VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

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Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNIT NOS. 1 AND 2
SUMMARY OF FACILITY CHANGES, TESTS AND EXPERIMENTS

Pursuant to 10 CFR 50.59 (d)(2), enclosed is a summary description of facility changes, tests and experiments, including a summary of the safety evaluations, that were implemented at North Anna Power Station during 2000. Also enclosed in a commitment evaluation summary that was performed in 2000.

If you have any questions, please contact us.

Very truly yours,

DAH

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Enclosures

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IE47

SAFETY EVALUATION LOG
SPECIAL TESTS
2000

S.E. #	Unit	Document	System	Description	SNSOC Date
00-SE-ST-01	1,2	0-ST-77.0 (Rev. 0)		"ECCS Pump Room Exhaust Air Cleanup System (PREACS) Test" -configures the primary ventilation system in a line-up to simulate post-accident conditions.	3-07-00
00-SE-ST-01 REV. 1	1,2	0-ST-77.0 (Rev. 1)		"ECCS Pump Room Exhaust Air Cleanup System (PREACS) Test" -50.59 not affected – the revision to the ST only affects data collection points.	5-30-00
00-SE-ST-02	1,2	0-ST-78		"ECCS PREACS Single Failure Test" -verifies that the safeguards bldg. & charging pumps will receive adequate ventilation flow with a single active failure in the ventilation system or ventilation control system.	11-10-00

00-SE-OT-01

Description
0-ST-77.0, "ECCS Pump Room Exhaust Air Cleanup System (PREACS) Test"
0-ST-77 configures the primary ventilation system in a line-up to simulate post-accident conditions. The post-accident alignment is presently dictated by existing emergency operating procedures as a requirement of JCO 99-02. Additionally, the Special Test will configure the primary ventilation system to simulate conditions with one unit shutdown coincident with a CDA on the operating unit to ensure adequate cooling flows are present in the operating unit's Safeguards Area and the operating Charging Pump cubicles.

Summary
BACKGROUND
The UFSAR describes the operation of each of the primary ventilation areas to include ability to selectively pass the exhaust systems through the Iodine Filter Loops as it becomes necessary for decontamination. This includes exhaust from the Auxiliary Building (Central and General Exhaust), Decontamination and Waste Solids Building, Fuel Building, Containment (Shutdown Unit) and Engineered Safety Features Areas. During a design basis accident, the areas that require filtration include the Auxiliary Building Central, and the Engineered Safety Features Areas. Troubleshooting of the Primary Ventilation System (documented in ET SE-99-051) identified that various dampers for the Fuel Building, Auxiliary Building, and Unit 2 Safeguards leaked by. This leak-by potentially creates a flow path that bypasses the filter assembly following an accident (LOCA). The most conservative alignment of ventilation systems to ensure that exhaust flow will not bypass the charcoal and HEPA filter assembly is to secure all systems that are not aligned to the filters. As a result, Emergency Procedures 1/2-E-0, 1/2-ES-1.3, and 0-AP-36 were revised in support of JCO 99-02 to ensure that the Auxiliary Building Central and ESF areas were aligned to provide filtration following a DBA.

PROPOSED ACTIVITY
0-ST-77 is a Special Test designed to obtain primary plant ventilation data and balance the system to ensure proper flows in vital areas. The test is essentially configured to perform the following activities on EACH train:

- PREACS is aligned like existing EOPs dictate to simulate post-accident conditions (a Safeguards fan is RUNNING from each unit and one Central Exhaust fan is RUNNING through a single filter bank – Waste Solids & Decontamination, Fuel Building, Containment, and Auxiliary Building General Exhaust fans are OFF and dampers aligned in BYPASS). A Hydrogen Recombiner cooling fan is operating to simulate post-accident Recombiner backpressure. Engineering data is taken to ensure proper flows and pressures at key areas. If required, balancing dampers are adjusted to ensure proper system flows.
- DOP testing is performed to quantify leakage past the Iodine filters.
- One Safeguards fan is placed in BYPASS to simulate a one unit filtration condition. This configuration would occur with a CDA on one unit while the other unit shutdown with no safeguards fans running. Engineering data is acquired and if required, further balancing is performed. This data will support increases in allowable ECCS leakage from the current level of 600 cc/hr.
- A 2nd Charging Pump on one unit is started to simulate accident conditions on one unit. The Charging Pump motors have booster fans. Engineering data is acquired to ensure adequate cooling is available to the operating Charging Pumps. Additionally, this data will support increases in allowable ECCS leakage from the current level of 600 cc/hr.
- If desired, engineering data for Safeguards fan flow is acquired by placing Auxiliary Building Central Exhaust in BYPASS.
- Restoration to normal conditions is then performed.

