VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

December 17, 2004

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

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| License Nos. | NPF-4 |
| | NPF-7 |

**VIRGINIA ELECTRIC AND POWER COMPANY**
**NORTH ANNA POWER STATION UNITS 1 AND 2**
**PROPOSED TECHNICAL SPECIFICATIONS CHANGE REQUEST**
**INCREASED ECCS-LHSI, AFW, QUENCH SPRAY and CHEMICAL ADDITION SYSTEMS COMPLETION TIMES**

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion) requests amendments to Facility Operating License Numbers NPF-4 and NPF-7 in the form of changes to the Technical Specifications for North Anna Power Station Units 1 and 2. The proposed changes will increase the completion times for the Emergency Core Cooling System-Low Head Safety Injection subsystem, Auxiliary Feedwater, Quench Spray and Chemical Addition Systems from 72 hours to 7 days. The proposed changes are 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 Decisionmaking: Technical Specifications."

A discussion of the proposed changes is included in Attachment 1. Marked-up pages that identify the proposed changes and the Technical Specification pages that incorporate the proposed changes are provided in Attachments 2 and 3, respectively. Technical Specification Bases changes associated with the proposed changes are provided for information only. The Technical Specification Bases will be revised in accordance with the Technical Specification Bases Control Program, Technical Specification 5.5.13, following NRC approval of the license amendment.

The proposed changes have been reviewed and approved by the Station Nuclear Safety and Operating Committee and the Management Safety Review Committee.

In accordance with the requirements of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards. Our basis for that determination is included in Attachment 3. In addition, the proposed change has been determined to qualify for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). The basis for these determinations is included in Attachment 1.

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

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachments

Commitments made in this letter: None

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COMMONWEALTH OF VIRGINIA )
)
COUNTY OF HENRICO )

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz who is Vice President – Nuclear Engineering of Virginia Electric and Power Company. She has affirmed before me that she 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 her knowledge and belief.

Acknowledged before me this 17TH day of December, 2004.

My Commission Expires: May 31, 2006.

Vicki L. Hull
Notary Public

(SEAL)

**Attachment 1**

**Serial No. 04-381**

**Virginia Electric and Power Company**
**North Anna Power Station Units 1 and 2**
**Proposed Technical Specifications Changes For**
**Extended Fluid Systems Completion Times**

**Discussion of Change**

**North Anna Power Station**
**Units 1 and 2**
**Virginia Electric and Power Company**
**(Dominion)**

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**Discussion of Change**

## 1.0 Introduction

Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Dominion) requests an amendment to Facility Operating License Numbers NPF-4 and NPF-7 in the form of changes to the Technical Specifications (TS) for North Anna Power Station Units 1 and 2. The proposed changes will revise the completion time (CT) for the following systems: Low Head Safety Injection (LHSI) Emergency Core Cooling System (ECCS) Subsystem, Auxiliary Feedwater (AFW) System, and Quench Spray (QS) System including Chemical Addition System (CAS). The proposed changes are 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 Decisionmaking: Technical Specifications."

TS Bases changes, reflecting the proposed changes to the completion times associated with the Technical Specification changes discussed above, are included for information only. The TS Bases will be revised in accordance with the TS Bases Control Program, TS 5.5.13 following NRC approval of the license amendment.

The proposed changes qualify for categorical exclusion for 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

The completion time changes proposed in this license amendment request have been evaluated herein. The evaluation of the completion time changes is consistent with the Nuclear Regulatory Commission's approach for using probabilistic risk assessment in risk-informed decisions on plant-specific changes to the current licensing basis. This approach is discussed in 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," assessed 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. Tier 3, "Risk-Informed Plant Configuration Control and Management," assesses emerging plant conditions. Scheduling and performing maintenance and surveillance testing on these systems is evaluated and controlled in accordance with 10 CFR 50.65(a)(4), Maintenance Rule. As a consequence, use of the extended completion times will be minimized.

As discussed above, the proposed completion time changes are based on a risk-informed evaluation performed in accordance with RG 1.174 and RG 1.177. The CDF impact and the LERF impact, as well as the ICCDP and ICLERP associated with the proposed completion time changes, are presented in Tables 1 through 3. These values meet the acceptance criteria in RG 1.174 and RG 1.177 for the proposed changes. The cumulative CDF and LERF impact for the proposed completion time changes are presented in Tables 4 and 5 for North Anna Units 1 and 2. These values also meet the acceptance criteria in RG 1.174.

## 3.0 Need for Technical Specification Changes

The 72-hour completion time for the LHSI ECCS, QS, CAT, and AFW Systems creates an unnecessary burden for both emerging and preventive maintenance activities. Extending the completion time to 7 days will:

1) Reduce the need to shut down the plant to complete repair activities on these systems, which averts known risks from plant shutdown and startup evolutions. For example, a pipe replacement or major pump upgrade activity that could extend beyond the 72 hours,

2) Permit completion of additional planned maintenance activities at power that are difficult to perform within 3 days, which permits use of trained plant staff for maintenance instead of contractors. This will also reduce work load during refueling outages to allow focus on refueling tasks, and decreases the likelihood of maintenance errors,

3) Reduce or eliminate the need for preparing, reviewing and approving emergency/exigent Technical Specification changes or notices of enforcement discretion (NOED), which results in significant cost savings for plant staffs as well as the NRC. On July 23, 2004 Dominion submitted an Emergency Technical Specification Change that was approved by the NRC to extend the Completion Time to effect repairs on the "A" Low Head Safety Injection Pump.

## 4.0 Description of Proposed Changes

## 4.1 The proposed changes will revise the Technical Specifications as follows:

<u>TS 3.5.2 - ECCS-Operating</u>

1. A new Condition A is included to add an inoperable LHSI subsystem train with a completion time of 7 days.
2. Delete the note that permitted a one-time extension of the Completion Time for repair of the "A" LHSI system.

3. Adding a new Condition resulted in changes in Conditions B and C to account for the new Condition. These changes reflect the addition of the new Condition but do not revise the existing requirements or Completion Times of those conditions.

TS 3.6.6 - QS System

4. The completion time for Required Action A.1 is revised from 72 hours to 7 days for an inoperable QS train.

TS 3.6.8 - Chemical Addition System

5. The completion time for Required Action A.1 is revised from 72 hours to 7 days for an inoperable Chemical Addition System.

TS 3.7.5 - AFW System

6. The second condition of Condition A in TS 3.7.5, "One turbine driven AFW pump inoperable in Mode 3 following refueling," and the associated note are being deleted. The Required Action for A.1 is revised to "Restore steam supply to OPERABLE status."

7. The second completion time for Required Actions A.1 is revised from 10 days to 14 days, to include the 7-day completion time of Required Action B.1 of TS 3.7.5, AFW System.

8. The completion time for Required Action B.1 of TS 3.7.5, AFW System, is revised from 72 hours to 7 days.

9. The second completion time for Required Action B.1 is revised from 10 days to 14 days to incorporate the Required Action extension of the completion time of Required Action B.1 of TS 3.7.5, AFW System.

TS Bases

Changes, reflecting the proposed changes to the Conditions, Required Actions, completion times and the other associated changes are included for information only. The TS Bases will be revised in accordance with the TS 5.5.13, TS Bases Control Program, following NRC approval of the license amendment.

### 4.2 Basis for the Technical Specification Changes

The proposed completion time changes from 72 hours to 7 days for the systems are based on a risk-informed analysis performed in accordance with RG 1.174 and RG 1.177. The technical analysis is described in Section 5.0.

The 7-day Completion Time is only for the LHSI trains of the ECCS. Thus, a new Condition is being incorporated for the LHSI trains and the remaining conditions are being renumbered to accommodate this additional Condition. In addition, the Note in the Completion Time for the existing Condition A in TS 3.5.2 is being deleted. The one time July 21, 2004 7-day Completion Time extension for the repair of the "A" LHSI train has expired.

The separate Condition in TS 3.7.5 (Change 6 above) and associated completion time of 7 days for the turbine driven AFW pump in Mode 3 for Condition A is no longer needed due to the proposed completion time of 7 days in Required Action B.1 for an inoperable AFW train.

The change to the second completion times in Required Actions A.1 and B.1 for TS 3.7.5 (Change numbers 7 and 9 above) is necessary since the completion time for B.1 is revised to 7 days. This will increase the total time from discovery of failure to meet the LCO from 10 days to 14 days for Required Actions A.1 and B.1. The 7-day completion time for Required Action A.1 is not risk-informed, and the total time of 14 days is also not risk-informed. Only the new completion time for Required Action B.1 is risk-informed.

The Required Action A.1 of TS 3.7.5 is revised to address only the steam supply to the turbine drives AFW pump since the second condition in A is being deleted (Change 6 above).

## 4.3 System Descriptions

Emergency Core Cooling System

The ECCS consists of two separate subsystems: the high head safety injection (HHSI) subsystem and the low head safety injection (LHSI) subsystem. Each subsystem consists of two redundant, 100% capacity trains. System diagrams are shown on Figures 1a and 1b attached. The HHSI subsystem consists of three charging pumps providing normal charging and seal injection and safety injection to RCS cold and hot legs. The charging pump C is a swing pump that can be powered from either safety bus, but needs to be started manually. In addition, there is a unit-to-unit crosstie between the HHSI systems. For injection, these pumps take suction from the RWST. The LHSI system consists of two 100% capacity trains, with one LHSI pump per train, providing flow to the cold legs or hot legs. For injection, these pumps also take suction from the RWST.

During cold leg and hot leg recirculation phase, the LHSI pump suction is transferred to the containment sump. The LHSI pumps supply flow to the RCS hot and cold legs and the HHSI pumps suction during the recirculation phase.

Containment Spray Systems

The Containment spray system consists of two systems: the quench spray (QS) and the recirculation spray (RS) subsystems as shown in Figures 2a and 2b.

