VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

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United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

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Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2
SUMMARY OF FACILITY CHANGES, TESTS AND EXPERIMENTS

Pursuant to 10 CFR 50.59(d)(2), a report containing a brief description of any changes, tests, and experiments, including a summary of the evaluation of each, must be submitted to the NRC, at intervals not to exceed 24 months. Attachment 1 provides a summary description of Facility Changes, Tests and Experiments identified in 10 CFR 50.59 Evaluations implemented at the North Anna Power Station during 2013. Attachment 2 provides a Commitment Change Evaluation Summary that was completed.

If you have any questions, please contact Page Kemp at (540) 894-2295.

Very truly yours,

Gerald T. Bischof
Site Vice President

Attachments

1. 10 CFR 50.59 Summary Description of Facility Changes, Tests and Experiments
2. Commitment Change Evaluation Summary

cc: Regional Administrator
United States Nuclear Regulatory Commission
Region II
Marquis One Tower
245 Peachtree Center Ave., NE, Suite 1200
Atlanta, Georgia 30303-1257

NRC Senior Resident Inspector
North Anna Power Station

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NRR

ATTACHMENT 1

10 CFR 50.59 SUMMARY DESCRIPTION OF
FACILITY CHANGES, TESTS AND EXPERIMENTS

NORTH ANNA POWER STATION UNITS 1 AND 2
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

NORTH ANNA UNITS 1 AND 2

10 CFR 50.59 SUMMARY DESCRIPTION OF FACILITY CHANGES, TESTS AND EXPERIMENTS

10 CFR 50.59 EVALUATION: 13-SE-MOD-01

Document Evaluated: Design Change 12-00052, Unit 1 Station Service Bus Relays

Brief Description: This modification replaces the Station Service (SS) Bus GE analog "NGV" Under Voltage (UV) relays with Beckwith M-2410A digital relays. The new microprocessor controlled relay has a reduced Channel Statistical Average (CSA) that allows the UV setpoint to be lowered, while maintaining the voltage to keep the Reactor Coolant Pump (RCP) running. The setpoint needs to be lowered to allow a Main Feed Water (MFW) Pump to auto-start. A new annunciator window will be provided to notify Operations of a UV, self test error, or blown fuse. The Beckwith relay will also replace the time delay function used to preclude nuisance trips.

Reason for Change: On May 28, 2010, a cascaded series of events led to the loss of power to Unit 2 "B" RCP (2-RC-P-1B) resulting in a trip of North Anna Unit 2. During this event, the standby MFW pump tandem motors on Unit 2 "B" SS Bus simultaneously auto started, which caused the non-safety (NS) related SS Bus NGV Under Voltage (UV) relays to trip due to a voltage dip, resulting in the loss of power to the RCP. Restoration of power to the RCP by fast transfer did not occur due to the loss of "B" Reserve Station Service Transformer (RSST) earlier in the event.

Request for Engineering Assistance (REA) R2010-056B was issued in response to Root Cause Evaluation (RCE) 001012. The REA requested to replace the GE "NGV" UV relays with newer relays that have a better accuracy and less drift. The REA also requested an evaluation of the UV setpoint to minimize the actuation of the SS Bus UV relays during a MFW pump start. This REA has resulted in development of several modifications to replace the NGV relays on the SS Buses and the Emergency Buses.

Summary: This modification replaces the SS Bus GE analog "NGV" Under Voltage (UV) relays with a digital microprocessor relay, integrates the time delay function for a load shed, and lowers the UV setpoint. The method of performing the UV protection with a microprocessor controlled relay is different. The Beckwith relay will use software, the protection will incorporate the time delay function of the analog GE SAM 11 time delay relay, and the microprocessor relay will require 125 VDC.