FAILURE MODES
Failure of the activity is considered failure of a damper in a non-ESF area to position to the Bypass mode with its associated fan secured. The most limiting condition would be during the simulated condition of one unit operation and a CDA occurs on the unit where the Safeguard fan is not in BYPASS. Since no supply fans are operating, it is unlikely that the bypass leakage would be so bad that combined flow

provided by the operating Safeguards fan and one Aux. Building Central fan would not be adequate to meet TS flow requirements, maintain the Safeguards area in a vacuum, and provide cooling flow to ESF equipment. This is inconsequential, however, since existing Emergency Procedures will place the system in an analyzed condition with 2 Safeguard fans and 1 Auxiliary Building Central fan aligned in FILTER mode, and all other areas secured. Additionally, it should be emphasized (regardless of actual flows obtained) that our current JCO configuration does not take credit for filtration through the charcoal and HEPA filter assemblies since analysis has shown that if all exhaust is unfiltered, and ECCS leakage is less than 600 cc/hr, we will still be below allowable release limits. For the non-ESF areas, the associated ventilation fans will be secured during the accident, and their dampers will be placed in the BYPASS position. Temperatures in these areas will change according to outside ambient conditions and equipment loading when their respective ventilation fans are secured. Ambient temperatures in these areas are recorded by Operations on rounds periodically. Temperatures outside of normal bounds will cause the Operator to take actions prescribed by existing procedures to correct the condition or declare the associated equipment inoperable. The ambient temperature monitoring instrumentation will still be available.

UNREVIEWED SAFETY QUESTION DETERMINATION

Operation of the ventilation system as proposed by this Special Test creates no unique precursors or precursor events for Chapter 15 accidents. The proposed Special Test does not change the intended operation of the charcoal filter bank or equipment required for accident mitigation. Existing EOP procedures will ensure that flow from ESF areas does not bypass the Iodine Filter Loops due to potential leakage past system dampers. Non-ESF area temperatures are monitored by operators, and actions can be taken in accordance with existing procedures to mitigate high or low temperature conditions to prevent environmental challenges to equipment. The ESF Area ventilation system will continue to receive its automatic swap-over to the Iodine Filter Loops. Thus, any leakage of radioactive materials from ECCS equipment in the pump room following a LOCA will be filtered prior to reaching the environment, and the margin of safety as defined in the TS bases is not reduced. Current accident analyses do not take credit for filtration of ECCS leakage for the first hour of the accident. Since all equipment manipulations are performed from the control room, the one hour limitation does not impose a significant challenge to the operator. For these reasons, an unreviewed safety question does not exist.

Since the activity will ensure the design basis assumptions are satisfied by eliminating potential filter bypass flow paths and ensuring filtration of Safeguards ventilation exhaust during a design basis accident, the activity should be allowed.

00-SE-OT-02

Description
0-ST-78, "ECCS PREACS Single Failure Test"
This Special Test ensures that the ECCS Pump Room Exhaust Air Cleanup System (PREACS) will provide adequate cooling considering single active failures in the Primary Ventilation System.

Summary

BACKGROUND

The Primary Ventilation System needs to be able to sustain a single failure and still provide adequate cooling to the safety-related equipment. Plant Issue N-2000-1925 addresses this issue. The Plant Issue states that the Primary Ventilation System is not in full compliance with Reg. Guide 1.52 and UFSAR sections 9.4.6.3 and 9.4.8.2 which require ability to sustain single failures and still assure filtration. A single failure analysis was performed. The Memorandum (attached to Plant Issue), "Single Failure Evaluation of the North Anna Primary Ventilation System", discusses the Primary Ventilation System sustaining single failures. It was concluded that the system is not required to be single-failure proof regarding the filtration function but it is required for the cooling function of safety related equipment. Following the evaluation of the ventilation system this Special Test will provide corrective actions and confirm some assumptions that were made.