The QS subsystem consists of two separate and redundant trains of equal capacity, each capable of meeting the design basis. Each train includes a pump, spray headers, nozzles, valves, and piping. The RWST supplies water to the system during the injection phase of operation. The QS also provides flow to the suction of the Inside RS pumps to improve the NPSH available for the inside RS pumps.

During the recirculation mode of operation, the RS subsystems are used and the water supply is taken from the containment sumps. Heat is removed from the containment sump water by the recirculation spray heat exchangers. The RS system consists of two separate trains of equal capacity, each capable of meeting the design basis. Each train includes one RS subsystem outside containment and one RS subsystem inside containment. Each subsystem consists of one approximately 50% capacity pump, one spray cooler, and one 180° coverage spray header plus associated nozzles, valves, and piping. Two casing coolant pumps and the common casing cooling tank are designed to increase the net positive suction head (NPSH) available to the outside RS pumps. The casing cooling pumps are considered part of the outside RS subsystem.

## Chemical Addition System

The Chemical Addition System is a subsystem of the Quench Spray system. The Chemical Addition System consists of one chemical addition tank, two parallel redundant motor operated valves in the line between the chemical addition tank and the refueling water storage tank (RWST), instrumentation, and a recirculation pump. The sodium hydroxide (NaOH) solution in the chemical addition tank is added to the quench spray water by a balanced gravity feed from the chemical addition tank through the connecting piping into a weir within the RWST. There, it mixes with the borated water flowing to the quench spray pump suction. The parallel isolation valves eliminate the potential for a single active failure.

The Chemical Addition System supplies a NaOH solution into the quench spray. The NaOH added to the spray ensures an alkaline pH for the solution recirculated in the containment sump. The resulting alkaline pH of the recirculation spray enhances the ability of the recirculation 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. In a design basis loss-of-coolant accident (LOCA), the Containment cooling systems (Quench Spray and Recirculation Spray) are designed to return the Containment to sub-atmospheric conditions within 1 hour. Therefore, the potential for iodine leakage from the Containment following a DBA LOCA is limited to the first 60 minutes.

## Auxiliary Feedwater System

This AFW system consists of two motor-driven pumps and one steam turbine-driven pump configured into three trains shown in Figure 3. Each pump takes suction from the emergency Condensate Storage Tank and each pump normally provides AFW to one SG. The Turbine Driven pump receives steam from each of the three main steam lines.

## 5.0 Technical Analysis

## 5.1 Risk Assessment

This section presents the analysis and assumptions used to determine the impact on plant risk of the increased completion time (CT). This section addresses the three tiered approach in the evaluation of risk-informed Technical Specification changes. The three tiered approach is defined in Regulatory Guide 1.177. The first tier, discussed in Sections 5.1.1, addresses PSA insights and includes the risk analyses and sensitivity analyses to support the CT changes. The second tier, which addresses avoidance of risk-significant plant configurations, is addressed in 5.1.2. The third tier, which addresses risk-informed plant configuration control and management, is addressed in 5.1.3.

The North Anna WinNUPRA N0AA model used for the calculation was deemed suitable for use in this risk-informed application since it models the as-built and as-operated plant. The model has undergone a PRA Industry Peer review. A review of the Peer Review Findings and Observations (F&Os) was performed to ensure that none of the F&Os would invalidate the results of this evaluation. Enclosure 1 contains a matrix with the A and B significance level F&Os from the North Anna PRA Peer Review. The changes made to the North Anna PRA Model since it was developed for the Individual Plant Examination (IPE) are discussed in Enclosure 2.

The Chemical Addition System is not modeled in the North Anna PRA model due to its limited ability to impact the magnitude of a radioactive release from the Containment in severe accidents and the limited corrosion damage which might occur to equipment over the first 24 hours from a non-alkaline pH. In severe accidents, the iodine release is so large that the Chemical Addition System is assumed incapable of scavenging a significant portion of the iodine. Also, as long as the Containment integrity is maintained in a severe accident, studies have shown that the radioactive release from the Containment cannot cause a large early release as defined in Regulatory Guide 1.174. If the Containment fails in a severe accident, there is insufficient NaOH available in the Chemical Addition System to impact the consequences of the large iodine release. For these reasons, the Chemical Addition System was not modeled in the IPE for either Surry or North Anna, nor was it modeled in the NRC's NUREG-1150 risk study for Surry. North Anna's Chemical Addition System is similar to Surry's system.

The worst case DBA where the Chemical Addition System is credited for iodine removal is a large break LOCA. The likelihood of a large break LOCA (i.e., greater than 6 inch diameter based on the PRA model) is 4.5E-6 per year in the North Anna PRA. The probability of a large LOCA during a 7 day AOT of the Chemical Addition System is 8.6E-8 (i.e., 4.5E-6/yr /52 weeks/yr.), which is negligible.

When the Chemical Addition System is inoperable, the ability of the QS System flow to adjust sump water pH for enhanced iodine removal is eliminated. The Quench Spray and Recirculation Spray Systems would still be available to return the Containment to sub-atmospheric conditions within 1 hour and would remove some iodine from the containment atmosphere in the event of an accident. The 7-day completion time takes into account the ability of the spray systems to remove iodine at a reduced capability using the redundant Quench Spray flow path capabilities and the low probability of the worst case DBA occurring during this period.

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

In Tier 1, the impact of the CT change on core damage frequency (CDF), incremental conditional core damage probability (ICCDP), large early release frequency (LERF), and incremental conditional large early release probability (ICLERP) is determined. ICCDP and ICLERP are defined as:

1. ICCDP = [(conditional CDF with the subject equipment out of service) – (baseline CDF with nominal expected equipment unavailabilities)] X (duration of single AOT under consideration)

2. ICLERP = [(conditional LERF with the subject equipment out of service) – (baseline LERF with nominal expected equipment unavailabilities)] X (duration of single AOT under consideration)

LHSI ECCS Completion Time Change

The ECCS provides core cooling and negative reactivity to ensure that the reactor core is protected after the following events:

1. Loss of coolant accident, coolant leakage greater than the capability of the normal charging system

2. Rod ejection accident

3. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater

4. Steam generator tube rupture

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

There are three phases of ECCS operation, which include injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the RWST and injected into the reactor coolant system through the cold legs. When sufficient water is removed from the RWST to ensure 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. After an appropriate amount of time, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which will reduce the boiling in the top of the core and any resulting boron precipitation.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the LHSI pumps, the centrifugal charging 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, the low head pump suction is transferred to the containment sump. The low head pumps then typically supply the other ECCS pumps.

The high head subsystem also functions to supply borated water to the reactor core following increased heat removal events, such as a main steamline break.

This LCO provides assurance that in operating Modes 1, 2, and 3, sufficient ECCS flow is available, assuming a single failure affecting either train. In addition, individual components within the ECCS trains may be required to mitigate the consequences of other transients and accidents.

The North Anna PRA combines the test and maintenance unavailability into a single value. The impact of the CT change on a low head train is estimated as the ratio of the extended CT to the current CT (7 days/3 days = 2.33). Based on this, the following changes in the combined maintenance and test unavailability values are expected.

| **System** | **Maintenance Time With Current CT** | **Expected Maintenance Time With Extended CT** |
|---|---|---|
| LHSI | 26.1 hr/train/yr. | 60.8 hr/train/yr. |

The results for North Anna, in terms of the impact of the CT extension on CDF and LERF, in addition to the ICCDP and ICLERP values, are provided on Table 1. In calculating the ICCDP and ICLERP values, the analysis considered a train of low head ECCS out of service. In addition, the analyses differentiated between scheduled (or preventive) activities and repair (or unscheduled) activities. Evaluating both types of activities could lead to CT improvements for one type of activity and not the other. With a repair activity, the operable train could have the same problem as the inoperable train. Therefore, a higher failure probability is used for the operable train, which accounts for the possibility of common cause failure. With a scheduled activity there is no prior information concerning the out of service train with regard to failure, therefore, the random failure probability applies to the operable train.

The North Anna CDF and LERF values meet the Reg. Guide criteria for finding small increases in CDF and LERF acceptable. Only the internal event CDF and LERF values are provided on Table 1. External event impacts are address in Section 5.1.4.

The North Anna analysis results for the impact on CDF and LERF meet the Reg. Guide 1.174 guidelines for a small impact. The ICCDP and ICLERP were evaluated for an LHSI ECCS subsystem out of service. The results indicate that the ICCDP and ICLERP values for scheduled or repair activities meet the guidelines for a 7-day CT. Based on this, the PRA calculations support a LHSI train inoperable for 7 days.

Quench Spray Completion Time Change

The QS system provides containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a design basis accident, to within limits. 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 subatmospheric pressure in < 60 minutes following a DBA.

The QS system uses water from the RWST during the injection phase of operation to reduce pressure in and cool the containment. The recirculation spray subsystem is used to remove heat from the containment using water from the containment sumps. Heat is removed from the containment sump water by the recirculation spray system heater exchangers.

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.

This LCO provides assurance that in operating Modes 1, 2, 3, and 4 containment cooling is available to control containment peak pressure and temperature, and to remove iodine from the containment atmosphere.

The North Anna PRA combines the test and maintenance unavailability into a single value. The impact of the CT change on a quench spray train is estimated as the ratio of the extended CT to the current CT (7 days/3 days = 2.33). Based on this, the following changes in the combined maintenance and test unavailability values are expected.

| System | Maintenance Time With Current CT | Expected Maintenance Time With Extended CT |
|---|---|---|
| Quench Spray | 3.2 hr/train/yr. | 7.4 hr/train/yr. |

The results in terms of the impact of the CT extension on CDF and LERF, in addition to the ICCDP and ICLERP values, are provided on Table 2.

The North Anna CDF and LERF values meet the Reg. Guide criteria (CDF < 1E-04/yr and LERF < 1E-05/yr) for finding small increases in CDF and LERF acceptable. Only the internal events CDF and LERF are provided on Table 2. External event impacts are address in Section 5.1.4.

