) procedures for the operation of the system during normal and accident conditions (h) system configuration during normal and accident conditions	No	NAPS PRA SY notebooks generally meets this SR, however, the notebooks do not contain documentation on operating limitations, operating procedures (both normal and accident conditions) and system configuration.	The comments here reflect enhancements, if anything. The SR is basically met. No impact on the 1J1 LAR.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SY-A4	PERFORM plant walkdowns and interviews with system engineers and plant operators to confirm that the systems analysis correctly reflects the as-built, as-operated plant.	No	The NAPS PRA generally meets the requirements of Capability Category II/III of this SR based on IPE input sources include plant walkdowns or directly from the System Engineer. However, no evidence was found to support this claim.	The comments here reflect enhancements, if anything. The SR is basically met. No impact on the 1J1 LAR.
SY-A11	INCORPORATE the effect of variable success criteria (i.e., success criteria that change as a function of plant status) into the system modeling. Example causes of variable system success criteria are: (a) different accident scenarios. Different success criteria are required for some systems to mitigate different accident scenarios (e.g., the number of pumps required to operate in some systems is dependent upon the modeled initiating event); (b) dependence on other components. Success criteria for some systems are also dependent on the success of another component in the system (e.g., operation of additional pumps in some cooling water systems is required if noncritical loads are not isolated); (c) time dependence. Success criteria for some systems are time-dependent (e.g., two pumps are required to provide the needed flow early following an accident initiator, but only one is required for mitigation later following the accident); (d) sharing of a system between units. Success criteria may be affected when both units are challenged by the same initiating event (e.g., LOOP).	No	There are two issues the NAPS PRA model does not meet for this SR. 1) Modeling of inadvertent SI actuation. NAPS should create an event tree modeling the 6/30/07 event. 2) Dual unit initiating events. Unit 2 equipment may not be available during a dual unit event	This SR has a number of requirements (see Column B). Based on the knowledge of the model the majority (if not all) requirements are met. As far as this one time extension of the 1J1 CT is concerned, the two potential limitations stated here would not have a significant impact on the results on the basis that 1) in the inadvertent SI case, the valves which are supported by the 1J1 MCC will be in the required accident mitigating configuration and can be manually operated (e.g., AFW MOVs) and/or have redundant components that are supported by the H bus and are verified operable.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SY-A13	When identifying the failures in SY-A12 INCLUDE consideration of all failure modes, consistent with available data and model level of detail, except where excluded using the criteria in SY-A14. For example: (a) active component fails to start (b) active component fails to continue to run (c) failure of a closed component to open (d) failure of a closed component to remain closed (e) failure of an open component to close (f) failure of an open component to remain open (g) active component spurious operation (h) plugging of an active or passive component (i) leakage of an active or passive component (j) rupture of an active or passive component (k) internal leakage of a component (l) internal rupture of a component (m) failure to provide signal/operate (e.g., instrumentation) (n) spurious signal/operation (o) pre-initiator human failure events (see SY-A15) (p) other failures of a component to perform its required function	No	The majority of the failure modes listed in this SR is included in the NAPS system models. However, it is not clear if there is a distinction made between leakage (i), rupture (j), internal leakage (k), and internal rupture (l). In addition, (n) spurious signals, is generally met (e.g., the AOV-SC fault includes the fault of FT-116A/B/C spurious actuations); however, the spurious SI actuation event on 6/27/07 is not modeled.	The comments here reflect enhancements, if anything. The SR is basically met. No impact on the 1J1 LAR.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SY-A17	INCLUDE in either the system model or accident sequence modeling those conditions that cause the system to isolate or trip, or those conditions that once exceeded cause the system to fail, or SHOW that their exclusion does not impact the results. For example, conditions that isolate or trip a system include: (a) system-related parameters such as a high temperature within the system (b) external parameters used to protect the system from other failures [e.g., the high reactor pressure vessel (RPV) water level isolation signal used to prevent water intrusion into the turbines of the RCIC and HPCI pumps of a BWR] (c) adverse environmental conditions (see SY-A20)	No	Several different conditions that can result system isolation or failure are included in the system models. Examples include pump seal or lube oil cooling water to prevent trip on high temperature; availability of a pump miniflow path to prevent dead heading a pump; ventilation for rooms containing electrical equipment, EDGs, and certain pumps; and shedding of major loads to prevent potential EDG failure. Assumptions regarding whether a particular condition will result in system failure are documented in Table 1 of Volume SY.2. However, load sequencing for the diesels is not included in the model, nor are there any assumptions or referenced calculations pertaining to load sequencing. Load sequencing of the emergency buses is required to prevent overloading of the emergency diesel generator.	The comments here reflect enhancements, if anything. The SR is basically met. No impact on the 1J1 LAR.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SY-A19	IDENTIFY system conditions that cause a loss of desired system function, e.g., excessive heat loads, excessive electrical loads, excessive humidity, etc.	No	Electrical load shedding and sequencing does not appear to be modeled for the Emergency Diesel Generators. Without shedding loads and sequencing the loads back on the electrical bus, the emergency diesel generator will be overloaded, leading to EDG failure.	Based on the knowledge of the NAPS model the requirements for this SR are met (which is basically confirmed by the GAP reviewers, given that only one potential concern is provided). The reviewers have identified one potential issue (sequencing of the EDGs) and have repeated this issue to down grade a number of SRs, although the preponderance of evidence point to meeting the applicable SR at Category II level. Also, from the 1J1 MCC point of view, the potential impact on the J EDG's failure probability is not that consequential and the potential increase in the 1H EDG's failure probability, given all the other contributors (e.g., fail to start, fail to run), is not significant. Therefore, it is concluded that 1) this SR is basically met and 2) the one example of potential concern which is provided here, if true, would not change the delta risk estimate for this application.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SY-A20	TAKE CREDIT for system or component operability only if an analysis exists to demonstrate that rated or design capabilities are not exceeded.	No	There are several instances where it is assumed that ventilation is not needed for equipment like the CCW pumps, AFW pumps, and Low head Safety Injection pump rooms. These assumptions need to be verified with an engineering analysis. (Ref. table 1 SY.1)	The NAPS IPE PRA model includes an evaluation of room cooling dependencies over the range of accident scenarios (Reference Section 2.3.4 of the NAPS IPE). The room cooling dependency for each system is documented in Appendix A of the IPE. The NAPS PRA is in the process of being updated to address self-assessment findings. To address this comment, in particularly, preliminary heat-up calculations were performed for plant areas were ventilation was not credited in the PRA. These heat-up engineering assessments have not been verified by the PRA group but, conservatively, it is decided to add additional room HVAC dependencies into the forthcoming model. It is noted that there are only a few instances that all the components supporting both trains of the same system are not supported by the same ventilation system. For example, the two motor driven AFWs are supported by the same ventilation system and the Turbine Driven AFW pump is supported by another one. Since the 1J1 MCC unavailability has a minimal impact on the AFW function, this potential ventilation dependency addition to the PRA model is not expected to impact the result. Additionally, as far as the 1J1 LAR is concerned, maintenance on components important to safety will not be allowed and therefore, the failure probability of both train of HVAC will be very small.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SY-B3	ESTABLISH common cause failure groups by using a logical, systematic process that considers similarity in (a) service conditions (b) environment (c) design or manufacturer (d) maintenance JUSTIFY the basis for selecting common cause component groups. Candidates for common cause failures include, for example: (a) motor-operated valves (b) pumps (c) safety-relief valves (d) air-operated valves (e) solenoid-operated valves (f) check valves (g) diesel generators (h) batteries (i) inverters and battery charger (j) circuit breakers	No	Common cause failures are modeled for all of the candidate equipment listed in this SR as well as for transformers, PORVs, screens and filters, and hydraulic valves. The modeled failures are documented in the system and data analysis notebooks. However, documentation needs to be enhanced on justify the basis for selected CC group.	This SR is met. The documentation may have room for improvement but the intent of the SR is basically met.
SY-B6	PERFORM engineering analyses to determine the need for support systems that are plant-specific and reflect the variability in the conditions present during the postulated accidents for which the system is required to function.	No	There are several cases where not modeling of support systems is not adequately documented. In particular, room ventilation for several pumps, including CCW, AFW, etc.	See comment on SY-A20
SY-B7	BASE support system modeling on realistic success criteria and timing, unless a conservative approach can be justified, i.e., if their use does not impact risk significant contributors.	No	See SY-B6	See comment on SY-A20

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SY-B8	IDENTIFY spatial and environmental hazards that may impact multiple systems or redundant components in the same system, and ACCOUNT for them in the system fault tree or the accident sequence evaluation. Example: Use results of plant walkdowns as a source of information regarding spatial/environmental hazards, for resolution of spatial/environmental issues, or evaluation of the impacts of such hazards.	No	The current system notebooks reference separate fire, internal flood, and seismic analysis notebooks for discussion of spatial and environmental hazards. The current system notebooks do not include any discussion of plant walkdowns. Only the internal flooding notebooks are available for review (which is discussed with those SRs for the IF element), however the IF notebooks and model are based on a previous engineering calculation and have not yet been updated to the standard format. The current notebooks also refer to a previous version of NAPS PRA documentation for historical purposes. However, this historical version of the PRA also does not include discussion of spatial and environmental hazards nor walkdowns performed.	This requirement basically intends to ensure that the potential impact of the adverse environment, which may be created by a postulated initiating event, on the components modeled to mitigate the consequences of the initiating event, is addressed. For example, a steam line break outside containment may disable components in the AFW system. Based on the knowledge of the NAPS original model, such environmental impacts were considered during model development. The issue raised here seems to be a documentation issue only. Additionally, from the 1J1 LAR point of view, such potential environmental impact generally affects redundant components. Also, note that planned maintenance activities on risk significant components will not be performed during the proposed 1J1 CT extension. Therefore, the impact on a plant configuration where one redundant component is out of service is negligible. Therefore, there is no impact on 1J1 LAR the risk insights.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SY-B15	IDENTIFY SSCs that may be required to operate in conditions beyond their environmental qualifications. INCLUDE dependent failures of multiple SSCs that result from operation in these adverse conditions. Examples of degraded environments include: (a) LOCA inside containment with failure of containment heat removal (b) safety relief valve operability (small LOCA, drywell spray, severe accident) (for BWRs) (c) steam line breaks outside containment (d) debris that could plug screens/filters (both internal and external to the plant) (e) heating of the water supply (e.g., BWR suppression pool, PWR containment sump) that could affect pump operability (f) loss of NPSH for pumps (g) steam binding of pumps	No	Currently, the NAPS PRA model does not distinguish between PZR PORVs failing to reclose on water relief. See EPRI TR-1011047 'Probability of Safety Valve Failure-to-Reseat Following Steam and Liquid Relief'	Again, based on the familiarity with the original NAPS model, the requirements of this SR were generally addressed. Based on a review of "Assessment Comment" only one example is given where the model may not be adequate. Therefore, it is concluded that the intent of the SR is basically met and there is no impact on the 1J1 LAR.
SY-C1	DOCUMENT the systems analysis in a manner that facilitates PRA applications, upgrades, and peer review.	No	The SY.3 Notebooks do not document all the pertinent information from SY-A2 or SY-A3	Documentation only

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SY-C2	DOCUMENT the system functions and boundary, the associated success criteria, the modeled components and failure modes including human actions, and a description of modeled dependencies including support system and common cause failures, including the inputs, methods, and results. For example, this documentation typically includes: (a) system function and operation under normal and emergency operations (b) system model boundary (c) system schematic illustrating all equipment and components necessary for system operation (d) information and calculations to support equipment operability considerations and assumptions (e) actual operational history indicating any past problems in the system operation (f) system success criteria and relationship to accident sequence models (g) human actions necessary for operation of system (h) reference to system-related test and maintenance procedures (i) system dependencies and shared component interface (j) component spatial information (k) assumptions or simplifications made in development of the system models (l) the components and failure modes included in the model and justification for any exclusion of components and failure modes (m) a description of the modularization process (if used) (n) records of resolution of logic loops developed during fault tree linking (if used) (o) results of the system model evaluations (p) results of sensitivity studies (if used) (q) the sources of the above information (e.g., completed checklist from walkdowns, notes from	No	Several of the documentation requirements are not met. These include, (d) lack of room heatup calculations, (e) documentation of actual operating history, (o) no results of system model evaluations, (p) no sensitivity studies, and (q) walkdown documentation	Documentation only

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
	discussions with plant personnel) (r) basic events in the system fault trees so that they are traceable to modules and to cutsets. (s) the nomenclature used in the system models.			

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
SY-C3	DOCUMENT the key assumptions and key sources uncertainty associated with the systems analysis.	No	Assumptions have been identified and documented in SY.2 table 1; however, these assumptions have not been categorized as "Key" and "non-key". In addition, the sources of uncertainty, both key and non-key, have not been identified. Note, Reg Guide 1.200 requires ALL assumptions and ALL uncertainties be identified.	Documentation only
HR-A1	For equipment modeled in the PRA, IDENTIFY, through a review of procedures and practices, those test and maintenance activities that require realignment of equipment outside its normal operational or standby status.	No	NAPS PRA Model Notebook HR.1, "Pre-initiator Human Failure Event Analysis", (Rev. 1, December 2005) does not appear to document or reference an overall review of procedures and practices to identify those test, inspection and maintenance activities that require realignment of equipment outside its normal operational or standby status. Such a procedure review helps to ensure that no human failure events (HFEs) are overlooked that involve failure to restore equipment to the normal operational or standby status.	This seems to be a documentation issue. Also, generally when modeling pre-initiators, the impact is assumed to be common between redundant components. These are also generally low probability events and in the case of 1J1 MCC work, the redundant components will be verified operable (due to Tech Spec concerns as well as risk). Therefore, there is no impact on the 1J1 LAR.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
HR-A2	IDENTIFY, through a review of procedures and practices, those calibration activities that if performed incorrectly can have an adverse impact on the automatic initiation of standby safety equipment.	No	Comments and recommendations are similar to that for HR-A1.	This seems to be a documentation issue. Also, generally when modeling pre-initiators, the impact is assumed to be common between redundant components. These are also generally low probability events and in the case of 1J1 MCC work, the redundant components will be verified operable (due to Tech Spec concerns as well as risk). Therefore, there is no impact on the 1J1 LAR.