PROPOSED ACTIVITY

0-ST-78 is a Special Test designed to verify that the Safeguards Building and Charging Pumps would receive adequate ventilation flow with a single active failure in the Ventilation System or Ventilation Control System. The test is essentially configured to perform the following activities:

- The first test verifies adequate Charging Pump cooling under worst conditions: Auxiliary Building central fans off, filter and bypass dampers both closed. Normal ECCS PREACS alignment is established: two Safeguards Exhaust Fans running, one Auxiliary Building Central Exhaust fans running, all other fans off. The Hydrogen Recombiner is not running. Two Charging pumps are operating. Failure of 1-HV-AOD-103-3 is simulated. The Hydrogen Recombiner fan is started. A second Unit 1 Charging pump is started. Measurements are taken throughout these steps. Finally, 1-HV-AOD-103-3 is opened and 1-HV-F-8A, A Auxiliary Building Central Exhaust Fan is started. This places the system in the full ECCS PREACS condition.
- The next two tests verify adequate Safeguards Building cooling with two configurations: no Safeguards fan (Auxiliary Building Central fan is pulling from both Safeguards and Auxiliary Building Central areas), and non-safety system damper open (Safeguards fan and Auxiliary Building Central fan pulling from Safeguards, Auxiliary Building Central, and Non-safety system areas). The test starts in a full ECCS PREACS alignment, including a second Unit 1 Charging Pump running, and secures both Unit 1 and Unit 2 Safeguards Exhaust Fans. (Only one charging pump on Unit 2 should be operated due to system conditions at the present time.) Next the Safeguards Exhaust Fans are re-started and Unit 2 is placed in "Bypass". The Auxiliary Building General system is placed in the "Filter" mode to simulate a failure of a non-safety system damper. Measurements are taken throughout these steps.
- Finally, the system is re-aligned to its normal configuration.

FAILURE MODES

Existing EOPs ensure that there will be sufficient ventilation provided through the Iodine Filter to perform their safety functions. The loss of equipment cooling may lead to equipment failure if ambient temperatures exceed their allowable ranges for an extended period of time. However many controls have been included in the procedure to prevent this from happening. Failure of equipment in areas outside the Auxiliary Building Central and Safeguards Area due to poor ventilation was considered. The potential degradation of the charging pumps due to the loss of cooling was considered. Therefore, extra emphasis will be used to monitor the charging pumps to ensure that the pump motors do not become overheated. If they do become overheated there are steps within the procedure to stop the special test. This action will establish ventilation cooling to the charging pump cubicles prior to the pumps becoming compromised. Existing emergency procedures provide sufficient ventilation after a CDA, in the event a CDA occurs coincident with performing the Special Test.

UNREVIEWED SAFETY QUESTION DETERMINATION

Operation of the ventilation system as proposed by this Special Test creates no unique precursors or precursor events for Chapter 15 accidents. The proposed Special Test does not change the intended operation of the charcoal filter bank or equipment required for accident mitigation. However, this Special Test secures ventilation to various primary system areas. During this test the Safeguards Building and Auxiliary Building Central Area will have reduced flow. Ambient temperatures in these areas will be monitored along with Charging Pump temperatures throughout the test for this reason. Ambient temperatures in the Charging Pump cubicles should not exceed 115 °F. The monitoring will identify potential equipment problems. This Special Test may be aborted at any time as directed by Shift Supervisor based on Unit operating conditions (for example, Unit Trip or ESF actuation) or exceeding the given ambient temperature. Failure of Equipment in areas outside the Auxiliary Building Central and Safeguards due to poor ventilation were considered. Other ventilation systems interconnected with the Auxiliary Building Filter Banks are manually secured by emergency procedures. Existing emergency procedures provide sufficient ventilation after a CDA, in event a CDA occurs coincident with the performance of the Special Test. No changes will be made to the Operating License or Tech Specs. Appendix R and the environment will also not be impacted by this Special Test. For these reasons, an unreviewed safety question does not exist.