The North Anna analysis results for the impact on CDF and LERF meet Reg. Guide 1.174 guidelines for a very small impact, that is, the ΔCDF is less than 1E-06/yr and ΔLERF is less than 1E-07/yr. The ICCDP and ICLERP values for both scheduled maintenance and repair activities meet the acceptance guidelines in Reg. Guide 1.177, which are the ICCDP is less than 5E-07 and the ICLERP is less than 5E-08. Based on this, the PRA calculation supports a Quench spray train inoperable for 7 days.

Auxiliary Feedwater Completion Time Change

The AFW system automatically supplies feedwater to the steam generators (SGs) to remove decay heat from the reactor coolant system on loss of normal feedwater supply. The AFW pumps take suction through separate and independent suction lines from the emergency condensate storage tank (ECST) and pump to the SG secondary side. The SGs function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the SGs via the mainsteam safety valves or atmospheric dump valves (SG PORVs). When available, the main condenser is also used to dissipate the decay heat.

The AFW system consists of two motor driven (MD) pumps and one steam turbine driven (TD) pump configured into three trains. Each MD pump can provide 100% of the AFW flow required for accident mitigation and the TD pump can provides 200% of the required flow for accident mitigation. The TD pump receives steam from each main steamline and each steam supply provides 100% of the TD pump requirements. One of the three pumps is sufficient to remove decay heat and cool the unit to residual heat removal entry conditions.

The AFW system actuates automatically on SG water level low-low, loss of offsite power, safety injection, and trip of all main feedwater pumps. An AMSAC (ATWS mitigating system actuation circuitry) signal will also actuate the AFW system The AFW pumps can be manually started as well.

This LCO provides assurance that in operating Modes 1, 2, 3, and 4 (when a SG is relied on for heat removal) the AFW system will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary.

With the extended CTs, North Anna may complete additional test and scheduled maintenance activities while they are at-power or repair activities may now take longer to complete since round-the-clock repair efforts may be delayed. The impact of the extended CTs on the availability of the AFW system when at power was assessed. The following is a summary of these assessments.

The North Anna PRA combines the test and maintenance unavailability into a single value. The impact of the CT change on an AFW train is estimated as the ratio of the extended CT to the current CT (7 days/3 days = 2.33). Based on this, the following changes in the combined maintenance and test unavailability values are expected.

| System | Maintenance Time With Current CT | Expected Maintenance Time With Extended CT |
|---|---|---|
| AFW: MD Pumps | 12.3 hr/train/yr. | 28.6 hr/train/yr. |
| AFW: TD Pump | 25.2 hr/train/yr. | 58.8 hr/train/yr. |

The results for the NAPS analysis in terms of the impact of the CT extension on CDF and LERF, in addition to the ICCDP and ICLERP values, are provided on Table 3.

Extensions for both the MD and TD pumps were evaluated. The North Anna CDF and LERF values meet the Reg. Guide criteria for finding small increases in CDF and LERF acceptable. Only the internal event CDF and LERF values are provided on Table 3. External event impacts are address in Section 5.1.4.

The North Anna analysis results for the impact on CDF and LERF meet Reg. Guide 1.174 guidelines for a small impact. The ICCDP and ICLERP values for scheduled maintenance activities and repair activities on the MD and TD pump trains meet the acceptance guidelines in Reg. Guide 1.177.

The North Anna PRA model requires AFW for decay heat removal for all events except for large LOCA. Success criteria is 1 of 3 AFW pumps for all events requiring the AFW system for decay heat removal, except for 1) station blackout which requires 1 of 1 TD pump and 2) for small and medium LOCAs with HHSI failure which requires 2 of 3 AFW pumps. The frequency of events that require AFW is ~0.9/year. In addition to the AFW system, the following systems are credited in the PRA model for decay heat removal:

- recovery of main feedwater
- feed and bleed

The model does not credit depressurization of the SGs and then use of the condensate pumps.

Based on this, the PRA calculation supports either a MD AFW train or TD AFW train inoperable for 7 days.

Cumulative CDF and LERF

Reg. Guide 1.174 also requires the cumulative CDF and LERF impact to be considered. Table 4 provides the cumulative CDF impact and Table 5 provide the cumulative LERF impact for North Anna. The cumulative CDF value for North Anna is less than 1.0E-06/yr and the LERF cumulative value is less than 1.0E-07/yr, which meets the Reg. Guide acceptance criteria for "very small" increases in risk.

### 5.1.2 Tier 2: Avoidance of Risk-Significant Plant Configurations

Reasonable assurance must be provided that risk-significant plant equipment outage configurations will not occur when a system train is out of service consistent with the proposed technical specification change. This can be determined by comparing the basic event Risk Achievement Worth (RAW) importance data from the best estimate case (average annual maintenance), where the systems are available, to the best estimate case where a particular system is unavailable. When a component associated with a basic event RAW greater than 2 increases significantly (i.e., more than 10%), the component could potentially contribute to a Tier 2 configuration.

The comparison identified some plant configurations that should be avoided while the particular system train is unavailable during the extended AOT. These configurations are summarized in the Table 7.

In the case of the CAS, since this system is not modeled in the PRA and is not considered risk significant from PRA, a qualitative review was performed to determine if there were any other plant components whose simultaneous outage with the CAS would create a risk significant condition that would not have been present without the CAS outage. Due to the unique function of the CAS to scavenge iodine and control sump water pH, there were no other plant components identified which impact these functions or contribute to an increased need for these functions due to their outage. Therefore, no Tier 2 restrictions were identified for CAS outages.

### 5.1.3. Tier 3: Risk-Informed Plant Configuration Control and Management

North Anna Power Station's program for complying 10 CFR 50.65(a)(4) fully satisfies the guidance in Regulatory Guide 1.177 for Tier 3 Risk-Informed Configuration Risk Management. The North Anna 10 CFR 50.65(a)(4) program performs full model PRA analyses of all planned maintenance configurations at power in advance using the SCIENTECH Safety Monitor. The PRA model in the SCIENTECH Safety Monitor is a comprehensive, component level, core damage and large early release model. The North Anna risk-informed CRMP has been previously evaluated by the NRC in its review and approval of the following amendments: 1) 14-day allowed outage time for the emergency diesel generators (Amendment Nos. 214 and 195), 2) RPS/ESFAS analog instrument surveillance interval extension (Amendment Nos. 221 and 202), 3) 14-day allowed outage time for the PORV nitrogen accumulators (Amendment Nos. 214 and 199), and 4) 7-day allowed outage time for the instrument bus inverters (Amendment Nos. 235 and 217). Configurations that approach or exceed the NUMARC 93-01 risk limits (a 1.0E-6 cumulative increase in core damage probability) are avoided or addressed by compensatory measures per procedure. Historically, North Anna rarely approaches this limit. 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 10 CFR 50.65(a)(4).

The systems included in this licensee amendment request are explicitly included in the 10 CFR 50.65(a)(4) scope and their removal from service is monitored, analyzed and managed using the Safety Monitor tool. In addition, possible loss of offsite power hazards (grid loading/stability, switchyard or other electrical maintenance, external events such as severe weather) are all included in the Safety Monitor model and are explicitly accounted for in the (a)(4) program. When configuration risk approaches the (a)(4) risk limits, plant procedures direct the implementation of risk management actions in compliance with the regulations. If the configuration is planned, these steps must be taken in advance.

Individually, most system components do not approach the required risk management thresholds of the (a)(4) regulation. While combinations of unavailable equipment and/or evolutions, may approach the limits and even require risk management actions, the risks arising from these configurations will be managed in accordance with station procedures.

### 5.1.4 External Events

The internal events analysis used for the quantification of the risk impact of the proposed completion time changes includes internal initiating events and internal flooding. Qualitative assessments were performed for the risk impact of the proposed completion time changes on seismic, fire, floods and other external events evaluated in the Individual Plant Examination of External Events (IPEEE). The external event analyses have not been updated since completion of the IPEEE, and portions of these analyses were deterministic.

The seismic analysis in the IPEEE used the seismic margins method, which is entirely deterministic. The high confidence of low probability of failure (HCLPF) capacity of the plant was determined to be 0.16g, and was dominated by the overturning moment of large tanks.

The internal fire analysis in the IPEEE used the EPRI FIVE methodology with quantification of the unscreened fire areas. The core damage frequency from internal fires reported in the IPEEE was 4E-6 per year, which is a small fraction of the internal events core damage frequency.

The other events, including high winds, floods, transportation and nearby facility accidents analyses used a screening methodology with quantification of potentially significant events. The only aspect of the other events quantified was the nearby facility accidents analysis. The nearby facility accidents analysis resulted in core damage frequency of 4E-8 per year, which a very small fraction of the internal events core damage frequency.

Table 8 provides a summary of the qualitative assessments of the external event analyses for each requested completion time change.

### 5.1.6 Cumulative CDF and LERF Impact

The previously approved and proposed risk-informed changes at North Anna with their associated estimated increases in core damage risk are provided in Table 8.

The cumulative estimated increases in risk associated with all the approved and proposed risk-informed changes is less than 2.1E-06 per year for CDF and 1.7E-07 per year for LERF. These increases in risk are considered acceptably "small" per Regulatory Guide 1.174.

### 5.1.6 PRA Model

The PRA model utilized for the evaluation of the completion time changes is applicable to both Units 1 and 2, and the model reflects the as-built, as-operated plant. Furthermore, a program exists to periodically update the internal events PRA model in accordance with the Industry Peer Review guidance in NEI 00-02. Enclosure 1 provides a summary of the Findings and Observations from the North Anna (and applicable Surry) industry peer reviews and how this application is impacted by those peer review comments.

## 5.2 Defense-In-Depth Assessment

The proposed changes to the LHSI ECCS, QS including the CAS, and AFW System completion times maintain the system redundancy, independence, and diversity commensurate with the expected challenges to these systems operation. There are no proposed changes to the design or operation of the affected systems. The Work Management Program, Maintenance Rule (a)(4) Program and Corrective Action Program provide additional controls and assessments to preclude the possibility of simultaneous outages of redundant trains and ensure system reliability. The proposed increase in the completion time for the LHSI ECCS, AFW, and QS/CAS System will not alter the assumptions relative to the causes or mitigation of an accident. The risk impacts of the changes are also consistent with the acceptance criteria in RG 1.174 and RG 1.177. Therefore, there are no defense-in-depth impacts from the proposed change.