HR-A3	IDENTIFY which of those work practices identified above (HR-A1, HR-A2) involve a mechanism that simultaneously affects equipment in either different trains of a redundant system or diverse systems [e.g., use of common calibration equipment by the same crew on the same shift, a maintenance or test activity that requires realignment of an entire system (e.g., SLCS)].	No	Considerations of potential mis-calibration or restoration error events that simultaneously affect equipment in different trains of a redundant system are documented in HR.1 for each pre-initiator HFE. However, the potential for intra-system Type A dependencies is not discussed.	This SR is met. The documentation may have room for improvement but the intent of the SR is basically met.
HR-B1	If screening is performed, ESTABLISH rules for screening individual activities from further consideration. Example: Screen maintenance and test activities from further consideration only if (a) equipment is automatically re-aligned on system demand, or ((b) following maintenance activities, a post-maintenance functional test is performed that reveals misalignment, or (c) equipment position is indicated in the control room, status is routinely checked, and realignment can be affected from the control room, or (d) equipment status is required to be checked frequently (i.e., at least once a shift)	No	No documentation was found for the screening of NAPS maintenance, testing, inspection and calibration activities.	This seems to be a documentation issue. Also, generally, redundant components are affected. In the case of 1J1 MCC work, the redundant components will be verified operable (due to Tech Spec concerns as well as risk). Therefore, there is no impact on the 1J1 LAR.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
HR-B2	DO NOT screen activities that could simultaneously have an impact on multiple trains of a redundant system or diverse systems (HR-A3).	No	No assurance was found in HR.1 that activities that could simultaneously have an impact on multiple trains of a redundant system or diverse systems have been retained.	This seems to be a documentation issue. Also, redundant components are affected. In the case of 1J1 MCC work, the redundant components will be verified operable (due to Tech Spec concerns as well as risk). Therefore, there is no impact on the 1J1 LAR.
HR-C1	For each unscreened activity, DEFINE a human failure event (HFE) that represents the impact of the human failure at the appropriate level, i.e., function, system, train, or component affected.	No	Pre-initiator HFEs currently included in the PRA may not be defined at the appropriate level, i.e., function, system, train, or component affected. For example, restoration errors modeled at the MOV level may not be appropriate, as operators may review the position of MOVs as part of frequent status checks and discover misalignments. (This possibility was noted in of the internal review of HR.1; see comment number 1 in Attachment 2 of HR.1). The more credible restoration error may be the failure to realign manual valves that are used to isolate equipment such as MOVs prior to maintenance or other activity, as the status of manual valves may not be checked frequently (i.e., once per shift or day). Therefore, it may be appropriate to define such events at the train or flow-segment level (rather than at the MOV level in this example), which would include the potential mispositioning of manual valves.	The reviewer comments seem to be based on a modeler's preference for modeling not a real weakness in the model. Also, generally, redundant components are affected. In the case of 1J1 MCC work, the redundant components will be verified operable (due to Tech Spec concerns as well as risk). Therefore, there is no impact on the 1J1 LAR.
HR-C2	INCLUDE those modes of unavailability that, following completion of each unscreened activity, result from failure to restore (a) equipment to the desired standby or operational status (b) initiation signal or set point for equipment start-up or realignment (c) automatic realignment or power ADD failure modes identified during the collection of plant-specific or applicable generic operating experience that leave equipment unavailable for response in accident sequences.	No	Potential failure modes currently considered in the analysis include failure to restore: (a) equipment to the desired standby or operational status, (b) initiation signal or set point for equipment start-up or realignment. However, no documentation was found of considerations of the failure to restore automatic realignment or power. Also, no discussion is provided in HR.1 of a review for such failure modes as part of the collection of plant-specific or applicable generic operating experience.	A number of pre-initiator operator errors are included in the PRA model. Although a rigorous analysis of such events could result in the identification of additional items, pre-initiator operator errors are typically not important to the overall PRA results so it is not expected that resolving the unmet SRs for the pre-initiator HR element with the potential for model changes would alter the findings of this one time AOT extension. Additionally, any change in the risk estimation would impact the base case as well as the 1J1 case. Therefore, the overall risk-insights with respect to the change in risk for this one time extension of the AOT are judged to be unchanged.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
HR-D3	For each detailed human error probability assessment, INCLUDE in the evaluation process the following plant-specific relevant information: (a) the quality of written procedures (for performing tasks) and administrative controls (for independent review) (b) the quality of the human-machine interface, including both the equipment configuration, and instrumentation and control layout	No	The pre-initiator HFE assessment does not appear to provide documentation of the considerations for: (a) the plant-specific quality of written procedures and administrative controls; (b) the plant-specific quality of the human-machine interface. The HRA methodology suggests considerations have been made regarding the quality of administrative controls, but no documentation of these considerations was found.	This seems to be a documentation issue only. Also, as stated above, a number of pre-initiator operator errors are included in the PRA model. Although a rigorous analysis of such events could result in the identification of additional items, pre-initiator operator errors are typically not important to the overall PRA results so it is not expected that resolving the unmet SRs for the pre-initiator HR element with the potential for model changes would alter the findings of this one time AOT extension. Additionally, any change in the risk estimation would impact the base case as well as the 1J1 case. Therefore, the overall risk-insights with respect to the change in risk for this one time extension of the AOT are judged to be unchanged.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
HR-D4	When taking into account self-recovery or recovery from other crew members in estimating HEPs for specific HFEs, USE pre-initiator recovery factors consistent with selected methodology. If recovery of pre-initiator errors is credited (a) ESTABLISH the maximum credit that can be given for multiple recovery opportunities (b) USE the following information to assess the potential for recovery of pre-initiator: (1) post-maintenance or post-calibration tests required and performed by procedure (2) independent verification, using a written check-off list, which verify component status following maintenance/testing (3) original performer, using a written check-off list, makes a separate check of component status at a later time (4) work shift or daily checks of component status, using a written check-off list	No	The HRA evaluation appropriately follows the ASEP methodology and documents the failure event, assumptions, associated procedures, analysis, and results. The evaluation appropriately considers whether there is a compelling status indication, effective post-activity test, independent verification, and/or frequent status check. The existence of these features (yes/no) determines the applicable ASEP case for estimating the HEP. Regarding the specific elements of this SR, the pre-initiator HRA follows the ASEP methodology and thus credit for multiple recovery opportunities is appropriate. A minor recommendation is made to ensure this SR is fully met.	A stated under the "Assessment Comments" for this SR, the SR is met. The comment is a "minor" enhancement.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
HR-E2	IDENTIFY (a) those actions required to initiate (for those systems not automatically initiated), operate, control, isolate, or terminate those systems and components used in preventing or mitigating core damage as defined by the success criteria (e.g., operator initiates RHR) (b) those actions performed by the control room staff either in response to procedural direction or as skill-of-the-craft to recover a failed function, system or component that is used in the performance of a response action as identified in HR-H1.	No	As documented in HR.2 and the supporting documentation, the NAPS post-initiator HRA appears to satisfy this SR, as well as the RG 1.200 clarifications to this SR (the HRA quantification includes the need for diagnosis in identifying failure(s)). However, no documentation is provided of the specific considerations made to include post-initiator human actions in the model. Such documentation would facilitate PRA applications, upgrades, and peer review.	A stated under the "Assessment Comments" for this SR, the SR is met. .
HR-G1	PERFORM detailed analyses for the estimation of HEPs for significant HFEs. USE screening values for HEPs for non-significant human failure basic events.	No	As documented in HR.2, detailed analyses (HCR and THERP) have been performed for the estimation of all post-initiator HEPs. However, consideration for the appropriateness of the HCR method is not documented. As noted in the Probabilistic Risk Assessment Manual, Part II, Chapter E, Section 2, "Human Error Probability Assessment", if the time available for response is very long relative to the estimated time to formulate the response, the HCR method may yield a low probability of failure and other processes that were not captured in the correlation may become important. Also, for execution errors, the impact of stress is not documented.	The comment here is seems to be mainly, if anything, a documentation issue. The preponderance of evidence point to meeting the applicable SR at Category II level.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
HR-G3	When estimating HEPs EVALUATE the impact of the following plant-specific and scenario-specific performance shaping factors: (a) quality [type (classroom or simulator) and frequency] of the operator training or experience (b) quality of the written procedures and administrative controls (c) availability of instrumentation needed to take corrective actions (d) degree of clarity of the cues/indications (e) human-machine interface (f) time available and time required to complete the response (g) complexity of the required response (h) environment (e.g., lighting, heat, radiation) under which the operator is working (i) accessibility of the equipment requiring manipulation (j) necessity, adequacy, and availability of special tools, parts, clothing, etc.	No	Some consideration of performance shaping factors is documented in HR.2. However, thorough documentation of the impact of plant-specific and scenario-specific performance-shaping factors (PSFs) as outlined by this SR is not provided.	This seems to be a documentation issue only. As stated in "Assessment Comments" column for HR-G1, as documented in HR.2, detailed analyses (HCR and THERP) have been performed for the estimation of all post-initiator HEPs. Therefore, the HEP values are not expected to change significantly due to the potential concerns about this SR.
HR-G4	BASE the time available to complete actions on appropriate realistic generic thermal-hydraulic analyses, or simulation from similar plants (e.g., plant of similar design and operation). SPECIFY the point in time at which operators are expected to receive relevant indications.	No	Time windows for successful completion of actions in some instances may need to be updated (for example, those that are based on estimates made for the IPE) or in some instances documented more fully (provide a reference, for example, for the engineering estimate of how long the charging pumps can operate without service water flow).	This is an enhancement. No impact on the 1J1 LAR.
HR-G5	When needed, BASE the required time to complete actions for significant HFEs on action time measurements in either walkthroughs or talk-throughs of the procedures or simulator observations.	No	No documentation was found of the need for time measurements to confirm the required time to complete actions for significant HFEs.	Although a rigorous evaluation may result in some changes, it is not expected that the HEP values or the overall risk estimates will be changed significantly. Therefore, it is concluded that the issue does not have a significant impact on the 1J1 LAR.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
HR-G6	CHECK the consistency of the post-initiator HEP quantifications. REVIEW the HFEs and their final HEPs relative to each other to check their reasonableness given the scenario context, plant history, procedures, operational practices, and experience.	No	The PRA notebooks and supporting material do not appear document a review of the HFEs and their final HEPs relative to each other to check reasonableness given the scenario context, plant history, procedures, operational practices, and experience.	Although a rigorous evaluation may result in some changes, it is not expected that the HEP values or the overall risk estimates will increase significantly. Additionally, any change on the risk estimation would impact the base case as well as the 1J1 case. Therefore, the overall risk-insights with respect to the change in risk for this one time extension of the AOT are judged to remain unchanged.
HR-G7	For multiple human actions in the same accident sequence or cut set, identified in accordance with supporting requirement QU-C1, ASSESS the degree of dependence, and calculate a joint human error probability that reflects the dependence. ACCOUNT for the influence of success or failure in preceding human actions and system performance on the human event under consideration including (a) the time required to complete all actions in relation to the time available to perform the actions (b) factors that could lead to dependence (e.g., common instrumentation, common procedures, increased stress, etc.) (c) availability of resources (e.g., personnel) [Note (1)]	No	The NAPS PRA Model Notebook HR.3, "Recovery Action Analysis," (Revision 0, December 2005) appropriately evaluates the human error probability dependencies between post-initiator events modeled in the NAPS PRA. A recommendation is made to demonstrate that the population of cutsets included in the review (appears to be the top 100, with HEPs set to 0.1) is sufficient to capture the dependencies that can impact the risk profile of the PRA. Also, see HR-H3 for recommendations regarding the consideration of potential dependencies between post-initiator and recovery HFEs.	This SR is basically met. The comment in the "Assessment Comments" column is a suggestion for potential enhancement.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
HR-H2	CREDIT operator recovery actions only if, on a plant-specific basis: (a) a procedure is available and operator training has included the action as part of crew's training, or justification for the omission for one or both is provided (b) "cues" (e.g., alarms) that alert the operator to the recovery action provided procedure, training, or skill of the craft exist (c) attention is given to the relevant performance shaping factors provided in HR-G3 (d) there is sufficient manpower to perform the action	No	1) Operator recovery actions modeled in the PRA are credited in accordance with this SR, however it is recommended that documentation be included to demonstrate sufficient manpower availability to perform the actions and complete consideration of performance shaping factors. 2) Also WOG Peer Review F&O HR-02 Sub-element 13 indicates that basic events are deleted from the model using arguments that they would be easy to recover. The F&O resolution doesn't appear to resolve this comment.	This SR is basically met. The comment in the "Assessment Comments" column is a suggestion for potential enhancement.