SAFETY EVALUATION LOG
JCO
2000

S.E. #	Unit	Document	System	Description	SNSOC Date
99-SE-JCO-02 REV. 1	1,2	JCO-C-99-02 (R. 1) 1&2-E-0 1&2-ES-1.3 0-AP-36	HV	Primary ventilation Alignment Following a CDA to assure Filtration of ECCS Leakage - Revised to allow use of the 14-day AOT with 1 unit shutdown	3-15-00

99-SE-JCO-02 Rev 1

Description
JCO-C-99-02 - Revision 1 – Primary Ventilation Alignment Following a CDA to Assure Filtration of ECCS Leakage.
1&2-E-0 – Reactor Trip or Safety Injection, **1&2-ES-1.3** - Transfer to Cold Leg Recirculation, **0-AP-36** – Seismic Event
Station Procedures 1& 2-ES-1.3, "Transfer to Cold Leg Recirculation" and 1&2-E-0, "Reactor Trip or Safety Injection" will be revised to secure all primary ventilation fans (supply and exhaust) and place each exhaust system in "Bypass" with the exception of the Unit 1 and Unit 2 Safeguards, and the Aux. Building Central area, which will all be placed in the Filter Position. The JCO establishes compensatory measures that provide reasonable assurance that the Auxiliary Building Central exhaust system can be manually aligned to the filter position at all times. These compensatory measures will require the power supply and instrument air to the damper controls to be available. The damper controls will also be verified to be operable following a seismic event.

Summary
BACKGROUND
The UFSAR describes the operation of each of the primary ventilation areas to include ability to selectively pass the exhaust systems through the Iodine Filter Loops, as it becomes necessary for decontamination. This includes exhaust from the Auxiliary Building (Central and General Exhaust), Decontamination and Waste Solids Building, Fuel Building, Containment (Shutdown Unit) and Engineered Safety Features Areas. During a design basis accident, the areas that require filtration include the Auxiliary Building Central, and the Engineered Safety Features Areas. Troubleshooting of the Primary Ventilation System (to be documented in ET SE-99-051) identified that various dampers for the Fuel Building, Auxiliary Building, and Unit 2 Safeguards leaked by. This leak-by potentially creates a flow path that bypasses the filter assembly following an accident (LOCA). The most conservative alignment of ventilation systems to ensure that exhaust flow will not bypass the charcoal and HEPA filter assembly is to secure all systems that are not aligned to the filters.

PROPOSED ACTIVITY
EOP procedure revisions are proposed that require alignment of the Aux. Building Central and Safeguards Area ventilation exhaust from these areas through the Filter Loops during a design basis LOCA. Two Safeguards and one Aux. Building Central fan will be in operation. The proposed JCO establishes compensatory actions to ensure the Auxiliary Building Central Ventilation dampers can be aligned to the filters at all times.

For other areas, the associated ventilation fans will be secured during the accident, and their dampers will be placed in the Bypass position. Temperatures in these areas will change according to outside ambient conditions and equipment loading when their respective ventilation fans are secured. Ambient temperatures in these areas are recorded by Operations on rounds periodically. Temperatures outside of normal bounds will cause the Operator to take actions prescribed by existing procedures to correct the condition or declare the associated equipment inoperable. The ambient temperature monitoring instrumentation will still be available.

FAILURE MODES
Failure of the activity is considered failure of a damper in a non-ESF area to position to the Bypass mode with its associated fan secured. If this were to occur, the combined flow provided by the two operating Safeguards ventilation fans and one Aux. Building Central fan would be adequate to meet TS flow requirements and maintain the Safeguards area in a vacuum.

UNREVIEWED SAFETY QUESTION DETERMINATION
Operation of the ventilation system as proposed in this JCO and EOP revisions create no unique precursors or precursor events for Chapter 15 accidents. The proposed JCO and EOP changes do not change the intended operation of the charcoal filter bank or equipment required for accident mitigation. The EOP

procedure revisions are to ensure flow from ESF areas does not bypass the Iodine Filter Loops due to potential leakage past system dampers. Non-ESF area temperatures are monitored by operators, and actions can be taken in accordance with existing procedures to mitigate high or low temperature conditions to prevent environmental challenges to equipment. The ESF Area ventilation system will continue to receive its automatic swap-over to the Iodine Filter Loops. Thus, any leakage of radioactive materials from ECCS equipment in the pump room following a LOCA will be filtered prior to reaching the environment, and the margin of safety as defined in the TS bases is not reduced. Current accident analyses do not take credit for filtration of ECCS leakage for the first hour of the accident. Since all equipment manipulations are performed from the control room, the one-hour limitation does not impose a significant challenge to the operator. For these reasons, an unreviewed safety question does not exist.

Since the activity will ensure the design basis assumptions are satisfied by eliminating potential filter bypass flow paths and ensuring filtration of Safeguards ventilation exhaust during a design basis accident, the activity should be allowed.