The defense-in-depth philosophy is maintained since:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation based on the low risk impacts.
- There are no weaknesses in plant design associated with the affected systems.
- System redundancy, independence, and diversity are unaffected.
- The defenses against potential common cause failures are unaffected and the potential for the introduction of new common cause failure mechanisms is not impacted by the change.
- The independence of fission product barriers is not affected.
- The defenses against human errors are not affected.
- The General Design Criteria in Appendix A to 10 CFR Part 50 are maintained.

## 5.3 Safety Margin Assessment

The overall margin of safety is not decreased due to the increased completion times for the LHSI ECCS, QS including the CAS, and AFW since the systems design and operation are not altered by the proposed increase in completion times. The risk impacts of the changes are also consistent with the acceptance criteria in RG 1.174 and RG 1.177.

For the Chemical Addition System, which is not modeled in the PRA due to its limited capability to mitigate severe accidents, the proposed completion time change takes into account the ability of the spray systems to remove iodine at a reduced capability and the low probability of the worst case DBA occurring during this period.

For each change, the following safety margin attributes from RG 1.174 were reviewed to ensure no change in safety margin:

- Codes and standards or their alternatives approved for use by the NRC are met.
- Safety analysis acceptance criteria in the licensing basis (e.g., FSAR, supporting analyses) are met.

### 5.4 Summary

The proposed completion time changes are based on a risk-informed evaluation performed in accordance with RG 1.174 and RG 1.177. The CDF impact and the LERF impact, as well as the ICCDP and ICLERP associated with the proposed completion time changes meet the acceptance criteria in RG 1.174 and RG 1.177 for the proposed changes. The cumulative CDF and LERF impact for the proposed completion time changes also meet the acceptance criteria in RG 1.174 for the proposed changes. The defense-in-depth and safety margin are not impacted by the proposed changes.

## 6.0 Regulatory Safety Analysis

### 6.1 No Significant Hazards Consideration

The proposed changes will revise the completion times for the LHSI Emergency Core Cooling System (ECCS); Quench Spray (QS) System including Chemical Addition System (CAS), and Auxiliary Feedwater (AFW) System. The proposed changes are 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 Decisionmaking: Technical Specifications." Dominion has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

   The proposed changes do not alter any plant equipment or operating practices in such a manner that the probability of an accident is increased. The proposed changes will not alter assumptions relative to the mitigation of an accident or transient event.

   The CDF impact and the LERF impact, as well as the ICCDP and ICLERP, associated with the proposed completion time changes meet the acceptance criteria in RG 1.174 and RG 1.177 for the proposed changes. The cumulative CDF and LERF impact for the proposed completion time changes also meet the acceptance criteria in RG 1.174 for the proposed changes.

Therefore, the proposed changes do 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 changes do 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 changes do 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 overall margin of safety is not decreased due to the increased completion times for the LHSI ECCS, QS including the CAS, and AFW since the systems design and operation are not altered by the proposed increase in completion times. The risk impacts of the changes are also consistent with the acceptance criteria in RG 1.174 and RG 1.177.

   For the Chemical Addition System, which is not modeled in the PRA due to its limited capability to mitigate severe accidents, the proposed completion time change takes into account the ability of the spray systems to remove iodine at a reduced capability and the low probability of the worst case DBA occurring during this period.

   The codes and standards or their alternatives approved for use by the NRC continue to be met. In addition, the safety analysis acceptance criteria in the licensing basis (e.g., FSAR, supporting analyses) continue to be met.

   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 present 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.

### 6.2 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 changes meet 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.

## 7.0 Conclusion

The proposed changes will revise the Completion Times for the Emergency Core Cooling System Low Pressure Subsystem, Quench Spray, Chemical Addition, and Auxiliary Feedwater Systems. The risk-informed evaluation concludes that the increase in annual core damage and large early release frequencies associated with the proposed change are less than 6E-7 and less than 3E-8, 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 proposed changes will provide greater flexibility for the repair or replacement of the associated system components without having to shut down the plant if the activities take longer than the current completion times. These changes should also reduce the impact on refueling outage duration by permitting completion of additional planned maintenance activities at power that are difficult to perform within 3 days (e.g., pump rotating assemblies and mechanical seal replacements), which permits use of trained plant staff for maintenance instead of contractors, reduces work load during refueling outages to allow focus on refueling tasks, and decreases the likelihood of maintenance errors. Furthermore, the proposed extended completion times would eliminate the administrative burden of requesting an emergency or exigent Technical Specification change or notice of enforcement discretion for most repair activities.

The Station Nuclear Safety and Operating Committee (SNSOC) and the Management Safety Review Committee (MSRC) have reviewed these proposed changes to the Technical Specifications and have concluded that it does not involve a significant hazards consideration and will not endanger the health and safety of the public.

| Table 1: Low Head ECCS Completion Time Change | |
|---|---|
| **Parameter** | **Value** |
| Low head ECCS pump fail to start failure probability (per demand) | 3.13E-03 |
| Low head ECCS pump fail to run failure probability (per hour) | 2.95E-05 |
| Small LOCA ECCS required mission (run) time | |
| - Injection (hours) | 24 |
| - Cold leg recirculation (hours) | 24 |
| - Hot leg recirculation (hours) | Not modeled |
| Medium LOCA ECCS required mission (run) time | |
| - Injection (hours) | 24 |
| - Cold leg recirculation (hours) | 24 |
| - Hot leg recirculation (hours) | Not modeled |
| Large LOCA ECCS required mission (run) time | |
| - Injection (hours) | 1 |
| - Cold leg recirculation (hours) | 24 |
| - Hot leg recirculation (hours) | 24 |
| ECCS common cause failure model (MGL?, ALPHA?, etc.) | Alpha |
| CDF (current CT) (per yr) | 1.06E-05 |
| CDF (proposed CT) (per yr)[1] | 1.10E-05 |
| CDF increase (per yr)[1] | <4.0E-07 |
| LERF (current CT) (per yr) | 1.38E-06 |
| LERF (proposed CT) (per yr)[1] | 1.39E-06 |
| LERF increase (per yr)[1] | <1.0E-08 |
| CCDF (with one LHSI pump train out of service due to test or scheduled maintenance activity) (per year) | 2.06E-05 |
| CCDF (with one LHSI pump train out of service due to corrective/repair maintenance activity) (per year) | 3.22E-05 |
| ICCDP (for the 7 day AOT with one LHSI pump train out of service due to test or scheduled maintenance activity) | 1.92E-07 |
| ICCDP (for the 7 day AOT with one LHSI pump train out of service due to corrective/repair maintenance activity) | 4.14E-07 |
| CLERF (with one LHSI pump train out of service due to test or scheduled maintenance activity) (per year) | 1.52E-06 |
| CLERF (with one LHSI pump train out of service due to corrective/repair maintenance activity) (per year) | 1.53E-06 |
| ICLERP (for the 7 day AOT with one LHSI pump train out of service due to test or scheduled maintenance activity) | 2.68E-09 |
| ICLERP (for the 7 day AOT with one LHSI pump train out of service due to corrective/repair maintenance activity) | 2.89E-09 |

[1] CDF and LERF values are conservatively based on expected unavailability increase at the ECCS level, that is, if the CT were extended for all the ECCS subsystems (i.e., high head and low head safety injection).

| Table 2: Quench Spray Completion Time Change | |
|---|---|
| **Parameter** | **Value** |
| Quench spray pump fail to start failure probability (per demand) | 3.93E-03 |
| Quench spray pump fail to run failure probability (per hour) | 3.30E-05 |
| Quench spray system required mission (run) time | --- |
| - Injection (hours) | 2 |
| Quench spray system common cause failure model (MGL?, Alpha?, etc.) | Alpha |
| CDF (current CT) (per year) | 1.06E-05 |
| CDF (proposed CT) (per year) | 1.06E-05 |
| CDF increase (per year) | < 1.00E-07 |
| CCDF (one spray train out of service due to test or scheduled maintenance activity) (per yr) | 1.06E-05 |
| CCDF (one spray train out of service due to repair activity) (per yr) | 1.06E-05 |
| ICCDP (7 day CT with one spray train out of service due to test or scheduled maintenance activity) | < 1.00E-08 |
| ICCDP (7 day CT with one spray train out of service due to repair activity) | < 1.00E-08 |
| LERF (current CT) (per year) | 1.38E-06 |
| LERF (proposed CT) (per year) | 1.38E-06 |
| LERF increase (per year) | < 1.0E-08 |
| CLERF (one spray train out of service due to test or scheduled maintenance activity) (per yr) | 1.40E-06 |
| CLERF (one spray train out of service due to repair activity) (per yr) | 1.41E-06 |
| ICLERP (7 day CT with one spray train out of service due to test or scheduled maintenance activity) | 3.83E-10 |
| ICLERP (7 day CT with one spray train out of service due to repair activity) | 5.75E-10 |