HR-H3	ACCOUNT for any dependency between the HFE for recovery and any other HFEs in the sequence, scenario, or cutset to which the recovery is applied.	No	QU.2 Attachment 7 provides consideration of dependencies between recovery HFEs. However, no documentation was found regarding potential dependencies between the HFEs for recovery and any other HFEs in the sequence, scenario, or cutset to which the recovery is applied.	This SR is basically met. The comment in the "Assessment Comments" column is a suggestion for potential enhancement. Also, although a rigorous evaluation may result in some changes, it is not expected that the HEP values or the overall risk estimates will increase significantly. Additionally, any change on the risk estimation would impact the base case as well as the 1J1 case. Therefore, the overall risk-insights with respect to the change in risk for this one time extension of the AOT are judged to remain unchanged.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
HR-I2	DOCUMENT the process used to identify, characterize and quantify the pre-initiator, post-initiator and recovery actions considered in the PRA, including the inputs, methods, and results. For example, this documentation typically includes: (a) HRA methodology and process used to identify pre- and post-initiator HEPs (b) qualitative screening rules and results of screening (c) factors used in the quantification of the human action, how they were derived (their bases), and how they were incorporated into the quantification process (d) quantification of HEPs, including: (1) screening values and their bases (2) detailed HEP analyses with uncertainties and their bases (3) the method and treatment of dependencies for post-initiator actions (4) tables of pre- and post-initiator human actions evaluated by model, system, initiating event, and function (5) HEPs for recovery actions and their dependency with other HEPs	No	The HRA documentation complies with this SR, except as noted by the assessment comments and recommendations of other HR supporting requirements and specific recommendations cited below.	This SR is basically met. See comments above.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
HR-I3	DOCUMENT the key assumptions and key sources uncertainty associated with the human reliability analysis.	No	While assumptions and sources of uncertainty are documented to some degree in the HRA notebooks, a systematic review of uncertainty and assumptions that can impact the risk profile of the base PRA is not documented or referenced in the initiating events notebooks.	The issue indentified here seems to be an enhancement. Also, although a rigorous uncertainty identification and documentation may provide better insights when using the model for permanent changes, it is judged that the impact on this one-time extension is negligible.
DA-B2	DO NOT INCLUDE outliers in the definition of a group (e.g., do not group valves that are never tested and unlikely to be operated with those that are tested or otherwise manipulated frequently)	No	While the grouping appears to be appropriate, there is no discussion in DA.2 or other data notebooks describing the specific process used and how outliers (if they exist) are treated.	This seems to be a documentation issue, if anything. No impact on this one time extension.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
DA-C1	OBTAIN generic parameter estimates from recognized sources. ENSURE that the parameter definitions and boundary conditions are consistent with those established in response to DA-A1 to DA-A3. [Example: some sources include the breaker within the pump boundary, whereas others do not.] DO NOT INCLUDE generic data for unavailability due to test, maintenance, and repair unless it can be established that the data is consistent with the test and maintenance philosophies for the subject plant. Examples of parameter estimates and associated sources include: (a) component failure rates and probabilities: NUREG/CR-4639, NUREG/CR-4550 (b) common cause failures: NUREG/CR-5497, NUREG/CR-6268 (c) AC off-site power recovery: NUREG/CR-5496, NUREG/CR-5032 (d) component recovery	No	In general, the generic data used in the PRA meets the requirements of this SR. However, for loss of offsite power-related events, the data developed in Dominion generic notebook IE.2 is based on most recent NRC and industry data. However, there are various other OSP-related special events that are documented in notebook DA.4 that still appear to be based on NUREG-1150. These special events include some level 2 related power events as well as power recovery events. DA.4 should be reviewed to confirm that the NUREG-1150 events are still applicable, and if so, the notebook should explain how these events remain applicable.	Based on a review of the comments in "Assessment Comments column, it is concluded that the SR is basically met. The reviewers have identified a couple of data points that may need to be updated but they found that "in general, the generic data used in the PRA meets the requirements of this SR". The implementation of the suggested updated is not expected to result in significant change in the delta risk calculations.
DA-C3	COLLECT plant-specific data, consistent with uniformity in design, operational practices, and experience. JUSTIFY the rationale for screening or disregarding plant-specific data (e.g., plant design modifications, changes in operating practices).	No	Only recent data is being used in notebook DA.2 (last 3 years, 2000 to 2004) and DA.6 (2002 through 2004). There is no discussion of the rationale for excluding data from prior periods. As a minimum, data from all of the years for which Maintenance Rule data is available should be considered, since this data is generally of adequate quality for use in failure rate estimation.	Based on experience with dealing with similar comments, it is judged that implementation of the issue raised for this SR, would not significantly impact the overall or the case specific risk estimates. Note that, for this one time extension, the operability of the redundant components is expected to be verified (i.e., the failure probability of the redundant components will be lower than used in the base PRA model).

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
DA-C6	DETERMINE the number of plant-specific demands on standby components on the basis of the number of (a) surveillance tests (b) maintenance acts (c) surveillance tests or maintenance on other components (d) operational demands. DO NOT COUNT additional demands from post-maintenance testing; that is part of the successful renewal.	No	Notebook DA.2 references the data collection guidelines of NUREG/CR-6823, which addresses not counting PMT demands. The notebook presents plant-specific numbers of demands for various components, which are documented in the spreadsheets that are attached to this notebook. However, there is no documentation of the methodology used to obtain these estimates (e.g., from actual operational data, estimated from surveillance test frequency, etc.)	This appears to be a documentation issue only. Again, note that for this one time extension, the redundant components are expected to be verified operable.
DA-C7	BASE number of surveillance tests on plant surveillance requirements and actual practice. BASE number of planned maintenance activities on plant maintenance plans and actual practice. BASE number of unplanned maintenance acts on actual plant experience.	No	A portion of the requirements are met by notebooks DA.2 and DA.6. As noted in DA-C6, insufficient documentation is provided to describe how the number of operational demands was obtained. For maintenance unavailability, these estimates are obtained directly from Maintenance Rule information, thereby reflecting actual plant operating experience.	This appears to be a documentation issue only. Again, note that for this one time extension, the redundant components are expected to be verified operable.
DA-C8	When required, USE plant-specific operational records to determine the time that components were configured in their standby status.	No	Alignment fractions exist in the PRA models, but are not discussed in the documentation. The alignment fractions are based on assumed data (e.g., 33% run time in a three pump system) instead of using actual alignment fractions from operating experience. Therefore, while Capability Category I is met and the assumed fractions are reasonable, this is not sufficient to support Capability Category II as it does not appear that plant records were reviewed/considered. Also, BED file includes various MULT events that are not documented anywhere.	Based on experience with dealing with similar comments, it is judged that implementation of the issue raised for this SR, would not significantly impact the overall or the case specific risk estimates.
DA-C9	ESTIMATE operational time from surveillance test practices for standby components, and from actual operational data.	No	As noted in DA-C6, there is insufficient documentation in the DA.2 notebook to describe how operational time was determined. The plant-specific data is presented in the spreadsheets but without documentation of how the operational time was derived.	Similar to the other issues raised for DA, the SR's requirements are there to ensure that a realistic failure probability is calculated. For this one time extension, for the 1J1 case, the operability of the redundant components is expected to be verified. Therefore, because the calculation does not credit this operability-verification aspect, the delta risk calculation is judged to be conservative.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
DA-C10	When using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. COUNT only completed tests or unplanned operational demands as success for component operation. If the component failure mode is decomposed into sub-elements (or causes) that are fully tested, then USE tests that exercise specific sub-elements in their evaluation. Thus, one sub-element sometimes has many more successes than another. [Example: a diesel generator is tested more frequently than the load sequencer. IF the sequencer were to be included in the diesel generator boundary, the number of valid tests would be significantly decreased.]	No	There is no evidence in notebook DA.2 to indicate that a review of the surveillance procedures was performed.	Based on experience with dealing with similar comments, it is judged that implementation of the issue raised for this SR, would not significantly impact the overall or the case specific risk estimates. Note that, for this one time extension, the operability of the redundant components is expected to be verified (i.e., the failure probability of the redundant components will be lower than used in the base PRA model).
DA-C11a	When an unavailability of a front line system component is caused by an unavailability of a support system, COUNT the unavailability towards that of the support system and not the front line system, in order to avoid double counting and to capture the support system dependency properly.	No	Yes, guidelines of NUREG/CR-6823 are used, as noted in notebook DA.2, which is consistent with the requirements of this SR. It appears that the DAS.6 notebook also properly assigns unavailability to the appropriate support system, but the documentation of this could be improved.	This SR is met.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
DA-C12	EVALUATE the duration of the actual time that the equipment was unavailable for each contributing activity. Since maintenance outages are a function of the plant status, INCLUDE only outages occurring during plant at power. Special attention should be paid to the case of a multi-plant site with shared systems, when the Specifications (TS) requirements can be different depending on the status of both plants. Accurate modeling generally leads to a particular allocation of outage data among basic events to take this mode dependence into account. In the case that reliable estimates for the start and finish times are not available, INTERVIEW the plant maintenance and operations staff to generate estimates of ranges in the unavailable time per maintenance act for components, trains, or systems for which the unavailabilities are significant basic events.	No	Unavailability data is based solely on plant-specific data and is documented in the DA.6 notebook. However, it is not clear if both units' data is being used to compute unavailability. For components with no observed unavailability, a floor value of 1e-6 is used. It is appropriate to assume a minimal value for UA; however, the floor value chosen may be too small (~30 seconds per year). Also as North Anna has a number of shared systems, the documentation should discuss how multiple unit impacts (considering Tech Spec requirements, etc.) has been accounted for.	Although unavailability data is based on plant-specific data and is documented in the appropriate notebook, the SR is characterized as unmet because it was not clear to the reviewers that both units' data was being used to compute unavailability. Also, the reviewers concluded that using a floor value of 1.0E-6 is too low. This particular issue does not have any impact on this one time AOT extension since maintenance activities on all other risk significant components is prohibited.
DA-C15	Data on recovery from loss of offsite power, loss of service water, etc. are rare on a plant-specific basis. If available, for each recovery, COLLECT the associated recovery time with the recovery time being the period from identification of the system or function failure until the system or function is returned to service.	No	The DOM IE.2 notebook presents OSP frequencies and recovery data for all Dominion plants. OSP Recovery is calculated in IE.2, but is not discussed (only presented in a spreadsheet). However, some PRA events in the database (see notebook DA.4) indicate that they are based on older NUREG-1150 power recovery and LOSP frequency numbers (e.g., PROB-OSP-8-1P5-S and similar events). It appears that the correct (current) OSP information is being used but the documentation in DA.4 is out of date	Documentation only. No impact.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
DA-D2	If neither plant-specific data nor generic parameter estimates are available for the parameter associated with a specific basic event, USE data or estimates for the most similar equipment available, adjusting if necessary to account for differences. Alternatively, USE expert judgment and document the rationale behind the choice of parameter values.	No	Generic or plant-spec data appears to be used for most component events. However, the special BE notebook (DA.4) does include various items based on expert judgment. In this notebook, the degree of explanation of that judgment is not very detailed and should be expanded.	Documentation issue. No impact. Also, for this one time extension, the failure probability of redundant components does not pay as bigger role, since these components will be verified operable.
DA-D3	PROVIDE a mean value of, and a statistical representation of the uncertainty intervals for, the parameter estimates of significant basic events. Acceptable systematic methods include Bayesian updating, frequentist method, or expert judgment.	No	Uncertainty parameters are not provided for CCF and maintenance unavailability events. In the BED file, the T&M terms appear to have a 3.0 error factor (but this is not documented).	Although a rigorous uncertainty identification and documentation may provide better insights when using the model for permanent changes, it is judged that the impact on this one-time extension is negligible.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
DA-D4	When the Bayesian approach is used to derive a distribution and mean value of a parameter, CHECK that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data. Examples of tests to ensure that the updating is accomplished correctly and that the generic parameter estimates are consistent with the plant-specific application include the following: (a) confirmation that the Bayesian updating does not produce a posterior distribution with a single bin histogram (b) examination of the cause of any unusual (e.g., multimodal) posterior distribution shapes (c) examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate (d) confirmation that the Bayesian updating algorithm provides meaningful results over the range of values being considered (e) confirmation of the reasonableness of the posterior distribution mean value	No	ReDAT plots are included in notebook DA.2 (can visually determine that distribution is reasonable). Notebook DA.2 also presents percentage deviations from prior for all events. Some generic explanations for the observed differences are provided, but specific discussion of dominant items could be improved.	This SR requirement is basically met. The issue seems to be documentation enhancement only.
DA-D6	USE generic common cause failure probabilities consistent with available plant experience. EVALUATE the common cause failure probabilities consistent with the component boundaries.	No	Generic common cause failure factors are used. However, it is not clear from notebook DA.3 documentation if any review of these factors against plant experience was performed. Also, discussion of how the CCF boundaries were compared to the component boundary definitions was not provided.	This seems to be an enhancement. Also, from this 1J1 LAR point of view, for this one time extension request, the reliability of the redundant components are more important the CCF probabilities. CCF of redundant components is generally expected to result in reduction in the delta risk associated with the 1J1 unavailability

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
DA-D6a	If screening of generic event data is performed for plant-specific estimation, PERFORM screening on both the CCF events and the independent failure events in the data base used to generate the CCF parameters.	No	It is not clear from the DOM DA.3 or NAPS DA.3 notebooks if screening was performed. As a minimum, these notebooks should clearly state if performed or not performed. If performed, a description of how the screening was performed for both independent and CCF events should be provided.	See comment on DA-D6
DA-D7	If modifications to plant design or operating practice lead to a condition where past data are no longer representative of current performance, LIMIT the use of old data: (a) If the modification involves new equipment or a practice where generic parameter estimates are available, USE the generic parameter estimates updated with plant-specific data as it becomes available for significant basic events; or (b) If the modification is unique to the extent that generic parameter estimates are not available and only limited experience is available following the change, then ANALYZE the impact of the change and assess the hypothetical effect on the historical data to determine to what extent the data can be used.	No	Notebook DA.2 does not include any discussion concerning whether or not any plant changes resulted in any data exclusion. It does not appear that any data was excluded, but the notebook should definitively state whether or not any exclusions were needed. The spreadsheets that are attached to the notebook show if specific failure events were discarded (e.g., if the "failure" did not disable the function of the component), but the spreadsheet does not indicate if plant changes resulted in the reason for any data being discarded.	This appears to be a documentation issue only. Again, note that for this one time extension, the redundant components are expected to be verified operable.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
DA-E2	DOCUMENT the processes used for data parameter definition, grouping, and collection including parameter selection and estimation, including the inputs, methods, and results. For example, this documentation typically includes: (a) system and component boundaries used to establish component failure probabilities (b) the model used to evaluate each basic event probability (c) sources for generic parameter estimates (d) the plant-specific sources of data (e) the time periods for which plant-specific data were gathered (f) justification for exclusion of any data (g) the basis for the estimates of common cause failure probabilities, including justification for screening or mapping of generic and plant-specific data (h) the rationale for any distributions used as priors for Bayesian updates, where applicable (i) parameter estimate including the characterization of uncertainty, as appropriate	No	Many of the requirements of this SR are met. However, Documentation is missing for various items noted in other SRs, including T&M and CCF uncertainty parameters, alignment fractions, and various MULT events that appear in the model.	This SR is basically met. The potential issues identified here are minor and are not expected to impact the delta risk calculations.