SAFETY EVALUATION LOG
MODIFICATIONS
2000

S.E. #	Unit	Document	System	Description	SNSOC Date
95-SE-MOD-82	1,2	DCP 94-295		Provides core drilled penetrations in the walls of station battery rooms 1-I, 1-III, 2-I and 2-III for test equipment cables	12-12-95
98-SE-MOD-08	1,2	DCP 98-002	EE	Anchorage modification of "H" and "J" 4160/480 volt emergency bus transformers	3-25-98
98-SE-MOD-12	1,2	DCP 98-801	EE	Resets MCC feeder overload devices to an improved setting for 10 MCCs	4-10-98
98-SE-MOD-18	1	DCP 98-017	CC	Component cooling water containment cross-tie	5-28-98
98-SE-MOD-28, Rev. 1	1,2	DCP 98-800		Nonregenerative heat exchanger outlet temperature setpoint change	6-16-00
99-SE-MOD-07, Rev. 1	1,2	DCP-99-132		Replacement of refueling cavity seal ring lifting rig	10-06-99
99-SE-MOD-05	1,2	DCP 98-171 & 172		Inspection ports to be added for inspection of the service water to recirculation spray heat exchanger check valves	4-15-99
99-SE-MOD-15	1,2	DCP 98-001		Fuel assembly repair	8-05-99
99-SE-MOD-17	1,2	DCP 99-146	FW	Feedwater pump discharge MOV control modification	8-12-99
99-SE-MOD-21	1,2	DCP 99-148	FW	Permanent installation of thermocouple cards into 2-MUX-21	8-31-99
99-SE-MOD-26	1	DCP 99-154		Replace blowdown manifolds (1-MS-MAN-1494B and C)	12-16-99
00-SE-MOD-01	1	DCP 97-012	FH	Fuel Handling Manipulator Crane Electrical Upgrades	1-21-00
00-SE-MOD-04	1	DCP 00-101		RVLIS Sensor Bellows Reorientation – to preclude air intrusion into the sealed tubing system	3-09-00
00-SE-MOD-05	1	DCP 00-116 UFSAR FN 00-015	RC	Modifies the reactor head vent line to allow installation of the seal ring without angling it to miss the extended portion of the vent line	3-21-00
00-SE-MOD-06	1	DCP 00-120		Replacement of blowdown manifold with two instrument valves in series. The upstream valve of the low side manifold is leaking through (1-RC-FC-1482A).	3-27-00
00-SE-MOD-07	1	DCP 00-121		Replace blowdown manifold 1-RC-MAN-1415B	3-30-00
00-SE-MOD-08	1	DCP 00-122		Replace blowdown manifold 1-RC-MAN-1434C	3-31-00

95-SE-MOD-82

Description

The purpose of the modification is to provide core drilled penetrations in the walls of Station Battery Rooms 1-I, 1-III, 2-I and 2-III, located in the Service Building at elevation 294'-0". During unit outages when battery discharge testing is performed, the battery room doors have to be left open for the cables to connect to the test equipment. Breaching the Battery Rooms, being part of the control room boundary, requires entering a 24 hour action statement per Technical Specification (TS) 3.7.7.1 and allows no movement of fuel. A core drilled hole will provide access for the test cables and at the same time maintain control room boundary integrity without the need to enter the TS action.

Summary

At present some problems and delays are encountered while performing battery discharge testing on the batteries located in the cable spreading room. The test equipment, due to its size, has to be kept outside the battery rooms Technical Specification (and making battery connection by running cables with the door open. Keeping the battery room doors open results in breaching the Control Room Pressure Boundary Envelope and requires entering a 24 hour action of TS 3.7.7.1 since the breach results in the probable inability of the Emergency Ventilation System and the Bottle Air Pressurization System to pressurize the Control Room Envelope to the differential pressure and allows no movement of fuel. Also, environmental qualification (EQ) and fire boundary breaches are created by having the doors open.

To preclude entering the 2 hour action statement of TS 3.8.2.3, utmost care and precaution shall be taken when working around the batteries during modes 1 through 4. These precautions have been addressed in the special implementing requirements of the design change package and also in the construction implementing procedure GMP-C-117. To facilitate battery discharge testing to be performed without breaching the control room pressure, fire, and EQ boundaries, a penetration in the walls of the battery room will be provided for the test cables.