| Table 3: AFW Completion Time Change | |
|---|---|
| **Parameter** | **Value** |
| AFW MD pump fail to start failure probability (per demand) | 1.39E-03 |
| AFW MD pump fail to run failure probability (per hour) | 2.95E-05 |
| AFW TD pump fail to start failure probability (per demand) | 1.53E-02 |
| AFW TD pump fail to run failure probability (per hour) | 2.68E-03 |
| AFW system required mission (run) time (hours) | 24 |
| AFW system common cause failure model (MGL?, Alpha?, etc.) | Alpha |
| CDF (current CT) (per yr) | 1.06E-05 |
| CDF (proposed CT) (per yr) | 1.06E-05 |
| CDF increase (per yr) | < 1.00E-07 |
| CCDF (one MD pump AFW train out of service due to test or scheduled maintenance activity) (per yr) | 1.83E-05 |
| CCDF (one MD pump AFW train out of service due to repair activity) (per yr) | 2.95E-05 |
| CCDF (one TD pump AFW train out of service due to test or scheduled maintenance activity) | 1.49E-05 |
| CCDF (one TD pump AFW train out of service due to repair activity) (per yr) | 1.87E-05 |
| ICCDP (one MD pump AFW train out of service due to test or scheduled maintenance activity) | 1.48E-07 |
| ICCDP (one MD pump AFW train out of service due to repair activity) | 3.62E-07 |
| ICCDP (one TD pump AFW train out of service due to test or scheduled maintenance activity) | 8.25E-08 |
| ICCDP (with one TD pump AFW train out of service due to repair activity) | 1.55E-07 |
| LERF (current CT) (per yr) | 1.38E-06 |
| LERF (proposed CT) (per yr) | 1.38E-06 |
| LERF increase (per year) | < 1.00E-08 |
| CLERF (one MD pump AFW train out of service due to test or maintenance activity) (per yr) | 1.50E-06 |
| CLERF (one MD pump AFW train out of service due to repair activity) (per yr) | 1.62E-06 |
| CLERF (one TD pump AFW train out of service due to test or scheduled maintenance) (per yr) | 1.52E-06 |
| CLERF (one TD pump AFW train out of service due to repair activity)(per yr) | 1.48E-06 |
| ICLERP (one MD pump AFW train out of service due to test or maintenance activity) | 2.30E-09 |
| ICLERP (one MD pump AFW train out of service due to repair activity) | 4.60E-09 |
| ICLERP (one TD pump AFW train out of service due to test or scheduled maintenance) | 1.92E-09 |
| ICLERP (one TD pump AFW train out of service for repair activities) | 2.68E-09 |

| Table 4: Summary of CDF Increase for each CT Change | |
|---|---|
| **CT Change** | **Delta CDF (per year)** |
| Emergency Core Cooling System | <4.0E-07 |
| Quench spray System | <1.0E-07 |
| Auxiliary Feedwater System | <1.0E-07 |
| Total | <6.0E-07 |

| Table 5: Summary of LERF Increase for each CT Change | |
|---|---|
| **CT Change** | **Delta LERF (per year)** |
| Emergency Core Cooling System | <1.0E-08 |
| Quench Spray System | <1.0E-08 |
| Auxiliary Feedwater System | <1.0E-08 |
| Total | <3.0E-08 |

| Table 6: Summary of Justified Completion Times | |
|---|---|
| **System** | **Justified Completion Time** |
| ECCS: LHSI System, Scheduled Activity | 7 days |
| ECCS: LHSI System, Repair Activity | 7 days |
| Quench Spray: Scheduled Activity | 7 days |
| Quench Spray: Repair Activity | 7 days |
| AFW: MD Pump Train, Scheduled Activity | 7 days |
| AFW: MD Pump Train, Repair Activity | 7 days |
| AFW: TD Pump Train, Scheduled Activity | 7 days |
| AFW: TD Pump Train, Repair Activity | 7 days |

| Table 7: Planned Maintenance Tier 2 Restrictions | |
|---|---|
| **System Train OOS** | **Restricted Components** |
| One AFW Pump OOS | • Alternate AC Diesel<br>• Unit 1(2) EDGs (see note 1)<br>• 1(2)- CH-P-1A/B/C (Charging Pumps)<br>• Charging Pump (CH) Crosstie (or all CH pumps on opposite unit)<br>• 1(2)-RC-PCV-1(2)455C/1(2)456 (pressurizer power operated relief valves)<br>• 1(2)-SI-P-1A/B (low head safety injection pumps) |
| One LHSI Pump OOS | None |
| One Containment Spray Pump OOS | None |

Note: 1. Only the EDGs on the same unit as the AFW pump. Likewise for the other restricted components.

| Table 8 – Cumulative Impact of Risk-Informed Changes | | |
|---|---|---|
| **North Anna Risk-Informed Change** | **Estimated increase in CDF per year** | **Estimated increase in LERF per year** |
| | | |
| Approved 14 day emergency diesel generator allowed outage time extension | 1.3E-06 | *1.3E-07 |
| Approved 7 day inverter allowed outage time extension | 8.1E-08 | 4.6E-10 |
| Approved reactor protection system and engineered safety features actuation system analog channel surveillance test internal extensions from monthly to quarterly and allowed outage time extensions | 3E-09 | 3E-10 |
| Proposed 7 day emergency core cooling system low pressure subsystem allowed outage time extension | <4.0E-07 | <1.0E-08 |
| Proposed 7 day quench spray system allowed outage time extension | <1E-07 | <1E-08 |
| Proposed 7 day chemical addition system allowed outage time extension | <1E-07 | <1E-08 |
| Proposed 7 day auxiliary feedwater system allowed outage time extension | <1E-07 | <1E-08 |
| **Cumulative Total** | <2.1E-06 | <1.7E-07 |

*LERF was not calculated, but was estimated based on generic 0.1 containment failure probability for large, dry PWRs.

| Table 9- External Event Assessment | |
|---|---|
| **Completion Time Change - External Event Analysis** | **Qualitative Assessment** |
| Emergency Core Cooling System (ECCS) | |
| Internal Fire | ECCS was not associated with any vulnerabilities or unique significance in fire events. |
| Seismic | ECCS is seismically qualified and was not associated with any vulnerabilities or unique significance in seismic events. |
| High Winds, Floods, Transportation and Nearby Facility Accidents | ECCS was not associated with any vulnerabilities or unique significance in these events |
| | |
| Quench Spray System | |
| Internal Fire | QS was not associated with any vulnerabilities or unique significance in fire events. QS is only needed in medium to large LOCA events. Fires are unlikely to lead to medium to large LOCA events |
| Seismic | QS is seismically qualified and was not associated with any vulnerabilities or unique significance in seismic events. QS is only needed in medium to large LOCA events. Seismic events are unlikely to lead to medium to large LOCA events due to the high fragility of the RCS pressure boundary. |
| High Winds, Floods, Transportation and Nearby Facility Accidents | QS was not associated with any vulnerabilities or unique significance in these events. QS is only needed in medium to large LOCA events. These other events are unlikely to lead to medium to large LOCA events |
| | |
| Chemical Addition System | |
| Internal Fire | CAS was not associated with any vulnerabilities or unique significance in fire events. The system is not modeled in the NAPS PRA. |
| Seismic | CAS is seismically qualified and was not associated with any vulnerabilities or unique significance in seismic events. The system is not modeled in the NAPS PRA. |
| High Winds, Floods, Transportation and Nearby Facility Accidents | CAS was not associated with any vulnerabilities or unique significance in these events. The system is not modeled in the NAPS PRA. |
| | |
| Auxiliary Feedwater (AFW) | |
| Internal Fire | AFW was not associated with any vulnerabilities or unique significance in fire events. |
| Seismic | AFW is seismically qualified and was not associated with any vulnerabilities or unique significance in seismic events. |
| High Winds, Floods, Transportation and Nearby Facility Accidents | AFW was not associated with any vulnerabilities or unique significance in these events |