DA-E3	DOCUMENT the key assumptions and key sources of uncertainty associated with the data analysis.	No	Assumptions are listed in the various DA notebooks (DA.1 through DA.6). However, sources of uncertainties associated with the data development process are not identified or discussed.	Although a rigorous uncertainty identification and documentation may provide better insights when using the model for permanent changes, it is judged that the impact on this one-time extension is negligible.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IF-A4	CONDUCT a plant walkdown to verify the accuracy of information obtained from plant information sources and to obtain or verify (a) spatial information needed for the development of flood areas, and (b) plant design features credited in defining flood areas Note: Walkdown(s) may be done in conjunction with the requirements of IF-B3a, IF-C9 and IF-E8.	No	Draft Volume IF.1 refers frequently to walkdowns performed in 2001 and discusses findings from this walkdown. There is no other record of this walkdown, however.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
IF-B2	For each potential source of flooding water, IDENTIFY the flooding mechanisms that would result in a fluid releaser. INCLUDE: (a) failure modes of components such as pipes, tanks, gaskets, expansion joints, fittings, seals, etc. (b) human-induced mechanisms that could lead to overfilling tanks, diversion of flow through openings created to perform maintenance; inadvertent actuation of fire suppression system (c) other events releasing water into the area	No	Section 2.0 of draft Volume IF.1 states that "In practice, major internal floods have occurred in nuclear power plants, for example, from the rupture of pipes, valves and expansion joints as well as from operator errors during plant maintenance activities. All potential internal flood sources and causes are considered in this analysis." However, Section 5.2 of draft Volume IF.1 limits the applicable flooding mechanisms almost exclusively to pipe breaks and tank ruptures. Human-induced mechanisms are not included in the discussion.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IF-B3	For each source and its identified failure mechanism, IDENTIFY the characteristic of release and the capacity of the source. INCLUDE: (a) a characterization of the breach, including type (e.g., leak, rupture, spray) (b) flow rate (c) capacity of source (e.g., gallons of water) (d) the pressure and temperature of the source	No	Section 2.0 of draft Volume IF.1 states "Internal flooding analysis encompasses the effects from the accumulation, spraying or dripping of fluids arising from the rupture, cracking or incorrect operation of components within the station. In practice, major internal floods have occurred in nuclear power plants, for example, from the rupture of pipes, valves and expansion joints as well as from operator errors during plant maintenance activities. All potential internal flood sources and causes are considered in this analysis..." In general, Section 5.2 of draft Volume IF.1 identifies the flow rate of flooding event of interest as well as the capacity of the source. Given the flow rates of the pipe breaks reported, pipe ruptures are typically what is being assessed. The dripping and spraying of fluids is not included in the discussion. The pressure and temperature of the fluid source also are excluded from the discussion.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
IF-B3a	CONDUCT a plant walkdown(s) to verify the accuracy of information obtained from plant information sources and to determine or verify the location of flood sources and in-leakage pathways. Note: A walkdown(s) may be done in conjunction with the requirements of IF-A4, IF-C9, and IF-E8.	No	Draft Volume IF.1 refers frequently to walkdowns performed in 2001 and discusses findings from this walkdown. There is no other record of this walkdown, however.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
IF-C1	For each defined flood area and each flood source, IDENTIFY the propagation path from the flood source area to its area of accumulation.	No	Section 5.3 of draft Volume IF.1 discusses possible propagation paths for those flood areas that survived the initial screening process, however there is little detail included in the discussion. Information regarding doors that could fail need to be included as well as the amount of fluid that can pass under the door(s) while closed. This information can impact the scenario development.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IF-C2a	For each defined flood area and each flood source, IDENTIFY those automatic or operator responses that have the ability to terminate or contain the flood propagation.	No	Section 5.1 of draft Volume IF.1 identifies various sump pumps in the flood areas and implies that they start automatically, but this is never clearly stated. Section 5.5.3.2 describes the operator actions to isolate flooding events. While some flood alarms are identified in the analysis, these alarms are never tied to operator actions.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
IF-C2b	ESTIMATE the capacity of the drains and the amount of water retained by sumps, berms, dikes, and curbs. ACCOUNT for these factors in estimating flood volumes and SSC impacts from flooding.	No	Section 5.3 of draft Volume IF.1 provides some information regarding drainage capacities and retained volumes, however the information is incomplete. Section 5.2 identifies the critical flood volumes used in the analysis to determine equipment that is vulnerable to flooding events.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
IF-C2c	For each flood area not screened out using the requirements under other Internal Flooding supporting (Formerly requirements (e.g., IF-B1b and IF-C5), IDENTIFY the SSCs located in each defined flood area and IF-A2) along flood propagation paths that are modeled in the internal events PRA model as being required to respond to an initiating event or whose failure would challenge normal plant operation, and are susceptible to flood. For each identified SSC, IDENTIFY, for the purpose of determining its susceptibility per IF-C3, its spatial location in the area and any flooding mitigative features (e.g., shielding, flood or spray capability ratings).	No	Section 5.1 of draft Volume IF.1 identifies the major components located in each flood area, however very little spatial information is included. Section 5.2 identifies flood sources and the components in the vicinity, but does not identify the flood areas housing the flood sources. There is not enough information to accurately assess the effects of spray events on plant components.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IF-C3	For the SSCs identified in IF-C2c, IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. INCLUDE failure by submergence and spray in the identification process. EITHER: a) ASSESS qualitatively the impact of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions; OR b) NOTE that these mechanisms are not included in the scope of the evaluation.	No	Section 5.5.3.2 of draft Volume IF.1 lists the critical flood heights of various components as part of the development of flooding HEPs. However, it is unclear how these critical flood heights were determined. Typically one can measure the height of a pump motor or some other critical piece of electrical equipment to determine a pump's critical flood height. However, for electrical cabinets this is more difficult since the cabinet walls hide the contents. A conservative approach to electrical cabinets is to simply assume that the cabinet will fail whenever water touches its base. This sometimes leads to unacceptable results, especially if the cabinet is mounted to the floor and does not rest on some type of pad.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
IF-C3a	In applying SR IF-C3 to determine susceptibility of SSCs to flood-induced failure mechanisms, TAKE CREDIT for the operability of SSCs identified in IF-C2c with respect to internal flooding impacts only if supported by an appropriate combination of (a) test or operational data (b) engineering analysis (c) expert judgment	No	It is not clear from the documentation exactly what credit was taken during the actual quantification. The initial screening clearly assumes that all equipment in the flood area of interest is failed. However, for those flood areas that survive the screen and undergo more detailed analysis, it is not as clear. Table 1.2-1 of calculation 327MAF.N provides a list of equipment and the expected response to submergence, spray, and steam, but it is not clear how it was applied to the detailed analysis.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IF-C3b	IDENTIFY inter-area propagation through the normal flow path from one area to another via drain lines; and areas connected via back flow through drain lines involving failed check valves, pipe and cable penetrations (including cable trays), doors, stairwells, hatchways, and HVAC ducts. INCLUDE potential for structural failure (e.g., of doors or walls) due to flooding loads.	No	Section 5.3 of draft Volume IF.1 identifies some potential propagation paths for those flood scenarios undergoing detailed analysis; however, there is no indication that this offers a complete list of potential propagation paths. There is little mention of doors, wall penetrations, or HVAC ducts, all of which typically offer pathways for propagation. All of this information should be available from the walkdowns.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
IF-C3c	PERFORM any necessary engineering calculations for flood rate, time to reach susceptible equipment, and the structural capacity of SSCs in accordance with the applicable requirements described in Table 4.5.3-2(b).	No	Flood rates are readily documented in draft Volume IF.1. Section 5.5.3.2 of draft Volume IF.1 identifies the amount of time required for the water depth to reach critical heights. This information is apparently used in the development of HEPs; however this development is not documented as part of the flooding analysis. Additionally, while Table 1.2-1 of calculation 327MAF.N provides a list of equipment and the expected response to submergence, spray, and steam, there is minimal information that justifies the conclusions reached in this table regarding equipment survivability.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
IF-C4	DEVELOP flood scenarios (i.e., the set of information regarding the flood area, source, flood rate and source capacity, operator actions, and SSC damage that together form the boundary conditions for the interface with the internal events PRA) by examining the equipment and relevant plant features in the flood area and areas in potential propagation paths, giving credit for appropriate flood mitigation systems or operator actions, and identifying susceptible SSCs.	No	Section 5.4 of draft Volume IF.1 discusses flooding scenarios in a roundabout way, but does not meet the spirit of this requirement. The flood area in which the event originates, the source, and the SSC damage is discussed, but the flood rate, source capacity, and operator actions are discussed elsewhere in the report. The report should present each scenario in a near-chronological order describing the movement of fluid and its impacts from the initiation of the flood to the isolation of the event. Flow rates, component failures, times to subsequently reach critical water heights, and necessary operator actions need to be included as part of each scenario.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IF-C4a	For multi-unit sites with shared systems or structures, INCLUDE multi-unit scenarios.	No	See IF-C4.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
IF-C5a	SCREEN OUT flood areas where flooding of the area does not cause an initiating event or a need for immediate plant shutdown, AND the following applies: The flood area contains flooding mitigation systems (e.g., drains or sump pumps) capable of preventing unacceptable flood levels, and the nature of the flood does not cause equipment failure (e.g., through spray, immersion, or other applicable failure mechanisms). DO NOT CREDIT mitigation systems for screening out flood areas unless there is a definitive basis for crediting the capability and reliability of the flood mitigation system(s).	No	Section 5.1 of draft Volume IF.1 describes flood areas that were screened because failure of equipment in the area would not cause a reactor trip or a need for immediate plant shutdown and there are no flood sources in the zone. However, it also describes flood areas (Section 5.1.23) where a flooding event would not cause a reactor trip but might cause equipment failure.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
IF-C9	CONDUCT a plant walkdown(s) to verify the accuracy of information obtained from plant information sources and to obtain or verify (a) SSCs located within each defined flood area (b) flood/spray/other applicable mitigative features of the SSCs located within each defined flood area (e.g., drains, shields, etc.) (c) pathways that could lead to transport to the flood area Note: A walkdown(s) may be done in conjunction with the requirements of IF-A4, IF-B3a, and IF-E8.	No	Draft Volume IF.1 refers frequently to walkdowns performed in 2001 and discusses findings from this walkdown. There is no other record of this walkdown, however.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IF-D4	For multi-unit sites with shared systems or structures, INCLUDE multi-unit impacts on SSCs and plant-initiating events caused by internal flood scenario groups.	No	Some sections of draft Volume IF.1 address the issue of multi-unit impacts of flooding. For example, Section 5.1.7 addresses both units' power conversion equipment being housed in the Turbine Building and Section 5.1.13 addresses both units' charging, CCW, and SI pumps being housed in the Auxiliary Building basement. However, it is not clear how these impacts are addressed in the scenario development. For Turbine Building floods it is simply stated in Section 5.4.2.1 discusses propagation from the Unit 1 side to the Unit 2 side through "limited pathways" but these pathways are not identified nor is their capability of withstanding a flood addressed. The conclusion is simply that the opposite unit's power conversion equipment is not in jeopardy because the water level will never get high enough to threaten it. I would expect to see at the very least some simple calculation that estimates the resulting water level and the time required to reach that level. Similarly for Auxiliary Building floods it is not clear in Sections 5.4.3 and 5.5 if the scenario analyzed includes failure of seal cooling in one unit or both.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
IF-D5	DETERMINE the flood-initiating event frequency for each flood scenario group by using the applicable requirements in Table 4.5.1-2(c).	No	Flooding initiating event frequencies are calculated using IPE-vintage methodologies. Newer industry standards now exist for such calculations.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact. Also, based on work done for similar comments, it is expected that the updated flooding frequencies will be lower than the previous values.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IF-D5a	GATHER plant-specific information on plant design, operating practices, and conditions that may impact flood likelihood (i.e., material condition of fluid systems, experience with water hammer, and maintenance-induced floods). In determining the flood-initiating event frequencies for flood scenario groups, USE a combination of (a) generic and plant-specific operating experience; (b) pipe, component, and tank rupture failure rates from generic data sources and plant-specific experience; and (c) engineering judgment for consideration of the plant-specific information collected.	No	Attachment E (Tables E.6-1, 2, 3, 4, 5, 6, and 7) list industry flooding events and generic flooding frequencies. However, it is not at all clear how this information is transformed into initiating event frequencies for NAPS. The frequencies are documented in draft Volume IF.1 (Section 5.4), but the methodology used is unclear. A better description of how this information was incorporated in the study needs to be included. Also, this data is quite dated and needs to be updated since nearly 20 years have passed since the IPE was developed.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
IF-D6	INCLUDE consideration of human-induced floods during maintenance through application of generic data.	No	While Tables E.6-1 and E.6-3 list human-induced flood events, it is unclear how these events were applied in the development in flood initiating event frequencies for NAPS.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IF-D7	SCREEN OUT flood scenario groups if (a) the quantitative screening criteria in IE-C4, as applied to the flood scenario groups, are met; OR (b) the internal flood-initiating event affects only components in a single system, AND it can be shown that the product of: the frequency of the flood, and the probability of SSC failure given the flood is two orders of magnitude lower than: the product of the non-flooding frequency for the corresponding initiating events in the PRA, and the random (non-flood-induced) failure probability of the same SSCs that are assumed failed by the flood. If the flood impacts multiple systems, DO NOT screen on this basis.	No	The only quantitative screening criteria used in the current NAPS flooding analysis appears to be listed in Section 2.1 of draft Volume IF.1 as any flood area that contributes less than 1.0E-06/yr to overall CDF. SR IE-C4 requires using 1.0E-07/yr as the screening criteria. The criteria under item (b) for this SR were not addressed.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
IF-E1	For each flood scenario, REVIEW the accident sequences for the associated plant-initiating event group to confirm applicability of the accident sequence model. If appropriate accident sequences do not exist, MODIFY sequences as necessary to account for any unique flood-induced scenarios and/or phenomena in accordance with the applicable requirements described in para. 4.5.2.	No	Section 5.5 of draft Volume IF.1 identifies the plant-initiating event group to which the flooding events are added, but there is insufficient discussion of the flooding scenarios, especially the equipment failed by the flood and the timing of the failures, to assess whether the flood event belongs in that group or if a new group is warranted.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IF-E3a	SCREEN OUT a flood area if the product of the sum of the frequencies of the flood scenarios for the area, and the bounding conditional core damage probability (CCDP) is less than 10-9/reactor year. The bounding CCDP is the highest of the CCDP values for the flood scenarios in an area.	No	The only quantitative screening performed screened flood areas if their contribution to CDF was less than 1.0E-06/yr. The screening criteria identified in this SR are not currently part of NAPS flooding analysis.	This SR does not have an impact on the 1J1 LAR for NAPS because, in the most important flood areas, the redundant components are located in the same area. So, even if some areas were optimistically screened out, the change in risk due to the unavailability of 1J1 MCC is not expected to be significantly affected.