When these penetrations are not used they will be covered up by installing plates on both sides of the walls so as to maintain the design basis of the UFSAR for control room habitability. The outside cover plate will also provide missile protection and maintain the control room EQ boundary.

An unreviewed safety question does not exist because:

1) Control Room Pressure Boundary will be maintained during fuel movement or work over the spent fuel pool is performed in tandem with battery discharge testing.

2) The battery room missile protection is maintained by the cover plates when the penetrations are not used.

3) EQ and fire barriers of the control room boundary envelope will be maintained by the installation of the cover plates over the wall penetrations.

4) TS compliance and UFSAR design basis is being maintained for Units 1 and 2 therefore, the margin of safety is not reduced.

98-SE-MOD-08

Description
The purpose of the change is to enhance the anchorage of the transformers to ensure sufficient margin of safety to withstand a design basis earthquake.

Summary
This change is being implemented to enhance the seismic capacity of the Emergency Bus Transformers by supporting them with additional anchorage. It is noted that the existing anchorage of the transformers is adequate to withstand a design basis earthquake and meets the design and licensing basis criteria for functionality and structural integrity. The proposed modifications will, however, increase the safety margin of the transformer units to withstand a design basis earthquake.

No unreviewed question exists because no impact on the function, reduction in safety margin or change in procedures, operating license, technical specifications, or environmental conditions will result from this modification. The change is only limited to installing anchor bolts or, as an option, weld the existing transformer frames to the embedded channels in the emergency switchgear room (Service Building) and in the rod control drive room (Auxiliary Building). Since none of the areas are radiation areas, no increase in exposure will result.

98-SE-MOD-12

Description
The short time setting on the feeder breaker overload device is reset to an improved setting for 10 MCCs (1-EE-BKR-14H-3, 5,14H1-7, 14J-5, 6 2-EE-BKR-24H-3, 5,24H1-3, 24J-5, 6).

Summary
The major issues considered were the need to assess and take action on breaker coordination for ten of the emergency electrical power motor control center (MCC) feeder breakers. Calculation EE-0395 rev 2 addendum 2F found that when tolerances were applied to the magnetic (instantaneous) settings of the largest breakers in the MCCs, there was insufficient margin between setpoints. As such new setpoints were developed and recommended.

The reason(s) the change should be allowed: It provides an improved coordination margin between the MCC feeder breakers and the individual load breakers fed from the MCC bus.

An unreviewed safety question does not exist because:

1) The implementation of this DCP does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the UFSAR assumes a properly coordinated emergency bus. As such, the reliability of the power supply system is enhanced, by ensuring that the lowest echelon breaker clears a postulated fault.

2) The implementation of this DCP does not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because: the physical configuration of the emergency power supply system is preserved, all loads remain as fed and modeled. The likelihood of a loss of a MCC due to a fault on a load breaker is reduced.

3) The implementation of this DCP does not reduce the margin of safety as defined in the basis of any Technical Specification because the integrity of the emergency power supply system is preserved, if not improved. The bases of the Tech. Specs assume at least one train (following single failure) available of EDG backed power to perform the various safety functions. The margin of safety is preserved by improving the coordination of the emergency power supply.

98-SE-MOD-18

Description
The design change will install an additional Component Cooling (CC) water return line from the PDTT cooler, excess letdown heat exchanger, RHR seal coolers and the neutron shield tank heat exchangers to the CC return header of the 'A' RHR heat exchanger.

Summary
Component Cooling trip valve 1-CC-TV-103B is the containment isolation valve for the CC return line from the Unit 1 'B' RHR heat exchanger. The valve also isolates return cooling water from both trains of several miscellaneous heat exchangers including both RHR pump mechanical seal coolers, PDTT cooler, Excess Letdown heat exchanger and the Neutron Shield Tank coolers. 1-CC-TV-103B must be open for either train of RHR to operate as designed as well as to allow the 'B' RHR heat exchanger to remain in service. Single failure of this CC trip valve in the closed direction will block return CC flow from these various coolers inside containment and could potentially jeopardize the operation of both RHR pumps due to the loss of RHR pump mechanical seal cooling capability. This could render both trains of RHR inoperable.