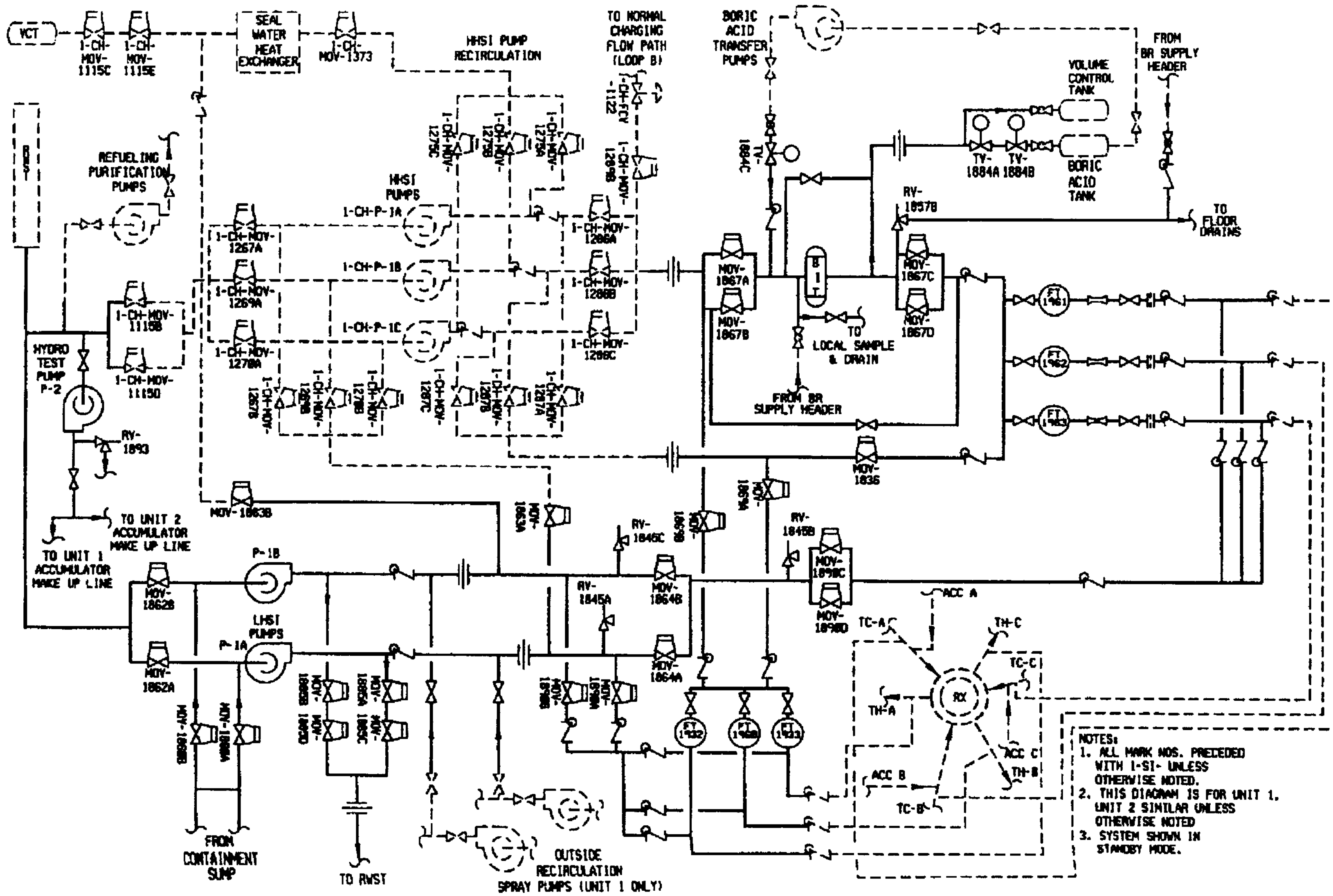


**Figure 1a ECCS Safety Injection System**

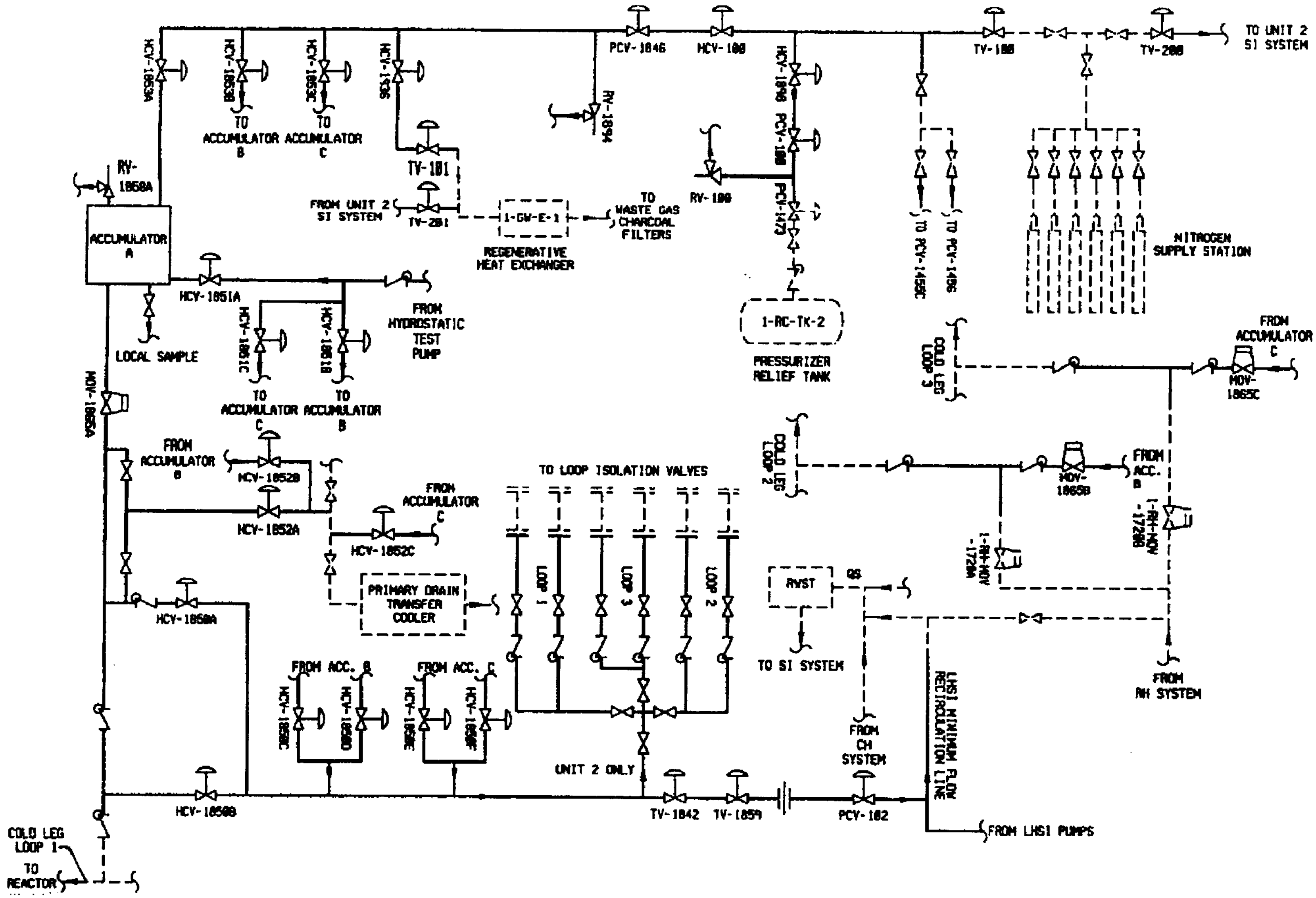

Figure 1b ECCS Safety Injection System

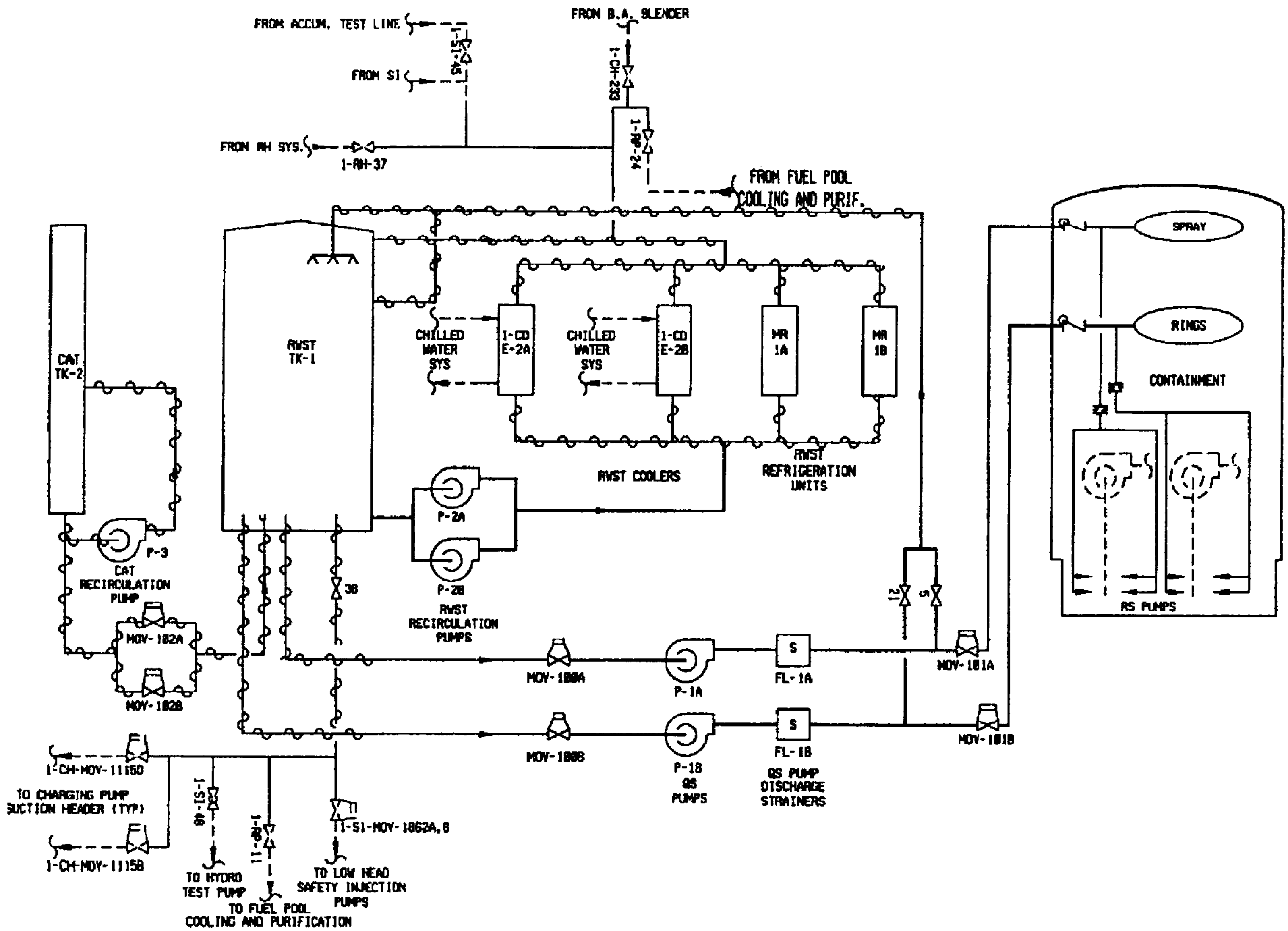


Figure 2a Quench Spray System

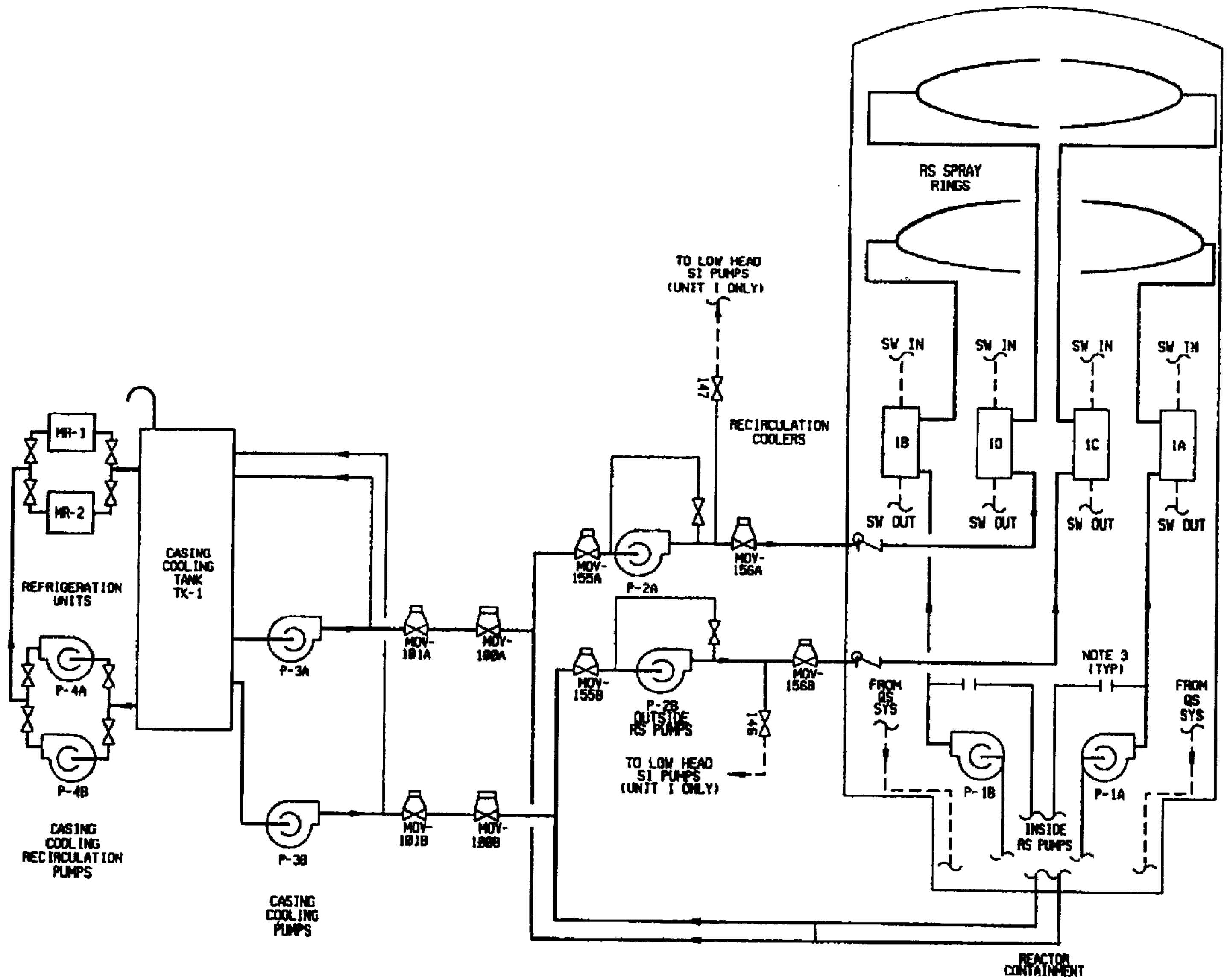

Figure 2b Recirculation Spray System

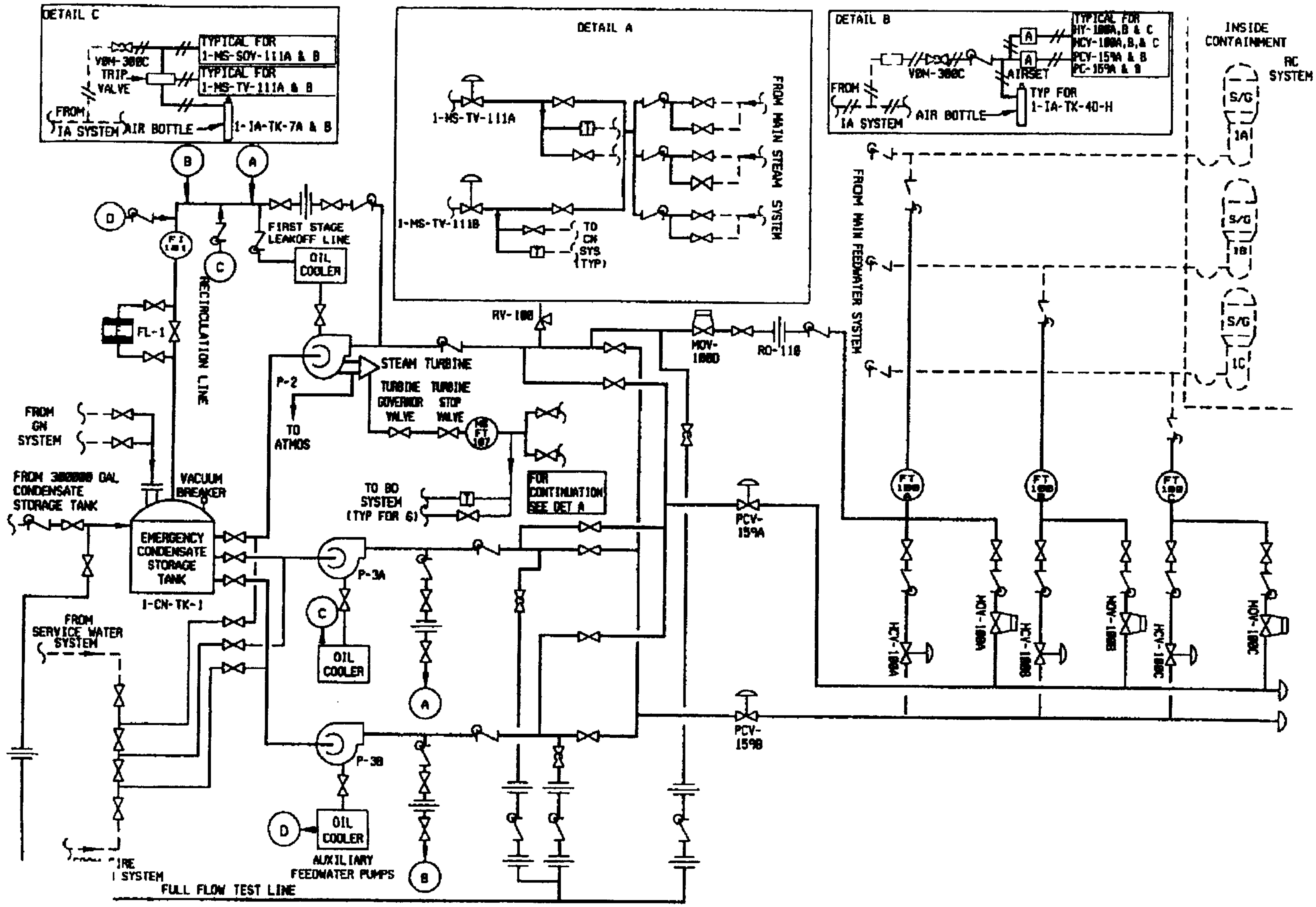


Figure 3 AFW System

# Enclosure 1

## Virginia Electric and Power Company
## North Anna Power Station Units 1 and 2
## Proposed Technical Specification Changes For
## LHSI ECCS, QS and AFW Systems Completion Times

## Industry Peer Comments on North Anna PRA Model and Impacts from Proposed Changes

**North Anna Power Station**
**Units 1 and 2**
**Virginia Electric and Power Company**
**(Dominion)**

North Anna PRA Peer Assessment A & B Level F&O Review Summary

The following matrix contains the A and B significance level F&Os from the North Anna PRA Peer Assessment

| Element | F/O | Level of Significance | Description | Impact on Application |
|---|---|---|---|---|
| AS – Accident Sequence Dev | AS-01/ AS-10 | B | Containment vulnerability following LOCAs is overly conservative (i.e., core damage assumed if containment integrity lost) | None: Addressed by recent update. |
| | AS-02 | B | LOCA event trees do not have a loss of emergency switchgear cooling (HVAC) top event | None: Do not concur with peer review significance of observation. Concurrent loss of emergency switchgear room cooling function and LOCA is a very low likelihood event based on the lack of common cause contributors and the fact that loss of emergency switchgear room cooling can not be a hidden standby system failure. Further, due to redundancy in emergency switchgear room cooling and the slow heatup, which results following cooling failures, there is adequate time to respond to such failures before loss of emergency systems needed for LOCA mitigation. |