IF-E4	If additional analysis of SSC data is required to support quantification of flood scenarios, PERFORM the analysis in accordance with the applicable requirements described in para. 4.5.6.	No	The current analysis does not explicitly model several components that are unique to internal flooding. Components such as sump pumps and level switches likely have an impact on the development of flooding scenarios, yet such components are currently missing from the model. Such components would require data development. Additionally, structures such as doors can have a considerable impact on water propagation and there is no data analysis on the water heights required to fail such structures. These SSCs need to be explicitly addressed in the flooding analysis.	Based on the familiarity and involvement of this analyst with the original NAPS internal flooding analysis, in the analysis, when flood mitigating features were credited, the appropriate components were modeled. Also, potential door failures were considered. This is evident by the discussion of the analysis as provided in part in section 1.2.9 of NAPS, calculation file 327MAF.N, where a description of probabilistic evaluation of flood induced accident sequence frequencies is provided. This is mostly a documentation issue. Although, form the base model and model maintenance points of view this lack of documentation is not acceptable, for this one time extension request, the impact on the risk insights is considered to be negligible.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IF-E5	If additional human failure events are required to support quantification of flood scenarios, PERFORM any human reliability analysis in accordance with the applicable requirements described in Tables 4.5.5-2(e) trough 4.5.5-2(h).	No	Section 5.5.3 of draft Volume IF.1 discusses operator actions associated with the flooding analysis. First, all HEPs in the internal events analysis were reviewed to determine if they would be impacted by flooding. Next, new operator actions to isolate floods were developed. None of the HEPs (and especially those new operator actions to isolate floods) is discussed in any detail. The text offers no information regarding the cues that inform the operator that a flood is occurring, the procedures in place to mitigate a flood, any training the operator receives that is pertinent to flooding, and the equipment that must be operated (even if it is a remote switch in the control room) to isolate floods.	Again, based on the familiarity with the original NAPS internal flooding model, the requirements of this SR were generally addressed. This is mostly a documentation issue. Although, form the base model and model maintenance points of view this lack of documentation is not acceptable, for this one time extension request, the impact on the risk insights is considered to be negligible.
IF-E5a	For all human failure events in the internal flood scenarios, INCLUDE the following scenario-specific impacts on PSFs for control room and ex-control room actions as appropriate to the HRA methodology being used: (a) additional workload and stress (above that for similar sequences not caused by internal floods) (b) cue availability (c) effect of flood on mitigation, required response, timing, and recovery activities (e.g., accessibility restrictions, possibility of physical harm) (d) flooding-specific job aids and training (e.g., procedures, training exercises)	No	Section 5.5.3 of draft Volume IF.1 discusses operator actions associated with the flooding analysis. First, all HEPs in the internal events analysis were reviewed to determine if they would be impacted by flooding. Next, new operator actions to isolate floods were developed. None of the HEPs (and especially those new operator actions to isolate floods) is discussed in any detail. The text offers no information regarding the cues that inform the operator that a flood is occurring, the procedures in place to mitigate a flood, any training the operator receives that is pertinent to flooding, and the equipment that must be operated (even if it is a remote switch in the control room) to isolate floods.	Again, based on the familiarity with the original NAPS internal flooding model, the requirements of this SR were generally addressed. This is mostly a documentation issue. Although, form the base model and model maintenance points of view this lack of documentation is not acceptable, for this one time extension request, the impact on the risk insights is considered to be negligible.
IF-E6b	INCLUDE, in the quantification, both the direct effects of the flood (e.g., loss of cooling from a service water train due to an associated pipe rupture) and indirect effects such as submergence, jet impingement, and pipe whip, as applicable.	No	The flooding analysis addresses submergence of equipment, but does not fully investigate failures due to spray, humidity, heat, pipe whip, and jet impingement that are associated with flooding events. After investigation it is possible that these failures will not be applicable, but the analysis needs to investigate to be certain.	Again, based on the familiarity with the original NAPS internal flooding model, the requirements of this SR were generally addressed. This is mostly a documentation issue. Although, form the base model and model maintenance points of view this lack of documentation is not acceptable, for this one time extension request, the impact on the risk insights is considered to be negligible.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IF-E7	For each flood scenario, REVIEW the LERF analysis to confirm applicability of the LERF sequences. If appropriate LERF sequences do not exist, MODIFY the LERF analysis as necessary to account for any unique flood-induced scenarios or phenomena in accordance with the applicable requirements described in para. 4.5.9.	No	LERF was not included in the current flooding analysis.	This is a limitation of the assessment. However, as stated previously, this SR's requirements does not have an impact on the 1J1 LAR for NAPS because, in the most important flood areas, the redundant components are located in the same area. So, even if the LERF impact is optimistically not modeled adequately, the change in LERF due to the unavailability of 1J1 MCC is not expected to be significantly affected. That is, form the base model point of view, it is important to address the LERF impact due to the internal flooding hazard. However, form the 1J1 MCC point of view, the impact on the change is minimal.
IF-E8	CONDUCT a walkdown(s) to verify the accuracy of information obtained from plant information sources and to obtain or verify inputs to (a) engineering analyses (b) human reliability analyses (c) spray or other applicable impact assessments (d) screening decisions Note: A walkdown(s) may be done in conjunction with the requirements of IF-A4, IF-B3a, and IF-C9.	No	Draft Volume IF.1 refers frequently to walkdowns performed in 2001 and discusses findings from this walkdown. There is no other record of this walkdown, however.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IF-F1	DOCUMENT the internal flooding analysis in a manner that facilitates PRA applications, upgrades, and peer review.	No	HLR-IF-F specifies that the internal flooding analysis shall be documented consistent with the applicable supporting requirements. The NAPS Flooding Analysis is IPEEE vintage. A new draft document (Volume IF.1) has been developed that compiles the information from the original IPEEE and its supporting documents, but this new draft document does not update the model. A significant amount of detail needs to be added to the current documentation to satisfy the documentation requirements of the ASME PRA standard. The methodologies for defining flood areas and screening flood areas from further analysis need to be clearly described. Propagation of fluid throughout the plant needs to be better described. The development of initiating event frequencies needs more detail. The flooding HRA is sorely lacking in detail. Flooding scenarios need to be described in detail including flood sources, the characteristics of the break, equipment failures, flood propagation and associated timing, flood mitigation, and operator actions. Assumptions associated with the flooding analysis need to be clearly stated and technically justified. Survivability of structures such as doors and walls need to be technically assessed. Also, sources of uncertainty associated with the flooding analysis need to be identified.	This is an important enhancement. However, as stated in response to the other internal flooding "not met" SRs, this is mostly a documentation issue and due to the nature of flooding hazard at NAPS (location of redundant components), it is not expected that the internal flooding risk would be significantly impacted by the 1J1 MCC unavailability.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IF-F2	DOCUMENT the process used to identify flood sources, flood areas, flood pathways, flood scenarios, and their screening, and internal flood model development and quantification. For example, this documentation typically includes (a) flood sources identified in the analysis, rules used to screen out these sources, and the resulting list of sources to be further examined (b) flood areas used in the analysis and the reason for eliminating areas from further analysis (c) propagation pathways between flood areas and key assumptions, calculations, or other bases for eliminating or justifying propagation pathways (d) accident mitigating features and barriers credited in the analysis, the extent to which they were credited, and associated justification (e) key assumptions or calculations used in the determination of the impacts of submergence, spray, temperature, or other flood-induced effects on equipment operability (f) screening criteria used in the analysis (g) flooding scenarios considered, screened, and retained (h) description of how the internal event analysis models were modified to model these remaining internal flooding scenarios (i) flood frequencies, component unreliabilities/unavailabilities, and HEPs used in the analysis (i.e., the data values unique to the flooding analysis) (j) calculations or other analyses used to support or refine the flooding evaluation (k) results of the internal flooding analysis, consistent with the quantification requirements provided in HLR QU-D	No	See assessment comments for IF-F1	See comment on IF-F1.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
IF-F3	DOCUMENT the key assumptions and key sources of uncertainty associated with the internal flooding analysis.	No	See assessment comments for IF-F1	See comment for IF-F1
QU-A2a	PROVIDE estimates of the individual sequences in a manner consistent with the estimation of total CDF to identify significant accident sequences/cutsets and confirm the logic is appropriately reflected. The estimates may be accomplished by using either fault tree linking or event trees with conditional split fractions.	No	As noted in Section 2.1.1 of the QU.1 (input) notebook, NAPS uses a linked fault tree quantification approach. The QU.2 (results) notebook documents the CDF for all accident sequences that contribute at least 0.1% of total CDF and comprise >95% of total CDF. Section 2.4.1 describes the focused verification review performed of the model changes to ensure the results are consistent with the logic changes. The attachment 6 spreadsheet contains the changes made to the model along with the results impacts for each model change, and the attachment 8 document contains the results of the focused review. The QU.2 notebook shows that attachment 6 contains the CDF importance data and attachment 8 contains a bulleted summary of the model and data changes made (inconsistent with what the separate documents for those attachments show).	Based on a review of the comments in "Assessment Comments column, it is not clear why this SR is characterized as "Not Met." The characterization could be due to documentation only. Therefore, it is concluded that the SR is basically met.
QU-A2b	ESTIMATE the mean CDF from internal events, accounting for the "state-of- knowledge" correlation between event probabilities when significant [Note (1)].	No	The intent of this SR is to provide a true mean value for the CDF. In order to calculate the true MEAN value it is necessary to perform an uncertainty calculation, including correlation of data from similar sources. The QU.1 (input) and QU.2 (results) notebooks do not include a parametric uncertainty analysis. Although QU.1 does note that the BED file contains uncertainty distribution data and the basic event uncertainty data in the parameter file is documented in the data notebooks (section 2.5), and that uncertainty analyses can be performed on the equation files (section 4.0), there is no such analysis mentioned in QU.2. There are a few basic events in the parameter file (N05A_16C.prm) that do not contain uncertainty distribution data.	Although a rigorous analysis may provide better insights, it is judged that the impact on this one-time extension is negligible (i.e., the impact on the delta is negligible).