This design change will mitigate RHR operability concerns by installing a cross connect inside containment from the 6" CC return line of the various containment coolers (6"-CC-161-151-Q3) and tie this cross connect into the 18" 'A' RHR heat exchanger CC return header upstream of 1-CC-TV-103A. This configuration will then allow CC return flow from the containment coolers to be directed to both 18" CC return headers and provide additional assurance of cooling capability for the containment heat exchangers and coolers. RHR operability concerns with respect to 1-CC-TV-103B valve position will be eliminated by the addition of the alternate CC return flow path.

SUMMARY OF SAFETY ANALYSIS

The modification did not constitute an unreviewed safety question as defined in 10CFR50.59 since it did not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in UFSAR.

   The activity does not generate new initiators that would affect the probability of occurrence for analyzed accidents. The CC and RHR systems do not perform a design basis accident mitigation function. Installation of the CC cross tie to the 'A' RHR heat exchanger CC return header will provide an additional flowpath for the miscellaneous containment cooler's CC return in the event that 1-CC-TV-103B fails in the closed direction or the 'B' RHR heat exchanger CC return header is taken out of service. The availability of CC for normal cooldown of the plant during Phase A isolation is not impacted by this design change. Continual flow through the RH seal coolers will ensure RHR pump and system operability. The ability of the component cooling water system to perform its design function will not be affected by this activity.

B) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR.

   No new accident scenarios will be created as a result of this modification. The modification will be installed using qualified materials for the CC system, approved maintenance weld procedures and tested prior to being placed in service. The new cross tie will be included in the ASME XI ISI/IST program. As a result, failure of this cross tie piping that could potentially affect the CC supply/return headers is not considered likely. The modification will not adversely affect the CC system hydraulic characteristics and the cross tie line is seismically designed and located within a missile protected area. Installation of the containment cross tie will increase reliability and flexibility to the CC system. The activity will not affect the CC systems ability to comply with GDC 57 requirements for containment isolation. Accidents or malfunction of equipment of a different type than was previously evaluated is not credible due to nature of the modification.

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

Tech Spec compliance will not be challenged with this activity. Installation of the containment CC cross tie will not reduce the margin of safety of the CC system as described in the Tech Specs.

98-SE-MOD-28, Rev. 1

Description

The setpoint for the nonregenerative heat exchanger temperature control valves, ½-CC-TCV-106/206, is to be changed from 109°F to 105°F. The normal valve line up will change so that the tube sheet inlet and outlet bypass valves are open.

Summary

The setpoint for the nonregenerative heat exchanger temperature control valves, 1-CC-TCV-106 and 2-CC-TCV-206, is being changed from 109°F to 105°F. The normal valve line up is to be changed by opening tube sheet inlet and outlet bypass valves.

The temperature control valves control CC flow through the heat exchanger based on the letdown outlet temperature. The heat exchanger is currently being operated with the letdown discharge temperature lower than the current setpoint. This is due to the CC outlet temperature exceeding its pipe design temperature when the heat exchanger is operated at the setpoint. Opening the bypass valves will cause the heat exchanger to be less efficient resulting in an increase in CC water flow through the heat exchanger. This increased flow with a decreased mass flow rate in letdown will decrease the CC water discharge temperature.

The setpoint is being changed to 105°F because, in the past, it has been difficult to achieve the 109°F letdown discharge temperature while maintaining CC temperature within limits. Currently the station is operating at a setpoint of approximately 100°F. Adjustment of the controls is not desirable as letdown temperature affects the amount of boron removed by the demineralizers. At the same time, letdown temperature has an effect on reactor coolant pump seal water temperature. An increase in seal water temperature, above current operating conditions, may improve seal leakage on the unit 2 "A" reactor coolant pump. This setpoint change and open bypass valves will result in an increase in letdown temperature from current operating conditions to 105°F while maintaining CC temperature within design limits.

The amount of boron removed by the demineralizers decreases with an increase in letdown temperature. Due to the effect on RCS reactivity, this change will be implemented in small increments to decrease unit perturbations. This shall be controlled by the Operations procedure used to implement this change.

UNREVIEWED SAFETY QUESTION ASSESSMENT

The accidents applicable to this change are a LOCA and uncontrolled boron dilution.

1. Accident probability has not been affected. The change in setpoint and valve position will not affect the probability of either a break in the letdown line or failure of the RCP seals. As the actual change to the station will decrease the amount of boron removed by the demineralizers, the probability of uncontrolled boron dilution is not affected.