| | AS-06 | B | Expand dependency matrix to plant dependencies for IE's and systems | None: Modeling of dependencies for the affected systems in this application are detailed and well documented. |

| Element | F/O | Level of Significance | Description | Impact on Application |
|---|---|---|---|---|
| | AS-08 | B | Address items for ATWS model | None: ATWS is a small contributor to the overall risk and the recommended changes to the ATWS model would not lead to ATWS becoming a dominant contributor. The observations on the ATWS model pertained primarily to the pressurizer PORVs, which are not included in the systems affected by this application. |
| | AS-09 | B | Enhance documentation of accident sequence development to better characterize the interface with IE's and EOP's | None: Documentation issues; does not impact modeling of the affected systems in this application. |
| | AS-12/ DA-15 | B | Switch to use a 24-hour mission time instead of 6 hours. | None: Applies only to emergency diesel generator mission time. Emergency diesel generators are not included in the systems affected by this application. |
| DA – Data Analysis | DA-04 | B | Justify using data collection dates of 1/1/97 – 12/31/1999 | None: Use of different data collection periods for reliability and unavailability data has minimal impact on the results. The plant specific data collection periods are recent enough to ensure the data matches the current plant operation and design. |
| | DA-08 | B | Provide appropriate documentation of equipment boundary and population definition for data and CCF update | None: This observation is limited to documentation issues associated with equipment boundaries. No errors were discovered in the data analysis related to equipment boundaries. |