The design function of the SS Bus UV relay is to ensure sufficient voltage is available to provide the maximum torque the RCP motor will carry without a step decrease in speed. These relays trip the affected SS Bus motor loads when the UV setpoint is actuated. This relay will also trip the respective RCP if the voltage is not restored after 30 seconds. The SS Bus UV relays are associated the Updated Final Safety Analysis

(UFSAR) discussion in Section 8.3.1.1.1, and Section 15.3.4.2. UFSAR Section 8.3.1.1.1 states an UV will trip the normal bus feeder breakers. UFSAR Section 15.3.4.2, states the loss of three out of three RCP's due to an undervoltage was analyzed, and that a loss of flow due to an undervoltage is assumed to trip the Reactor on low flow in any loop.

The relay has been tested, and the software has been verified in accordance with Industry Guidelines for Safety Related applications, for this Non-Safety Related installation. Replacing the GE SAM 11 time delay function with the Beckwith relay also reduces the 125 VDC battery load. The relay is more accurate, accommodates the voltage dip during the auto-start of a MFP motor, and annunciates a loss of potential signal, loss of control power signal, or a "Self Test" error, is acceptable.

A random relay was subjected to a 120 hour burn-in test, at 120 degree F, for 5 continuous days, and was then subjected to a successful Factory Acceptance Test. The relay meets the normal operating environmental conditions of the Normal Switchgear Room. The Station Service Bus UV relay is a Non-Safety Related application, and does not require Seismic Qualification.

The replacement of the "NGV" UV relay with a digital relay does not result in any new types of system level failure modes. The modification does not physically alter any equipment, system performance, or operator actions that could affect an UV or loss of RCP flow, such the the current UFSAR analyses remain bounding.

10 CFR 50.59 EVALUATION: 13-SE-MOD-02

Document Evaluated: Design Change NA-11-01082, Main Steam Radiation Monitor Replacement

Brief Description: This Design Change (DC) replaces the Unit 1 and Unit 2 existing Nuclear Research Corporation (NRC) High Range Post-Accident radiation monitors for the Main Steam Lines and Auxiliary Feedwater Turbine Exhaust with Mirion Technologies (MGPI) radiation monitors. Nuclear Research Corporation is out of business; technical support and replacement parts are no longer available. Due to obsolescence of NRC components, workarounds for detectors and data logger problems have become increasingly difficult.

Reason for Change: The existing NRC high range radiation monitors are degrading. The Main Steam Radiation Monitors (MSRMs) are Regulatory Guide (RG) 1.97 components and are designed to monitor the effluent radiation release from the Main Steam System in post-accident events. The manufacturer of the currently installed radiation monitors, Nuclear Research Corporation, is out of business, technical support and replacement parts are no longer available.

Summary: Design Change NA-11-01082 is replacing eight obsolete radiation monitors on the main steam lines (six total) and auxiliary turbine exhaust (two total) with reliable radiation monitors that will meet the requirements of Regulatory Guide 1.97 and NUREG-0578. This will be accomplished with new MGPI equipment that will be easier to repair and service due to availability of replacement parts and technical support. The new MGPI Local Processing and Display Units (LPDUs) will be located in the Quench Spray Pump House (QSPH) and the Turbine Driven Auxiliary Feedwater Pump House (AFWPH) for each Unit and will be powered from the associated Unit Semi-Vital bus. MGPI and Yokogawa equipment will provide control room display and recording capabilities in the existing Spillway Supervisory Panel #1 (1-EI-CB-121) in the Unit 2 control room. Panel 1-EI-CB-121 will be renamed.

The new monitors will provide post-accident monitoring capability only (no control functions) for UFSAR Chapter 15 Condition IV Events.

The failure modes of the existing system bound the failures of the new system. Regarding power supplies, both the existing design and the new design provide a highly reliable power supply that meets the requirements of RG 1.97 for Category 2, Type E variables. The new power source is backed by the emergency diesels and in the event that normal AC power is lost the new equipment has been verified during the Factory Acceptance Test (FAT) to restart in less than 15 minutes with no required operator action upon restoration of emergency power. This restart time is similar to the existing equipment.