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
QU-A4	INCLUDE recovery actions in the quantification process in applicable sequences and cutsets (see HR-H1, HR-H2, and HR-H3).	No	The QU.1 (input) notebook describes the process for incorporating recovery actions into the quantification (section 2.1.1), describes the basis for developing the recovery actions (section 2.4.3), and presents the specific recovery events modeled in the fault trees or event trees, the events used in the rule-based recovery files, and the rules for applying those events (attachment 4). It is noted that one recovery event (MULTIPLYBY2P0, in T6P09.edt) has a value greater than 1.0, which could result in cutsets missing from the final total CDF equation, if they were dropped by the truncation before the multiplier was applied. However, the BED file is updated at the start of the model quantification batch process to set basic events to 1 so that the cutset are not truncated. The events are then reset to their nominal values at the end of the batch process. This process should be described as part of the quantification methodology. The recoveries and their application and reasonable, although some of the tables in attachment 4 do have some bits of information missing (i.e., blank fields).	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
QU-B1	PERFORM quantification using computer codes that have been demonstrated to generate appropriate results when compared to those from accepted algorithms. IDENTIFY method-specific limitations and features that could impact the results.	No	The current quantification code (WinNupra) for NAPS is described in section 4.0 of the QU.1 (input) notebook. The description includes reference to a Dominion technical report that demonstrates the code generates appropriate results, and to the WinNupra user's manual for a discussion of the software limitations. Both of these references pertain to version 2.1 of WinNupra, although the model currently uses version 3.0 (service release 4). An updated version of the code qualification technical report exists for version 3.0 (service release 2) and includes acknowledgement of the thorough V&V performed by the WinNupra vendor and presents the testing and results comparison performed with the NAPS model. Model size, file capacity and display limitations are discussed in the manual. (Upon conversion from WinNupra to CAFTA, the entire discussion of code acceptability and limitations will require revision.) Section 2.0 of the QU.1 notebook indicates that model and solution limitations are documented in are documented in Part II of the PRA model notebook, but this part has not yet been developed (although some limitations may be available in the IPE submittal).	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
QU-B3	ESTABLISH truncation limits by an iterative process of demonstrating that the overall model results converge and that no significant accident sequences are inadvertently eliminated. For example, convergence can be considered sufficient when successive reductions in truncation value of one decade result in decreasing changes in CDF or LERF, and the final change is less than 5%.	No	The QU.2 (results) notebook includes a truncation sensitivity for the overall CDF equation (section 2.4.7) that adequately demonstrates that the overall results converge such that no significant accident sequences are eliminated. Model solution at the final truncation value resulted in less than a 5% increase in CDF that at the truncation level a decade higher. A similar truncation sensitivity is not performed for LERF.	Based on a review of comments provided in the "Assessment Column" it seems that from the CDF point of view the intent of the SR's requirements is met. From the LERF point of view, it is judged that, although a similar evaluation is not performed for LERF, it is expected that 1) the results would not be significantly different to the CDF evaluation and 2) from the delta CDF calculation point of view, the impact on LERF would not be significant.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
QU-B7a	IDENTIFY cutsets (or sequences) containing mutually exclusive events in the results.	No	As described in the QU.1 (input) notebook (sections 2.1.1 and 2.4.1), combinations of mutually exclusive (MEX) events (such as disallowed maintenance, DAM, combinations) have been identified and are represented logically in a DAM fault tree model. The MEX combinations and DAM logic are presented in the QU.1 notebook (Table 2-1 and attachment 3). Although Table 2-1 provides the basis for removing each MEX combination, there is a notation that the table is yet to be completed. From review of the DAM fault tree there appear to be a large number of combinations that need to be added to Table 2-1 and their basis documented.	This seems to be a documentation issue only. Also, as stated above, a number of pre-initiator operator errors are included in the PRA model. Although a rigorous analysis of such events could result in the identification of additional items, pre-initiator
QU-D1a	REVIEW a sample of the significant accident sequences/cutsets sufficient to determine that the logic of the cutset or sequence is correct.	No	Dominant accident sequences (those that contribute at least 0.1% of total CDF and comprise >95% of total CDF) are presented in section 2.3.2 of the QU.2 (results) notebook, and the top ten (contributing at least 2% and comprising roughly 64% of TCDF) are discussed in attachment 10 as verification that the sequence is correct. A comparison of the significant sequences to those from the prior model version is also performed (Table 3). In addition, all cutsets are documented in WinNUPRA file N105AC30.eqp, the top 100 (contributing at least 0.1% and comprising roughly 75% of TCDF) are presented in attachment 4 and the top three (contributing at least 2% and comprising roughly 43% of TCDF) are described in section 2.3.3. Finally, section 2.4 describes the approach to the results verification reviews performed, including review of the top 100 cutsets for sensibility and comparison to the previous model version results, but there are few details regarding the review.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
QU-D1b	REVIEW the results of the PRA for modeling consistency (e.g., event sequence models consistency with systems models and success criteria) and operational consistency (e.g., plant configuration, procedures, and plant-specific and industry experience).	No	As described for QU-D1a, dominant accident sequences are presented and the top ten discussed in attachment 10, in addition all cutsets are documented in a WinNUPRA file, the top 100 are presented and the top three are described in the QU.2 notebook. Section 2.4 indicates a review of model results was performed to ensure they are consistent with the logic changes made to the model, as well as a review to verify the final initiator contributions, top sequences and important basic events are reasonable, and a cutset review for sensibility and to ensure any differences from previous results can be explained. However, other than the focused review of logic changes that is documented in attachments 6 and 8, there are no records of the overall cutset review. It is noted that the cutset review identified some incorrect cutsets and the model was revised to correct the problem.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
QU-D1c	REVIEW results to determine that the flag event settings, mutually exclusive event rules, and recovery rules yield logical results.	No	While the QU.1 (input) notebook describes the application of recovery rules (section 2.4.3), deletion of disallowed cutsets (section 2.4.1), and describes the use of house events and flags (section 2.5), there is no discussion of any review performed to verify the flags, MEX and recovery rules are applied properly and give logical results.	Documentation only. Note that flooding generally affects both trains of a system. As a result, model changes will impact both the overall results and 1J1 CT results equally. Therefore, no impact.
QU-D3	COMPARE results to those from similar plants and IDENTIFY causes for significant differences. For example: Why is LOCA a large contributor for one plant and not another?	No	Section 2.4.8 of the QU.2 (results) notebook discusses a comparison of the NAPS CDF contributions by initiator to that from the previous NAPS model and the Dominion Surry and Millstone 3 units. There are stated to be substantial differences in some of the relative contributions, and key plant or modeling differences that may contribute to the differences in results are described. However, there is no comparison to any other (non-Dominion) Westinghouse plants, the actual comparison (e.g., a table listing the various plant results) is not documented, and specific causes for specific differences in results are not identified.	Based on the experience in dealing with this issue, this is mostly a documentation issue and the results are not affected by this apparent gap. Also, it is noted that during MSPI, the NAPS success criteria were not identified as outliers.
QU-D4	REVIEW a sampling of nonsignificant accident cutsets or sequences to determine they are reasonable and have physical meaning.	No	There is no documentation of a review of nonsignificant accident sequences or cutsets in either the QU.1 (input) or QU.2 (results) notebook.	Based on the experience in dealing with this issue, this is mostly a documentation issue and the results are not affected by this apparent gap.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
QU-D5a	IDENTIFY significant contributors to CDF, such as initiating events, accident sequences, equipment failures, common cause failures, and operator errors. INCLUDE SSCs and operator actions that contribute to initiating event frequencies and event mitigation.	No	Table 1 in the QU.2 (results) notebook identifies all of the initiating event contributions, Table 2 identifies the significant accident sequence contributions, attachment 6 (not 5) presents the top 100 events (of all types) based on CDF importance (RAW), Table 4 presents the top 6 operator errors, initiating events, and component failures based on RAW, and Table 5 presents the top 10 HFEs based on Fussell-Vesely. The listings in Tables 4 and 5 and attachment 6 do not include all significant (F-V greater than 0.005 or RAW greater than 2) contributors, there is no specific identification of the significant CCF contributors, and there is no specific identification of contributors to IE frequencies.	Based on the work that has already been performed (as described in "Assessment Comments" column, the intent of this SR is met.
QU-D5b	REVIEW the importance of components and basic events to determine that they make logical sense.	No	Tables 4 and 5 and attachment 6 in the QU.2 (results) notebook provide a list of top component importance measures, but there is no discussion of a specific review performed to determine the results are sensible.	Based on the work that has already been performed (as described in "Assessment Comments" column, the intent of this SR is met.
QU-E1	IDENTIFY key sources of model uncertainty.	No	Other than some mention in the QU.1 (input) notebook of data uncertainty parameters, there is no discussion of the sources of uncertainty in the NAPS model.	Although a rigorous uncertainty identification and documentation may provide better insights when using the model for permanent changes, it is judged that the impact on this one-time extension is negligible.
QU-E2	IDENTIFY key assumptions made in the development of the PRA model.	No	The QU.1 (input) notebook indicates that key modeling assumptions are documented in are documented in Part II of the PRA model notebook, but this part has not yet been developed (although some key assumptions may be available in the IPE submittal). The different element notebooks (IE, AS, SC, SY, etc.) do include specific assumptions related to the development of that element, but there is typically no discussion of the sources of uncertainty those assumptions relate to and the impacts of those assumptions.	Although a rigorous identification and documentation of key assumptions may provide better insights when using the model for permanent changes, it is judged that the impact on this one-time extension is negligible.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
QU-E3	ESTIMATE the uncertainty interval of the overall CDF results. ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G9, IE-C13), taking into account the "state-of-knowledge" correlation	No	The QU.1 (input) and QU.2 (results) notebooks do not include a parametric uncertainty analysis. Although QU.1 does note that the BED file contains uncertainty distribution data and the basic event uncertainty data in the parameter file is documented in the data notebooks (section 2.5), and that uncertainty analyses can be performed on the equation files (section 4.0), there is no such analysis mentioned in QU.2. There are a few basic events in the parameter file (N05A_16C.prm) that do not contain uncertainty distribution data.	Although a rigorous uncertainty identification and documentation may provide better insights when using the model for permanent changes, it is judged that the impact on this one-time extension is negligible.
QU-E4	EVALUATE the sensitivity of the results to key model uncertainties and key assumptions using sensitivity analyses [Note (1)].	No	Section 2.7 of the QU.1 (input) notebook contains a sensitivity analysis for the probability of maintaining flow from the AFW TDP following battery depletion, and mentions a truncation sensitivity analysis (which is documented in the QU.2) notebook. These sensitivity cases are not related to the identification of sources of model uncertainty and the associated assumptions, and there are no other sensitivity cases mentioned.	Although a rigorous uncertainty identification and documentation may provide better insights when using the model for permanent changes, it is judged that the impact on this one-time extension is negligible.
QU-F1	DOCUMENT the model quantification in a manner that facilitates PRA applications, upgrades, and peer review.	No	The NAPS QU.1 (inputs) and QU.2 (results) notebooks are dated 12/05 and 03/07 respectively, so it is not clear that the two notebooks document the same model version (QU.1 refers to the N05A model and QU.2 refers to the N105A model). The model quantification files are all dated 12/05. It is also not clear if the notebooks have been signed off and released.	This is a documentation issue only. The model has been affectively used for a number of years. No impact on the LAR.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
QU-F2	DOCUMENT the model integration process, including any recovery analysis, and the results of the quantification including uncertainty and sensitivity analyses. For example, documentation typically includes (a) records of the process/results when adding nonrecovery terms as part of the final quantification (b) records of the cutset review process (c) a general description of the quantification process including accounting for systems successes, the truncation values used, how recovery and post-initiator HFEs are applied (d) the process and results for establishing the truncation screening values for final quantification demonstrating that convergence towards a stable result was achieved (e) the total plant CDF and contributions from the different initiating events and accident classes (f) the accident sequences and their contributing cutsets (g) equipment or human actions that are the key factors in causing the accidents to be nondominant (h) the results of all sensitivity studies (i) the uncertainty distribution for the total CDF (j) importance measure results (k) a list of mutually exclusive events eliminated from the resulting cutsets and their bases for Elimination (l) asymmetries in quantitative modeling to provide application users the necessary understanding regarding why such asymmetries are present in the model (m) the process used to illustrate the computer code(s) used to perform the quantification will yield correct results process	No	The NAPS quantification documentation in the QU.1 (input) and QU.2 (results) notebooks addresses many of the items noted in this SR, but does not include discussion on: records/results from addition of recovery terms (a), records from cutset review (b), the contributing cutsets from the various accident sequences (f), equipment or human failures that cause cutsets to be non-dominant (g), sensitivity studies (h), uncertainty distribution (i), modeling asymmetries and their bases (l), quantification software acceptance (m), and other items noted in previous SRs.	The intent of the SR is mostly met. Based on experience dealing with the enhancements identified here, the delta risk estimates are not expected to change significantly.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
QU-F3	DOCUMENT the significant contributors (such as initiating events, accident sequences, basic events) to CDF in the PRA results summary. PROVIDE a detailed description of significant accident sequences or functional failure groups.	No	Table 1 in the QU.2 (results) notebook identifies all of the initiating event contributions, Table 2 identifies the significant accident sequence contributions (the top ten accident sequences are discussed in attachment 10), and attachment 6 (not 5) presents the top 100 events (of all types) based on CDF importance (RAW). There is not a detailed description of all significant accident sequences, and the event listing in attachment 6 does not include all significant (F-V greater than 0.005 or RAW greater than 2) basic events.	The intent of the SR is mostly met. Based on experience dealing with the enhancements identified here, the delta risk estimates are not expected to change significantly.
QU-F4	DOCUMENT key assumptions and key sources of uncertainty, such as: possible optimistic or conservative success criteria, suitability of the reliability data, possible modeling uncertainties (modeling limitations due to the method selected), degree of completeness in the selection of initiating events, possible spatial dependencies, etc.	No	Although the different element notebooks (IE, AS, SC, SY, etc.) do include specific assumptions related to the development of that element, there is no discussion in the QU.1 (input) and QU.2 (results) notebooks of the sources of uncertainty in the NAPS model, nor of the assumptions associated with those uncertainties.	The intent of the SR is mostly met. Based on experience dealing with the enhancements identified here, the delta risk estimates are not expected to change significantly.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
QU-F5	DOCUMENT limitations in the quantification process that would impact applications.	No	Section 2.0 of the QU.1 Input) notebook indicates that model and solution limitations are documented in Part II of the PRA model notebook, but this part has not yet been developed (although some limitations may be available in the IPE submittal). The quantification code (WinNupra) for NAPS is described in section 4.0 of the QU.1 notebook. The description includes reference to the WinNupra user's manual for a discussion of the software limitations. Model size, file capacity and display limitations are discussed in the manual. This reference pertains to version 2.1 of WinNupra, although the model currently uses version 3.0.	The intent of the SR is mostly met. Based on experience dealing with the enhancements identified here, the delta risk estimates are not expected to change significantly.