| Element | F/O | Level of Significance | Description | Impact on Application |
|---|---|---|---|---|
| | DA-09 | B | Complete plant specific data update. | None: Addressed by recent update. |
| | DA-12 | B | Provide additional CCFs for support systems. | None: Potentially risk significant CCFs were incorporated in the recent update for all the affected systems in this application. |
| | DA-13 | B | Re-evaluate CCF screening criteria. | None: Potentially risk significant CCFs were incorporated in the recent update for all the affected systems in this application. |
| DE - Dependency | DE-01/ DE-02 | B | Minimum volume in the aux building internal flooding analysis appears inconsistent. | None: Internal flooding results do not dominate the risk significance of the affected systems in this application. |
| | DE-03 | B | Screening out of turbine building for flooding doesn't make sense. | None: Internal flooding results do not dominate the risk significance of the affected systems in this application. |
| | DE-04 | B | Unit 2 CH & CC crosstie was not included in the flood analysis. | None: Internal flooding results do not dominate the risk significance of the affected systems in this application. |
| HR – Human Reliability | HR-01 | A | Review HEP dependencies and provide documentation of results. | None: Addressed by recent update. |
| | HR-02 | A | Review REC screening values and verify appropriateness of leaving them at 0.1. | None: Addressed by recent update. |

| Element | F/O | Level of Significance | Description | Impact on Application |
|---|---|---|---|---|
| | HR-03 | B | The HRA approach provides a thorough analysis of time but there is little or no evidence of other performance shaping factors. | None: Do not concur with the significance of this observation. The other performance shaping factors are not as important in determining the failure probability. The HRA results have been subject to significant review in past comparisons to NUREG-1150 study for Surry. |
| | HR-05 | B | No evidence that the current HRA, including non-updated and updated HEPs, has been reviewed recently by operations and/or training personnel. | None: Do not concur with the significance of this observation. HRA has been subject to significant review in past comparisons to NUREG-1150 study for Surry. |
| | HR-06/ HR-11 | B | Improve the guidance for HRA. | None: Documentation issue. No technical issues identified which would impact importance of affected systems in this application. |
| | HR-08 | B | Review event trees to identify human actions that need to be modeled. | None: Documentation issue. No technical issues identified which would impact importance of affected systems in this application. |
| | HR-09 | B | No systematic review of indications performed or documented for HEPs. | None: Documentation issue. No technical issues identified which would impact importance of affected systems in this application. |
| | HR-10 | B | Treatment of operator actions for dual unit system support is questionable in some cases. | None: None of the dual unit support actions impacts the affected systems in this application. |

| Element | F/O | Level of Significance | Description | Impact on Application |
|---|---|---|---|---|
| IE – Initiating Events | IE-04 | B | Include loss of IA as a specific IE. | None: The primary impact of modeling loss of IA as a specific IE would be increased importance of the pressurizer and steam generator PORVs, which are not included in systems affected by this application. |
| | IE-07 | B | Either include additional IE's (MSLB, FWLB, RCS PORV, SRV) in the model or provide rationale for not including. | None: Addressed by recent update. |
| L2, Containment Performance Analysis | L2-02 | B | Update LERF early containment failure model | None: Current LERF model is conservative. |
| | L2-03 | B | Update LERF PRA to include EOP & SAMG actions | None: Current LERF model is conservative. |
| | L2-04 | B | Provide LERF definition and consistent LERF assignment | None: Current LERF model is conservative. The significance of the observation is mitigated by the reasonableness of the assignments using NUREG/CR-6595 and the WOG LERF definitions. |
| | L2-06 | B | No LERF documentation | None: Documentation issue. Surry documentation was used as surrogate and is applicable to NAPS due to the design similarities. |
| | L2-09 | B | All SGTR sequences should not result in LERF | None: Current LERF model is conservative. |

| Element | F/O | Level of Significance | Description | Impact on Application |
|---|---|---|---|---|
| | L2-10 | B | Revise bypass screening criteria | None: Affected systems in this application do not impact interfacing system LOCA analysis. |
| MU, Maintenance & Update | MU-01 | B | Provide enough time & resources to improve Independent Review quality | None: Addressed during recent update. |
| | MU-02 | B | Several AFW components risk significant at other plants are not in final cut set | None: Major AFW components, including the pump are in the cutsets and their risk significance is similar to other PWRs. The absence of individual AFW components or backup systems such as firewater from the cutsets does not significantly impact the risk significance of the affected systems in this application. |
| | MU-04 | B | Maintenance and update procedures may not be sufficient or adequate | None: Observation did not identify any specific areas of the maintenance or update procedures, which were inadequate. The observation was based on the large number of other F&Os, which has subsequently been determined to be unrelated to the maintenance and update procedures. |
| QU, Quantification | QU-02 | B | Key limitations missing from quantification documentation | None: Generic data observation addressed during recent model update. |

| Element | F/O | Level of Significance | Description | Impact on Application |
|---|---|---|---|---|
| | QU-03 | B | PORV logic gate errors in FB1 | None: Subsequent review of the feed and bleed fault tree indicates that the existing logic is correct. No change is required. |
| | QU-04 | A | Concern with 3rd highest cut set | None: The numerous observations have either a minimal impact of risk or result in an over-estimation of the risk. |
| | QU-07 | B | Evaluate manual recovery of MS PORVs in SGTR | None: Minor conservatism in the model due to not modeling recovery. |
| SY, Systems Analysis | SY-01 | B | Fails to Run CCF mission time is not applied correctly | None: Addressed by recent update. |
| | SY-02 | B | AFW pump automatic actuation failure w/manual restart not modeled | None: Failure to include manual start of AFW pumps (upon failure of automatic actuation) is a conservatism. |
| | SY-09 | B | HHSI pump restart is not modeled following LOSP | None: Failure of a pump to restart after a LOSP is a small contributor and does not significantly impact affected system risk significance. |
| | SY-12 | B | Replacement Steam Generators not evaluated | None: Impact of steam generator replacement is insignificant in terms of affected system success criteria since the changes to SG overall size and heat transfer capacity were relatively minor. The major impacts of SG replacement are on the timing associated with HRA probabilities, which are expected to be minimal. |

| Element | F/O | Level of Significance | Description | Impact on Application |
|---|---|---|---|---|
| | SY-14 | B | Incorporate flood scenarios into internal events model | None: There are no dependences between increases in test and maintenance unavailability for the affected systems in this application and the flooding model. |
| | SY-15 | B | CCF models missing for CH-MOV-111B/D and C/E | None: This observation was incorrect. There are CCFs for these MOVs in the FB4 tree. |
| | SY-19 | B | SG PORV capability w/o IA needs additional manual recovery past 5 cycles | None: SG PORVs are not part of any of the affected systems in this application. |
| TH, Thermal Hydraulic Analysis | TH-04 | B | MAAP3B not sufficiently detailed to evaluate peak clad temperature success criteria | None: Only impact of using MAAP3B core damage criteria is the quantification of a few HEP analyses. Systems affected by this application do not use success criteria based on MAAP core damage definition. |
| | TH-09 | B | Uncertain about SBO evaluation of SG overfill on TDAFW pump at 10.4 hrs | None: Uncertainty in time to possible SG overfill and subsequent failure of TDAFW pump is not as important as the peer review indicates. The difference in a few hours is not critical to the results of the PRA or the importance of the affected systems in this application. |

**Enclosure 2**

**Virginia Electric and Power Company**
**North Anna Power Station Units 1 and 2**
**Proposed Technical Specification Changes For**
**LHSI ECCS, QS, and AFW Completion Times**

**Changes to North Anna PRA Model since IPE**

**North Anna Power Station**
**Units 1 and 2**
**Virginia Electric and Power Company**
**(Dominion)**

**North Anna Nuclear Station PRA Model Changes**

The North Anna PRA model has undergone numerous major updates since it was developed for the Individual Plant Examination (IPE). In general, due to the similarities between the North Anna and Surry Power Station designs, the PRA models for each station are very similar and changes to each station's PRA are evaluated for applicability to the other station. All A and B Level Findings and Observations from the Westinghouse Owners Group Peer Review of the North Anna PRA model were either addressed or determined not to impact this application. A listing of significant model changes incorporated since the IPE includes:

1. Updated plant specific initiating event, failure rate, and unavailability data.

2. The small break LOCA initiating event was updated to include additional small LOCA categories.

3. Internal flooding initiators were updated and added to the living PRA model.

4. The human reliability analysis was updated.

5. The steam generator tube rupture fault trees were revised to improve the modeling of the check valves in the steam generator lines to the decay heat release valve.

6. Numerous additional common cause failure basic events were added including failure combinations of standby and running components (e.g., component cooling water pumps).

7. The component cooling fault trees were revised to add logic for recovery of component cooling to the RCP thermal barriers.

8. The loss of service water event tree was revised to incorporate the unavailability of the service water (SW) during the loss of SW accident sequences. The event tree functions were revised to quantify the respective fault trees with SW unavailable.

9. The steam generator tube rupture event tree was modified to take into account the potential for the ruptured steam generator power operated relief valve (PORV) or a safety relief valve to reclose following success of cooldown function.

10. Fault trees were revised to include the configuration where charging pump 1C can be energized by either H or J buses. This configuration allows the 1C pump to start manually on the H bus when the 1A pump is unavailable, or on the J bus when the 1B pump is unavailable.

11. The service water fault trees were revised to incorporate the assumption that the Unit 1 pumps are running and the Unit 2 pumps are in standby.

12. A new circulating water (CW) system fault tree is developed to model the condenser dependency on the CW pumps.

13. The main steam fault tree was revised to include the steam valve failure due to the C9 interlock failure.

14. Revised the reactor trip function in the RP100 fault tree to indicate that both MG set supply breakers have to be open to de-energize the control rods.

15. The dependency of the reactor coolant pumps on the component cooling was added to the model.

16. The dependency of the component cooling heat exchangers on the service water was added to the model.

17. The dependency of bearing cooling on the condensate pump oil cooler was included in the model.

18. The cross-tie between the Unit 1 and Unit 2 bearing cooling systems was failed, since this crosstie is never expected to be used.

19. The valves supplying service water to instrument air compressors heat exchangers were added to the model.

20. Service water cooling to the Unit 2 charging pumps dependency was added to the model.

21. The crosstie between the Unit 1 and Unit 2 charging pumps was added.

22. The model was revised to include only the Unit 2 charging pumps' suction from the refueling water storage tank. The suction from the volume control tank was deleted.

23. The model was revised to include the ventilation dependency on the charging pump cubicles.

24. The alternate AC diesel generator was credited and included in the model.