10 CFR 50.59 EVALUATION: 13-SE-TM-01

Document Evaluated: Temporary Modification 2013-1312, 2-CC-TV-204B Instrument Air Jumper

Brief Description: A Temporary Modification (TM) is to be installed on 2-CC-TV-204B to jumper around the solenoid operated valves (SOVs) supplying air to the actuator. This modification will open the valve and keep it in the open position until removed. The action will make the valve unable to perform its design function of closing on a Containment Isolation "B" signal, requiring entry into Technical Specification (TS) 3.6.3, Containment Isolation Valves. This TM has been prepared as a contingency plan should maintenance on the 2-CC-TV-204B Control Room positioning indicating lights cause the SOVs supplying air to the 2-CC-TV-204B actuator to close. This TM is being performed in Mode 1. Valve 2-CC-TV-204B is expected to be restored to be able to perform its design function within the TS 3.6.3 allowed Completion Time of 4 hours.

Reason for Change: Both "A" and "B" train (2-CC-TV-204B-1 and 2-CC-TV-204B-2) open (red) indication lights are currently burned out and need to be replaced. These are penetration flow path containment isolation valve position indications required by TS 3.3.3, Post Accident Monitoring Instrumentation.

Summary: This TM does not result in more than a minimal increase in the frequency of occurrence of accidents and malfunctions of structures, systems, or components (SSCs) important to safety as the TM maintains 2-CC-TV-204B in the position assumed as an initial condition for accidents and malfunctions of SSCs. The consequences of accidents and malfunctions of SSCs important to safety will not result in more than a minimal increase due to isolation of the penetration flow path from the inside containment check valve, 2-CC-115. Accidents and malfunctions of a different type than previously evaluated are not possible as the failure of an active component to perform its design function has been previously evaluated. The design basis limit for a fission product barrier will not be exceeded as the containment penetration isolation is being maintained by the inside containment check valve, 2-CC-115. There are no methods of evaluation described in the UFSAR associated with the activity being evaluated.

10 CFR 50.59 EVALUATION: 13-SE-PROC-01

Document Evaluated: 0-OP-7.3, Revision 0, Filling U-1 or U-2 SI Accumulators with Air Operated Hydro Pump

Brief Description: 0-OP-7.3, Rev. 0 was written to control and install a temporary modification. The procedure allows the use of two (2) methods of attaching a temporary pump (with hoses) to the Refueling Water Storage Tank (RWST) on the suction side and to the Safety Injection (SI) system on the discharge side. 1-SI-P-2, Safety Injection Hydo Test Pump, is used under normal operating conditions to adjust the level in either unit's SI accumulators. 0-OP-7.3 provides methods to bypass the normal pump in the event the normal pump is inoperable or out for maintenance, which would allow for the adjustment of the SI accumulators for either unit. Method one requires that a spool piece tee be installed between the common RWST suction to 1-SI-P-2. From the tee properly rated safety-related hose would connect the pump to an instrumentation line. Method two directs the installation of a hose directly to an existing drain valve on the suction side of the RWST and high pressure hose on the discharge side of the temporary pump is connected to another drain valve on the SI System.

Reason for Change: 1-SI-P-2, Safety Injection Hydro Test Pump, and the associated piping is safety-related but non-ASME class and used to fill either unit's SI accumulators in all operating modes. Though this pump is safety-related, it is normally manually isolated from both the RWST and the SI accumulators, and performs no accident mitigation safety function. This procedure provides alternate methods of filling the SI accumulators upon failure of 1-SI-P-2 using a temporarily installed air operated pump.

Summary: This activity implements 0-OP-7.3, Filling U-1 or U-2 SI Acumulators with Air Operated Hydro Pump. The procedure provides two methods of filling either unit's SI accumulators. The first way installs a tee connection with an isolation valve at an existing spool piece. Safety-related hose is connected to the tee to provide a suction source to a temporary pump. The discharge of the pump is connected, via safety-related high pressure hose to an existing instrument control valve (ICV) and the appropriate valve configuration is established for filling. The second method installs the same safety-related hoses directly to existing drain and vent valves. Method one is used in the event that the normal fill pump, 1-SI-P-2, is removed from service for maintenance or is non-functional.