QU-F6	DOCUMENT the quantitative definition used for significant basic event, significant cutset, significant accident sequence. If other than the definition used in Section 2, JUSTIFY the alternative.	No	There is no definition in the QU.1 (input) and QU.2 (results) notebooks for significant accident sequence, significant basic event, or significant cutset. Such information might eventually be included in Part II of the PRA model notebook (PRA Model Discussion), but this part has not yet been developed. The results discussion in the QU.2 notebook appears to use the terms "dominant" and "significant" interchangeably, although neither term is defined, and many of the results do not include all significant (accident sequence, basic event, or cutset) contributors as defined in Section 2 of the standard.	The intent of the SR is mostly met. Based on experience dealing with the enhancements identified here, the delta risk estimates are not expected to change significantly. Note that for the 1J1 application, top sequences are reviewed.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
LE-A1	IDENTIFY those physical characteristics at the time of core damage that can influence LERF. Examples include (a) RCS pressure (high RCS pressure can result in high pressure melt ejection) (b) status of emergency core coolant systems (failure in injection can result in a dry cavity and extensive Core Concrete Interaction) (c) status of containment isolation (failure of isolation can result in an unscrubbed release) (d) status of containment heat removal (e) containment integrity (e.g., vented, bypassed, or failed) (f) steam generator pressure and water level (PWRs) (g) status of containment inerting (BWRs)	No	The physical characteristics that can influence LERF were identified in the IPE, with discussion of update provided in SM-1464, Rev. 0. For the first five examples listed, the applicable headings in the WinNUCAP diagram are RCSPRESS, INVESSINJ, CONISOLAT, CNHEATREM and CONISOLAT. For part f, the SG pressure and water level were not modeled because of the simplified Induced SGTR model (see LE-D5). Although the modeling of ISGTR is addressed in LE-D5, the LE-A1 SR will also be listed as not met since SG pressure and water level should be considerations in the PDS modeling.	Based on a review of SPS LE 1 volume 1 (Surry's Level 2 note book), updates to the LERF modeling (e.g. induced SGTR calculations and SPS-specific MAAP analyses) in the LE.2 and LE.4 notebooks resulted in new plant damage states and new conditional probabilities of large, early release. However, these changes impact both the base case and the 1J1 case and, as a result, are not expected to result in a significant change in the delta risk calculations.
LE-A3	IDENTIFY how the physical characteristics identified in LE-A1 and the accident sequence characteristics identified in LE-A2 are addressed in the LERF analysis. For example (a) which characteristics are addressed in the Level 1 event trees (b) which characteristics, if any, are addressed in bridge trees (c) which characteristics, if any, are addressed in the containment event trees JUSTIFY any characteristics identified in LE-A1 or LE-A2 that are excluded from the LERF analysis.	No	The NAPS IPE and SM-1464 describe how each of the characteristics is evaluated. The NAPS Level 1 model uses bridge trees in the baseline PRA model, so no separate bridge tree evaluation is required. Per SM-1464, the large containment isolation failures are excluded from the LERF analysis with basis given referencing a report that found no credible large failures. The small containment isolation failures are modeled. However, this SR is listed as "not met" because the exclusion of the SG characteristics from LE-A1 is not discussed in detail in the documentation (the documentation only states the simplified ISGTR modeling in SM-1243 without adequate justification.	Based on significant amount of work which meets the requirements of this SR and the fact that following closure of similar comment for SPS, the LERF estimate was decrease, it is judged that the impact of this potential issue on the delta risk calculation is minimal.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
LE-A5	DEFINE plant damage states consistent with LE-A1, LE-A2, LE-A3, and LE-A4.	No	The Plant Damage States are defined and binned in the IPE and then updated in SM-1243 and SM-1464. However, it appears that the updates have not been incorporated into the model (a final, signed version of SM-1464 could not be found, and the QU.2 LERF results appear to have used the conditional LER probabilities prior to the SM-1464 update), so the PDS definitions from the updates do not match the calculation updates. The WinNUPRA model provided for the self assessment did not have LERF PRM values, but it is not clear: The North Anna BED file shows 26 LERF basic events, which matches the 26 PDSs from the update (the NAPS IPE had only 25 PDSs), but the QU.2 LERF results appear to have used the old probabilities. Because it is not clear if the updates have been incorporated, and because of the confusion in the documentation status, this SR is considered not met.	Based on significant amount of work which meets the requirements of this SR and the fact that following closure of similar comment for SPS, the LERF estimate decreased, it is judged that the impact of this potential issue on the delta risk calculation is minimal.
LE-B3	UTILIZE supporting engineering analyses in accordance with the applicable requirements of Table 4.5.3-2(b).	No	The containment capacity analysis used in the Level 2 was developed for Surry (SWEC and NUREG-1150 analyses), but was justified as being applicable to NAPS based on a detailed comparison of the plant designs. To meet Category II of the Standard Table 4.5.3-2(b), appropriate realistic generic analyses/evaluations are acceptable. Therefore, the use of the Surry plant-specific assessments with justification is acceptable. However, this SR is considered not met because many of the containment challenges (pressure, temperature and timing considerations) were determined using MAAP 3 calculations at the time of the IPE. The ASME PRA Standard does not specify a particular version of MAAP to be "realistic", although some improvements in version 4 of the MAAP code are considered to be more realistic than the version 3 code. In any case, the selection of representative cases for MAAP analysis should be updated based on the current dominant results, which have changed significantly since the IPE.	Based on significant amount of work which meet the requirements of this SR and the fact that following update to SPS Level 2 model, the LERF estimate decreased, it is judged that the impact of this potential issue on the delta risk calculation is minimal.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
LE-C1	DEVELOP accident sequences to a level of detail to account for the potential contributors identified in LE-B1 and analyzed in LE-B2. Compare the containment challenges analyzed in LE-B with the containment structural capability analyzed in LE-D and identify accident progressions that have the potential for a large early release. JUSTIFY any generic or plant-specific calculations or references used to categorize releases as non-LERF contributors based on release magnitude or timing. NUREG/CR-6595, App. A [Note (1)] provides an acceptable definition of LERF source terms.	No	The accident sequences are developed in a Containment Event Tree (CET) that is detailed in the NAPS WinNUCAP model (documented in the IPE and updated in SM-1464). The CET evaluates the potential LERF contributors that were identified in LE-B. The containment structural capability was clearly reviewed for its effect on late releases, but not explicitly for early releases. Rather, the SM-1243 and SM-1464 updates rely on industry (NUREG) guidance that containment pressure challenges are not considered credible in causing LERF in large, dry containments. This basis appears valid, but as the ASME requirement calls for comparison of containment challenges with the containment structural capability, more justification should be provided.	Based on significant amount of work which meet the requirements of this SR and the fact that following update to SPS Level 2 model, the LERF estimate decreased, it is judged that the impact of this potential issue on the delta risk calculation is minimal.
LE-C2a	INCLUDE realistic treatment of feasible operator actions following the onset of core damage consistent with applicable procedures, e.g., EOPs/SAMGs, proceduralized actions, or Technical Support Center guidance.	No	System level actions for RS, SW, etc., are described in the Level 1 System Analysis notebooks. Offsite power recovery is the only action involving manipulation explicitly shown in the CET, and its probability is maintained within the Level 1 Data Analysis. SAMGs have not been incorporated into the NAPS Level 2 analysis, other than initiation of low pressure injection after the onset of core damage. The SAMGs should be reviewed for other relevant actions (e.g., late RCS depressurization to reduce the chance of induced SGTR).	Based on significant amount of work which meet the requirements of this SR and the fact that following update to SPS Level 2 model, the LERF estimate decreased, it is judged that the impact of this potential issue on the delta risk calculation is minimal.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
LE-C2b	REVIEW significant accident progression sequences resulting in a large early release to determine if repair of equipment can be credited. JUSTIFY credit given for repair [i.e., ensure that plant conditions do not preclude repair and actuarial data exists from which to estimate the repair failure probability (see SY-A22, DA-C14, and DA-D8)]. AC power recovery based on generic data applicable to the plant is acceptable.	No	Offsite power recovery after core damage is credited in the NAPS bridge trees and documented in the Level 1 analysis. However, no further review of the significant accident progression sequences for possible equipment repair credit was documented. If a review has been completed, it needs to be documented. If no repair of equipment is to be credited, then the review and disposition should be presented to meet the SR.	Based on significant amount of work which meet the requirements of this SR and the fact that following update to SPS Level 2 model (which dealt with the same issue), the LERF estimate decreased, it is judged that the impact of this potential issue on the delta risk calculation is minimal.
LE-C3	INCLUDE model logic necessary to provide a realistic estimation of the significant accident progression sequences resulting in a large early release. INCLUDE mitigating actions by operating staff, effect of fission product scrubbing on radionuclide release, and expected beneficial failures in significant accident progression sequences. PROVIDE technical justification (by plant-specific or applicable generic calculations demonstrating the feasibility of the actions, scrubbing mechanisms, or beneficial failures) supporting the inclusion of any of these features.	No	The CET top events, each of which represents a phenomenological consideration, are evaluated using the WinNUCAP Decomposition Event Trees (DETs) to assess the considerations as they apply to LERF (and to other CET end states). This computational methodology is appropriate for LERF analysis. However, this SR is considered not met because the evaluation of mitigating actions is not plant specific (see SR LE-C10 for specifics) and operator mitigating actions are not thoroughly evaluated (see SR LE-C6 for specifics). Beneficial failures are not credited in the model.	Based on significant amount of work which meet the requirements of this SR and the fact that following update to SPS Level 2 model (which dealt with the same issue), the LERF estimate decreased, it is judged that the impact of this potential issue on the delta risk calculation is minimal.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
LE-C4	USE appropriate realistic generic or plant-specific analyses for system success criteria for the significant accident progression sequences. USE conservative or a combination of conservative and realistic system success criteria for non-risk significant accident progression sequences.	No	Realistic, plant-specific success criteria were developed for significant sequences using the MAAP 3.0b code at the time of the IPE. These analyses were applied to system success criteria in the fault trees (Level 1 model) and to the containment (containment temperatures/pressures). However, the MAAP code has undergone significant upgrades and the dominant sequences have changed since the IPE, and cases should be analyzed/re-analyzed to ensure current accuracy.	Based on significant amount of work which meet the requirements of this SR and the fact that following update to SPS Level 2 model (which dealt with the same issue), the LERF estimate decreased, it is judged that the impact of this potential issue on the delta risk calculation is minimal.
LE-C6	In crediting HFEs that support the accident progression analysis, USE the applicable requirements of para. 4.5.5, as appropriate for the level of detail of the analysis.	No	System level operator actions are described in the Level 1 System Analysis notebooks. Offsite power recovery is the only action involving manipulation explicitly shown in the CET, and its probability is maintained within the Level 1 Data Analysis. SAMGs have not been incorporated into the NAPS Level 2 analysis, except initiation of low pressure injection after the onset of core damage, and its HEP has not been calculated in detail.	Based on significant amount of work which meet the requirements of this SR and the fact that following update to SPS Level 2 model (which dealt with the same issue), the LERF estimate decreased, it is judged that the impact of this potential issue on the delta risk calculation is minimal.
LE-C8a	JUSTIFY any credit given for equipment survivability or human actions under adverse environments.	No	It does not appear that equipment survivability was examined, beyond that for RS pumps in the consideration of containment failure in the RS-EARLY and RS-LATE events in Section 3. The equipment survivability should be examined and explicitly discussed to meet the SR. In general, equipment is capable (and credited) of performing at levels significantly worse than the design basis conditions. For example, even though PORVs, spray headers, and SG equipment are credited up until containment failure (pressures and temperatures far greater than design basis), they will be subject to worse than design basis conditions in a severe accident. Such credit should be provided in the documentation.	Based on the fact that following update to SPS Level 2 model (which dealt with the same issue), the LERF estimate decreased, it is judged that the impact of this potential issue on the delta risk calculation is minimal. Also, in response to the same comment in SPS model, the analyst note that the SPS's model does not credit equipment that is not EQ qualified for adverse environments. Likewise, operator actions are not credited for actions in areas where the environment is adverse. The NAPS model was based on the SPS model and is expected to be similar in this respect.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
LE-C8b	REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions during accident progression that could reduce LERF. USE conservative or a combination of conservative and realistic treatment for non-significant accident progression sequences.	No	The significant accident progression sequences were not reviewed explicitly for the consideration of continued equipment operation or operator actions to reduce the LERF.	Based on the fact that following update to SPS Level 2 model (which dealt with the same issue), the LERF estimate decreased and additionally significant changes to the model was not necessary, it is judged that the impact of this potential issue on the delta risk calculation is minimal.
LE-C9b	REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions after containment failure that could reduce LERF. USE conservative or a combination of conservative and realistic treatment for non-significant accident progression sequences.	No	Although a review of the significant accident progression sequences for post-containment failure operation might not identify any potential for LERF reduction, the review should be performed and documented to meet the SR.	This is a documentation issue only. No impact.