An evaluation is conservatively deemed required due to the fact that the suction source, RWST and piping, are seismically qualified and the assumption is made that the suction side hose connection will fail. The activity screened in due to having a potentially adverse affect on a SAR described design function. The SAR described design functions that are potentially affected are the RWST and Emergency Core Cooling System. No method(s) of evaluation are affected by this activity.

A seismic event during application of 0-OP-7.3 has the potential to cause a hose failure and the loss of RWST inventory. If not isolated, leakage could drain the RWST below Technical Specification limits causing the affected unit to operate outside of the assumptions of the accident analysis. Review of the activity shows that there is no increase in the frequency of occurrence of an accident. A seismic evaluation was performed on the temporary equipment (tee, hoses and pump) and it was shown that no adverse affect to the piping system was incurred. The evaluation shows that there is less than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety. Based on the evaluation of the most limiting conditions associated with 0-OP-7.3, the is no increase in the consequences of an accident, consequences of a malfunction of a SSC, the creation of an accident of a different type or the possibility for an accident of a different type to occur. The evaluation also states that no design basis limit for a fission product barrier is exceeded due to the activity. Finally, the evaluation documents that no method of evaluation is affected by this activity.

10 CFR 50.59 EVALUATION: 13-SE-OT-01

Document Evaluated: Engineering Technical Evaluation (ETE) NAF-2012-0157, Implementation of the North Anna Unit 2 Reactor Vessel Closure Head Analysis Supporting the Wesinghouse RFA-2 Fuel Transition

Brief Description: This activity involves revising the stress analyses for the North Anna Unit 2 Reactor Vessel Closure Head, Intermediate Lift Ring, and Control Rod Drive Mechanism (CRDM) Nozzle Plugs. These stress analyses were revised to account for the updated seismic and LOCA loads resulting from the analysis of the RFA-2 fuel transition and the revised stiffness value of the lower radial keys located at the bottom of the Reactor Pressure Vessel.

Reason for Change: The analyses supporting the RFA-2 fuel transition were implemented and screened in ETE-NAF-2011-0173, Rev. 0, which included the North Anna Unit 1 Reactor Vessel Closure Head, Intermediate Lift Ring and CRDM Nozzle Plugs. Due to time and resource constraints, the Unit 2 Reactor Vessel Closure Head, Intermediate Lift Ring, and CRDM Nozzle Plug analyses were excluded from ETE-NAF-2011-0173, Rev. 0. Thus, the North Anna Unit 2 Reactor Vessel Closure Head, Intermediate Lift Ring, and CRDM Nozzle Plug analyses are implemented and screened as part of ETE-NAF-2012-0157, Rev. 0. This 10 CFR 50.59 Evaluation addresses the implementation of these analyses.

Summary: This activity involves revising the stress analyses for the North Anna Unit 2 Reactor Vessel Closure Head, Intermediate Lift Ring, and CRDM Nozzle Plugs. These stress analyses were revised to account for the updated seismic and LOCA loads resulting from the analysis of the RFA-2 fuel transition and the revised stiffness value of the lower radial keys.

An Evaluation was required because the reanalysis of the stresses on SAR-described structures, systems, and components (SSCs) constitutes an adverse effect on the design function.

The design basis function of the Reactor Vessel Closure Head (including the CRDM Nozzle Plugs) is to maintain the Reactor Coolant System (RCS) pressure boundary. This is maintained by verifying that stresses on the associated structures on the reactor vessel head remain acceptable per the design specification requirements. The Reactor Vessel Closure Head (including the CRDM Nozzle Plugs) stresses for North Anna Unit 2 were reanalyzed as part of the RFA-2 fuel design transition at North Anna.

The design basis function of the Intermediate Lift Ring is to allow the lifting of the replacement Reactor Vessel Closure Head. This function is maintained by verifying stresses on the Intermediate Lift Rig remain acceptable per the design specification requirements. These stresses were reanalyzed for North Anna Unit 2 as part of the RFA-2 fuel design transition at North Anna.

The reanalysis of the North Anna Unit 2 Reactor Vessel Closure Head, Immediate Lift Rig, and CRDM Nozzle Plugs with the RFA-2 fuel and the revised lower radial key stiffness value is acceptable per the design specification requirements. All requirements, including component stress limits, have been met.

Part II-1 through Part II-6 of the 10 CFR 50.59 Evaluation confirmed that the reanalysis of the North Anna Unit 2 reactor vessel head has no impact on the associated design functions. Thel design basis limit for a fission produce barrier (DBLFPB) (ASME code stress limits) associated with RCS pressure boundaries remain unchanged and were shown to be met. Finally, there was no change in the method of evaluation. Therefore, the reanalysis of the North Anna Unit 2 Reactor Vessel Closure Head, Intermediate Lift Ring, and CRDM Nozzle Plugs can be implemented without NRC approval.

10 CFR 50.59 EVALUATION: 13-SE-OT-02

Documents Evaluated: Temporary Shielding Request 13-TSR-021

Brief Description: Temporary lead blanket shielding will be installed on operable Unit 2 Safety Injection (SI) System piping during the Unit 2 Spring 2013 refueling outage.

Reason for Change: The purpose of the shielding is to reduce the dose rates for 2-RC-MOV-2593 (Reactor Coolant Loop Stop Valve) repair activities. Lead blanket installation, use and removal will comply with VPAP-2105, Temporary Shielding Program requirements.

Summary: Temporary lead blanket shielding will be installed on operable Unit 2 SI System piping during the Unit 2 Spring 2013 refueling outage per details in Temporary Shielding Request (TSR) 13-TSR-021. The purpose of the shielding is to reduce dose rates for 2-RC-MOV-2593 repair activities. Lead blanket installation, use and removal will comply with VPAP-2105 program requirements. The proposed temporary lead shielding installation will be performed in a manner that allows the loaded Reactor Coolant (RC) and SI piping to remain operable with fuel in the vessel and the piping flooded; details of the loading installation and the removal deadline are provided in the TSR.

Reactor Coolant and SI accidents previously evaluated in the SAR are associated with expected plant operating conditions, including shutdown. Since the proposed temporary lead shield loading is acceptable, the installation will not cause an accident and there are no increased consequences to consider. The potental malfunction associated with attaching 600 pounds of lead blankets on the SI piping is pipe overstress/leakage. Since the proposed temporary lead shielding is acceptable, the installation will not cause a malfunction and there are no increased consequences to consider. All potential concerns associated with the safe use of temporary shielding have been addressed via the VPAP-2105 program. All appropriate precautions to avoid an accident of a different type or a malfunction with a different result have been identified via the Engineering Evaluation. The potential for exceeding or altering the design basis limit for fission product barrier is not considered credible based on the TSR's Engineering Evaluation of the proposed loading. A new method of evaluation is not part of the proposed TSR. Therefore, this change may be implemented without prior NRC approval.

10 CFR 50.59 EVALUATION: 13-SE-OT-03

Documents Evaluated: Engineering Technical Evaluation (ETE)-NAF-2013-0053, Implementation of a Revised Rod Ejection Analysis for North Anna Power Station

Brief Description: The activity being evaluated is the implementation of the revised Control Rod Ejection Accident analysis for North Anna Power Station (NAPS). A revised analysis was conducted to rectify a latent error in the accounting of bypass flow, found in the models used for the analysis. This analysis supports the design function of the fuel, specifically the fuel integrity. Results of the revised analysis are closer to the UFSAR specified design criteria than the current analysis. This activity also requires an update to a table and some figures corresponding to Section 15.4.6 of the NAPS UFSAR.

Reason for Change: A revised analysis was conducted to rectify a latent error in the accounting of bypass flow, found in the models used for the analysis.