LE-C10	PERFORM a containment bypass analysis in a realistic manner. JUSTIFY any credit taken for scrubbing (i.e., provide an engineering basis for the decontamination factor used).	No	Containment bypass is treated in a realistic manner. Plant-specific ISLOCA modeling is developed in the Level 1 PRA. The scrubbing factor is taken from NUREG/CR-4551, but should be reviewed in more detail for plant-specific considerations. SGTR is separated into LERF and non-LERF in SM-1464 (PDS event CONBYPASS), but the definition of SGTR LERF vs. non-LERF should be validated using MAAP (or other means) beyond the qualitative definition used.	As stated by the reviewers in the "Assessment Comments" column, containment bypass is treated in a realistic manner. The issues identified here are mostly appear to be documentation issues only and are not expected to result in changes in risk insights for the 1J1 LAR.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
LE-D1b	EVALUATE the impact of accident progression conditions on containment seals, penetrations, hatches, drywell heads (BWRs), and vent pipe bellows. INCLUDE these impacts as potential containment challenges, is required. If generic analyses are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.	No	The NUREG-1150 analysis performed for Surry considered failures of the containment building penetrations (electrical, fluid, personnel hatch, equipment hatch, etc.) and found to be significantly less significant than over-pressure failure of the cylinder wall. To meet the intent of this SR, the NAPS documentation should specifically include a description of the analysis.	Documentation only. No impact.
LE-D3	PERFORM a realistic interfacing system failure probability analysis for the significant accident progression sequences resulting in a large early release. USE a conservative or a combination of conservative and realistic evaluation of interfacing system failure probability for non-significant accident progression sequences resulting in a large early release. INCLUDE behavior of piping relief valves, pump seals, and heat exchangers at applicable temperature and pressure conditions.	No	A detailed ISLOCA analysis was performed in for the IPE (referenced in NAPS Accident Sequence Notebook, AS.1). However, neither the AS.1 nor AS.2 notebook (which has the ISLOCA section blank) mentions a review or update to the analysis since the IPE. The analysis should be updated and documented to ensure continued realistic analysis.	No evidence is given that an update to the ISLOCA analysis would have an impact on the integrity of the ISLOCA model. Also, any changes are most likely to affect the base case and the 1J1 case. Therefore, the impact on the delta risk estimates is considered to be negligible.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
LE-D4	PERFORM a realistic secondary side isolation capability analysis for the significant accident progression sequences caused by SG tube release. USE a conservative or a combination of conservative and realistic evaluation of secondary side isolation capability for non-significant accident progression sequences resulting in a large early release. JUSTIFY applicability to the plant being evaluated. Analyses may consider realistic comparison with similar isolation capability in similar containment designs.	No	Secondary side isolation is explicitly and realistically modeled in the Level 1 System Analysis notebooks for pre-core damage consideration. However, secondary side isolation during a SGTR should also consider the additional number of demands on the relief valves in the progression to core damage. It is possible that some sequences considered "isolated" in the Level 1 analysis could be unisolated in the Level 2 analysis. Also, version 4 of the MAAP code provides better SGTR analysis than had been used for the IPE with version 3 of the code.	The issue indentified here seems to be an enhancement. Also, although a rigorous analysis may provide better insights when using the model for permanent changes, it is judged that the impact on this one-time extension.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
LE-D5	PERFORM an analysis of thermal-induced SG tube rupture that includes plant-specific procedures and design features and conditions that could impact tube failure. An acceptable approach is one that arrives at a plant-specific split fraction by selecting the SG tube conditional failure probabilities based on NUREG-1570 [Note (3)] or similar evaluation for induced SG failure of similarly designed SGs and loop piping. SELECT failure probabilities based on a) RCS and SG post-accident conditions sufficient to describe the important risk outcomes (b) secondary side conditions including plant-specific treatment of MSSV and ADV failures JUSTIFY key assumptions and election of key inputs. An acceptable justification can be obtained by the extrapolation of the information in NUREG-1570 to obtain plant-specific models, use of reasonably bounding assumptions, or performance of sensitivity studies indicating low sensitivity to changes in the range in question.	No	Induced SGTR is considered in the SM-1243 and SM-1464 updates to the NAPS Level 2 (CET top event RCSFAIL) with a simplified conditional probability that was developed generically. A plant-specific assessment should be performed using a detailed method, such as is presented in NUREG-1570.	The 1J1 MCC availability would not have an impact on the identified enhancement, since it has not impact on the frequency of its occurrence or mitigation of its consequences.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
LE-E1	SELECT parameter values for equipment and operator response in the accident progression analysis consistent with the applicable requirements of paras. 4.5.5 and 4.5.6 including consideration of the severe accident plant conditions, as appropriate for the level of detail of the analysis.	No	The system level data are documented in the Level 1 analysis (SY and DA notebooks). The Level 2 parameters are documented in the IPE, with some updates in SM-1243 and SM-1464. The Level 2 data are generally well referenced, although some estimations are made, which should either be verified or examined in sensitivity analyses. The operator action for initiating LPI after the onset of core damage requires a more detailed HRA.	The 1J1 MCC availability would not have an impact on the identified enhancement. The current analysis seems to meet almost all the SR requirements.
LE-E3	INCLUDE as LERF contributors potential large early release (LER) sequences identified from the results of the accident progression analysis of LE-C except those LER sequences justified as non-LERF contributors in LE-C1.	No	The determination of LERF vs. non-LERF sequences is presented in SM-1243 Section 4.7.3, SM-1464 Sections 5.7 and 5.9, and in the IPE. Source term calculations were performed for the Surry IPE and utilized for North Anna. The updated calculations identify which sequences are LER, but should have more clear ties to plant-specific calculations.	Documentation only. No impact.
LE-E4	QUANTIFY LERF consistent with the applicable requirements of Tables 4.5.8-2(a), 4.5.8-2(b), and 4.5.8-2(c). NOTE: The supporting requirements in these tables are written in CDF language. Under this requirement, the applicable quantification requirements in Table 4.5.8-2 should be interpreted base on the approach taken for the LERF model. For example, supporting requirement QU-A2 addresses the calculation of point estimate/mean CDF. Under this requirement, the application of QU-A2 would apply to the quantification of point estimate/mean LERF.	No	Calculation SM-1464 generated updated conditional LER probabilities, but no quantification could be found with these values. The QU.2 notebook appears to have used the conditional LER probabilities.	Documentation issue. The LERF estimate for NAPS seems to be reasonable.
LE-F1a	PERFORM a quantitative evaluation of the relative contribution to LERF from plant damage states and significant LERF contributors from Table 4.5.9-3.	No	A basic event importance analysis appears in Attachment 7 of the QU.2 notebook. However, no ranking by PDS or significant LERF contributor was documented.	Documentation issue. No impact

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
LE-F1b	REVIEW contributors for reasonableness (e.g., to assure excessive conservatisms have not skewed the results, level of plant specificity is appropriate for significant contributors, etc.).	No	The NAPS LE documentation does not include a review for reasonableness. Such a review should be documented.	Based on experience in dealing with similar issues, this is a documentation only issue.
LE-F2	PROVIDE uncertainty analysis that identifies the key sources of uncertainty and includes sensitivity studies for the significant contributors to LERF.	No	No uncertainty or sensitivity analyses were performed on the LERF results.	Although a rigorous uncertainty and sensitivity identification and documentation would provide better insights when using the model for permanent changes, it is judged that the impact on this one-time extension is negligible.
LE-F3	IDENTIFY contributors to LERF and characterize LERF uncertainties consistent with the applicable requirements of Tables 4.5.8-2(d) and 4.5.8-2(e). NOTE: The supporting requirements in these tables are written in CDF language. Under this requirement, the applicable requirements of Table 4.5.8 should be interpreted based on LERF, including characterizing key modeling uncertainties associated with the applicable contributors from Table 4.5.9-3. For example, supporting requirement QU-D5 addresses the significant contributors to CDF. Under this requirement, the contributors would be identified based on their contribution to LERF.	No	LERF contributors were not identified or their uncertainties characterized.	Although a rigorous uncertainty and sensitivity identification and documentation would provide better insights when using the model for permanent changes, it is judged that the impact on this one-time extension is negligible.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
LE-G1	DOCUMENT the LERF analysis in a manner that facilitates PRA applications, upgrades, and peer review.	No	Portions of the Level 2 analysis are documented in the IPE and updates in SM-1243 and SM-1464, making it difficult to review the current form of the model. The documentation would be significantly easier to follow if the model were contained in one document (an LE notebook). Also, it appears that the updates to the Level 2 model were not implemented into the PRA analyses.	Documentation only. No impact.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
LE-G2	DOCUMENT the process used to identify plant damage states and accident progression contributors, define accident progression sequences, evaluate accident progression analyses of containment capability, and quantify and review the LERF results. For example, this documentation typically includes (a) the plant damage states and their attributes, as used in the analysis (b) the method used to bin the accident sequences into plant damage states (c) the containment failure modes, phenomena, equipment failures and human actions considered in the development of the accident progression sequences and the justification for their inclusion or exclusion from the accident progression analysis (d) the treatment of factors influencing containment challenges and containment capability, as appropriate for the level of detail of the analysis (e) the basis for the containment capacity analysis including the identification of containment failure location(s), if applicable (f) the accident progression analysis sequences considered in the containment event trees (g) the basis for parameter estimates (h) the model integration process including the results of the quantification including uncertainty and sensitivity analyses, as appropriate for the level of detail of the analysis.	No	The PDS, CET and DET diagrams and their supporting logic are documented in the IPE and the various updates. However, there is no documented quantification of the LERF for the various requirements of the ASME Standard. The only documented LERF calculation is in QU.2, which presents the top 100 LERF cutsets from the WinNUPRA model. It appears that these cutsets do not include the Level 2 updates.	Based on experience in dealing with similar issues, this is a documentation only issue.

SR	Category II Requirements	Met?	Assessment Comments	Rev. 1 1J1 AOT
LE-G3	DOCUMENT the relative contribution of contributors (i.e., plant damage states, accident progression sequences, phenomena, containment challenges, containment failure modes) to LERF.	No	The North Anna LERF results appear in cutset format in QU.2. No review of the PDS, accident progression sequence, etc. contributions to LERF was documented.	Based on experience in dealing with similar issues, this is a documentation only issue.
LE-G4	DOCUMENT key assumptions and key sources of uncertainty associated with the LERF analysis, including results and important insights from sensitivity studies.	No	Assumptions are listed as they are used in the Level 2 analyses, but no formal review to identify key assumptions and sources of uncertainties was performed. Also, no sensitivities were performed to determine the effect of assumptions/uncertainties on the LERF.	Based on experience in dealing with similar issues, this is a documentation only issue.
LE-G5	IDENTIFY limitations in the LERF analysis that would impact applications.	No	Limitations that could impact applications were not identified.	Based on experience in dealing with similar issues, this is a documentation only issue.
LE-G6	DOCUMENT the quantitative definition used for significant accident than the definition used in Section 2, JUSTIFY the alternative.	No	The NAPS Level 2 analysis did not identify "significant" accident sequences in terms of LERF.	Based on experience in dealing with similar issues, this is a documentation only issue.

Attachment F – Local Manual Operator Actions Credited in the Internal Events PRA Model

Local Manual Operator Actions Credited in the Internal Events PRA Model			
Basic Events	**Descriptions**	**Local Action?**	**Fault Trees**
HEP-0AP10	0-AP-10 LOSS OF ELECTRICAL POWER		EH1, EHA, EJ1, EJA, EP1, EPA
HEP-0AP55-10HR	0-AP-55 Loss Of ESGR/MCR HVAC	Yes	HVA
HEP-0OP49:4A	0-OP-49.4A VALVE CHECKOFF MCR A/C SERVICE WATER	Yes	HV1, HVA
HEP-0OP6:3	0-OP-6.3 Operation Of 0-AAC-DG-0M SBO Diesel	yes	AAC
HEP-1AP15-1E	1-AP-15 Loss Of CC Step 1 Restore SW To CC Heat Exchanger	Yes	CC1, CCA
HEP-1AP22:5	1-AP-22.5 Loss Of Emergency Condensate Storage Tank	Yes	AF1
HEP-1E0-16	1-E-0 Rx Trip Or SI Step 6, Attachment 5 Charging Pump Alignment	Yes	HH1, SC1
HEP-1E3-3	1-E-3 SGTR Step 3 Isolate Flow From Ruptured S/G	yes	SG1
HEP-1FRS:1-4	1-FR-S.1 ATWS Step 4 Initiate Emergency Borate	yes	CH1
HEP-1FRS:1-5	1-FR-S.1 ATWS Step 5 Do Attach 2 Remote Reactor Trip	yes	FFT, RP1
HEP-1OP14:1-5:13	1-OP-14.1 Residual Heat Removal	yes	RH1
HEP-1OP21:6	1-OP-21.6 MCR And Relay Room Air Conditioning	Yes	HV1, HVA
HEP-1OP8:1	1-OP-8.1 Fail to Align Alternate Path	yes	SC1
HEP-CROSSTIE-G	Recover Seal Injection During General Condition	Yes	CHA
HEP-CROSSTIE-SBO	Recover TB cooling During SBO follow ECA-0.0 & AP-33.2	Yes	CHA
HEP-CROSSTIE-T8	Recover Seal Injection During T8 Event	Yes	CHA
HEP-MMP-C-MR-2	0-MCM-0803-01 Trouble Shooting & Repair MCR Chiller Units	Yes	HVA
HEP-SW-CHP-FP-PG	0-AP-12 STEP 17 RESTORE CHP COOLING WITH PG OR FP	yes	SW1
HEP-SW-ESGR-BC	0-AP-12 STEP 18 RESTORE ESGR COOLING FROM BEARING COOLING	yes	SW1
HEP-SW-LAKE-LAKE	0-AP-12 STEP 1 SW LAKE TO LAKE OPERATION	yes	SW1, SWA
HEP-1AP15-23	Operator failed to align Charging Suction to RWST during loss of IA	Yes	IT4