Summary: The activity being evaluated is the implementation of the revised Control Rod Ejection Accident analysis for NAPS. The revised analysis was conducted to rectify a latent error found in the accounting of bypass flow in the models previously used for the analysis. Results of the revised analysis are closer to the UFSAR specified design criteria than the current analysis. However, all acceptance criteria specified in the NAPS UFSAR and Rod Ejection Topical Report, VEP-NFE-2-A continue to be met.

The evaluation of this activity confirmed that the reanalysis of the control rod ejection safety analysis does not result in an increase in the frequency of occurrence of an accident previously evaluated in the SAR, malfunction of a structure, system, or component (SSC) important to safety, increase in the consequences of an accident, a possibility for a malfunction of a SSC important to safety with a different result, or an accident of a different type. The results of the revised analysis show no new failure modes, new malfunctions of SSCs important to safety, or unanalyzed accidents being generated. Furthermore, the radiological consequences of the Control Rod Assembly Ejection Accident are shown to be unchanged based upon the acceptance criteria for cladding embrittlement and fuel melt being met. The design basis limit for a fission product barrier (DBLFPB) associated with control rod ejection safety analysis were shown to be met and there was no change in the method of evaluation. Therefore, the reanalysis of the control rod ejection safety analysis may be implemented without NRC approval.

ATTACHMENT 2

COMMITMENT CHANGE EVALUATION SUMMARY

NORTH ANNA POWER STATION UNITS 1 AND 2
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

Commitment Change Evaluation Summary

Original Commitment Description:

NRC Bulletin No. 88-08 was issued requesting that licensees identify sections of unisolable injection piping which could be potentially susceptible to thermal stresses from temperature stratification or oscillations induced by leaking valves. These thermal stresses caused through-wall cracks of 6-inch safety injection piping between the check valve and reactor coolant system at the Farley 2 and Tihange 1 stations. The bulletin also requested that the licensees take the appropriate actions to ensure that such piping would not be subjected to unacceptable thermal stresses which could cause fatigue failure.

As a result of the review, the following lines, where cooler charging system water could leak-by the Boron Injection Tank bypass valve or High Head Safety Injection (HHSI) motor operated valves (MOVs), were determined to be potentially susceptible to the thermal stress concern at North Anna Power Station:

<u>North Anna Unit 1</u>

6"-RC-17-1502-Q1
6"-RC-19-1502-Q1 SAFETY INJECTION SYSTEM (COLD LEG)
6"-RC-20-1502-Q1

6"-RC-16-1502-Q1
6"-RC-18-1502-Q1 SAFETY INJECTION SYSTEM (HOT LEG)
6"-RC-21-1502-Q1

<u>North Anna Unit 2</u>

6"-RC-417-1502-Q1
6"-RC-419-1502-Q1 SAFETY INJECTION SYSTEM (COLD LEG)
6"-RC-420-1502-Q1

6"-RC-416-1502-Q1
6"-RC-418-1502-Q1 SAFETY INJECTION SYSTEM (HOT LEG)
6"-RC-421-1502-Q1

In addition to the susceptibility of the above safety injection branch lines, the following auxiliary spray lines were added to the potentially susceptible list:

<u>North Anna Unit 1</u>

4"-CH-A14-1502-Q1

North Anna Unit 2

4"-CH-815-1502-Q1

Correspondence letters (Serial Numbers 88-433, -433A, -433B, -433C, -433D, and -433E) provided North Anna's response and actions taken to address the concerns outlined in NRC Bulletin No. 88-08. The susceptible lines were non-destructively evaluated addressing the concerns outlined in NRC Bulletin No. 88-08, Supplement 2. In addition, a temperature monitoring program was created which placed externally mounted thermocouples at locations in the piping where potential temperature stratification or oscillations would be expected to be the most severe. If any thermal fluctuations are determined to be outside the acceptable limits, then action by Engineering to evaluate the condition is required. The temperature monitoring program was considered an interim solution until long term solutions were implemented. NRC letter dated September 27, 1991 determined that the response from North Anna was consistent with the monitoring alternatives stated in Bulletin No. 88-08. The temperature monitoring used to meet the commitment to Bulletin 88-08 is performed through PT-44.10.9B (Thermal Stratification of Safety Injection Lines).

Revised Commitment Description:

The program to monitor temperatures in stagnant piping lines was considered to be an interim solution to the pipe thermal stress concern outlined in NRC Bulletin 88-08. Monitoring for thermal stratification by means of externally mounted thermocouples at North Anna Power Station may be discontinued based on the fact that no indications of significant temperature oscillations have been identified to date and MRP-146 (Materials Reliable Program Management of Thermal Fatigue in Normally Stagnant Non-Isolable RCS Branch Lines) has been successfully implemented.

Justification for the Commitment Change:

Over the past four years, a large number (between 50 to 70%) of the required externally mounted thermocouples used to perform PT-44.10.9B have failed at some point in time. Attempts have been made to ensure the thermocouples are properly attached and not damaged. Despite these efforts, the thermocouples continue to lose functionality over time and require considerable maintenance in dose intensive locations each refueling outage. Dominion finds no significant value added in maintaining the thermocouple monitoring system for the RCS branch lines since the piping is within the scope of Dominion's Thermal Fatigue Management Program. The Thermal Fatigue Management Program screens every normally stagnant RCS branch line and performs an initial inspection of every line that does not screen out. The program effectively evaluates the thermal cycle significance of the piping against the criteria in MRP-146.

The requirements for addressing potential thermal fatigue per MRP-146 includes breaking down the normally stagnant branch piping into two basic configurations relative to the RCS piping — Up-Horizontal (UH) and Down-Horizontal (DH). UH lines have a horizontal section that interfaces directly into the RCS piping from the side or a horizontal section that turns downward then intersects the RCS from the top. DH lines intersect the RCS piping at the bottom (or within the lower circumferential arc) of the RCS piping before turning horizontal.

If any of the normally stagnant piping systems do not screen out, the piping will require inspection each refueling outage. However, if in-leakage past the valves is less than the threshold in-leakage value established in PT-61.6 based on MRP-170 software results, then no thermal cycling is predicted to occur and inspection is not warranted. Ongoing valve leakage testing every refueling outage allows for the determination if valve in-leakage is below the threshold values stated in PT-61.6. North Anna performs PT-61.6 as the leakage test for the HHSI MOVs Isolation Valves (Unit 1: 1-SI-MOV-1836, -1869A, -1869B; Unit 2: 2-SI-MOV-2836, -2869A, -2869B) where charging header pressure is used to create a differential pressure across each MOV. Test rigs are used to measure the valve leakage on the low pressure side of the valves. Upon completion of a satisfactory in-leakage testing; the valves are not further manipulated or stroked.

If any in-leakage threshold values are reached, then the affected piping is examined through the Augmented ISI Program with inspection to the requirements of MRP-146. The ultrasonic (UT) examination method utilized is in accordance with ASME Section XI and MRP-146 inspection requirements.

Based on Engineering review, the 4-inch auxiliary spray lines have been screened out and excluded from the Thermal Fatigue Management Program per the guidance of MRP-146. This is because the velocities in the lines are not sufficient to produce thermal fatigue cycling due to insufficient swirl penetration.

In conclusion, Dominion finds that performing leakage testing at North Anna during refueling outages, and maintenance of the associated isolation valves (as required) to maintain leakage below the threshold value limits, is an acceptable alternative to the temperature monitoring currently being performed to meet the requirements of NRC Bulletin 88-08. Without substantial in-leakage past the valves, the thermal cycling during normal operating conditions is negligible and would not result in fatigue damage. Due to the configuration of the auxiliary spray lines; the spray lines are not considered susceptible to thermal fatigue per MRP-146. The temperature monitoring performed up to this date has not indicated any significant thermal oscillations in the identified safety injection or the auxiliary spray lines. The temperature monitoring program does not add significant value to the evaluation of piping already under the Thermal Fatigue Management Program; therefore, its continued operation does not justify the impact to ALARA in order to maintain the monitoring thermocouples.