2. The consequences of an accident will not be affected. The letdown flow path is automatically isolated when there is containment isolation and the letdown portion of the CVCS has no role in accident mitigation.

3. No unique accident probabilities are created. This is a change in setpoint and valve position only. The heat exchanger operating parameters will stay within design limits.

4. Margin of Safety is maintained. Compliance with Technical Specifications for the CVCS and CC systems is not affected.

99-SE-MOD-07, Rev. 1

Description

DC 99-132 will fabricate two new refueling cavity seal ring lifting rigs that will perform the same function as the two existing lift rigs but with a pinned design which can be removed from containment. The old lift rigs will be disposed after the new lift rigs have been used successfully. The new lift rigs may be stored inside containment during power operations. Procedure revisions are needed to show the modified rigging arrangements.

Summary

DC 99-132 will fabricate new refueling cavity seal ring lifting rigs that will perform the same function as the two existing lift rigs but with a pinned design which can be removed from containment. The existing lift rigs are welded assemblies, too large to be removed from containment. This forces 10-year ISI inspections (and potential weld repairs) to be performed inside containment during a refueling outage. New, pinned lift rigs, designed, fabricated, load tested and inspected in compliance with NUREG-0612 commitments, would allow 10-year ISI inspections to be performed during non-outage periods. The old lift rigs will be disposed after the new lift rigs have been used successfully. Procedure revisions are needed to show the modified rigging arrangements.

The review determined there were no accidents previously evaluated in the Safety Analysis Report associated with the Safety Evaluation scope of work. Potential concerns about seismic interaction between the refueling cavity seal ring and its lifting rig versus nearby safety related components during power operations have been resolved as described in the response to Question 46. NUREG-0612 commitments for Special Lifting Devices such as the refueling cavity seal ring lift rig explicitly do not require postulated failure of the lift and evaluation of the ensuing consequences. These Phase II scenarios were rescinded by the NRC via Generic Letter 85-11. Postulated events such as the fuel handling accident will not be affected by the scope of this Safety Evaluation.

The review determined there were no equipment malfunctions previously evaluated in the Safety Analysis Report associated with the Safety Evaluation scope of work. The SAR does not address the postulated failure of the lift rig. The SAR indirectly addresses possible refueling water leakage past the seal ring seals when it is installed for refueling; however, the scope of this Safety Evaluation will have no affect on the ring's ability to maintain a watertight seal.

The review determined the Safety Evaluation scope of work caused no reduction in the margin of safety of any part of the Technical Specifications bases sections and there is no need to change the Technical Specifications. The Tech Spec basis descriptions do not address lifting the seal ring. Requirements address the minimum depth of water over the reactor vessel flange in order to move fuel assemblies. Existing Tech Spec LCOs, basis descriptions, surveillance requirements and margins of safety will be unaffected by the proposed changes.

This Safety Evaluation evaluated a rigging evolution and has no potential to:

- adversely affect the ability of the station to achieve and maintain safe shutdown in the event of a fire, or
- adversely impact the environment.

The proposed change to use a new pinned lift rig for the refueling cavity seal ring does not constitute an unreviewed safety question because it does not:

- increase the probability of occurrence for accidents,
- increase the consequences of accidents,
- create the possibility for an accident of a different type,
- increase the probability of occurrence of equipment malfunctions,
- increase the consequences of equipment malfunctions,
- create the possibility for a malfunction of equipment of a different type,
- reduce any Tech Spec margin of safety,

- require a change to the Operating License or Tech Specs,
- affect the ability of the station to achieve and maintain safe shutdown in the event of a fire, or
- adversely impact the environment.

99-SE-MOD-05

Description
Inspection ports are to be added for inspection of the service water to RSHX check valves. Each port is to consist of a sockolet and a blind flange with pipe as required.

Summary
Inspection ports are to be added to the SW to RSHX lines. The ports are to be used to inspect the SW to RSHX check valves to ensure that they are normally closed. The IST Program requires that the check valves be inspected. Removal of the valves is labor intensive and a visual inspection is an acceptable method of testing. The ports are to include a sockolet, blind flange and a short section of pipe.

Pipe stress and supports were evaluated and found acceptable for all specified loading conditions including seismic.

The accidents considered were those which result in containment depressurization, including LOCA and Main Steam Line Break.

UNREVIEWED SAFETY QUESTION ASSESSMENT

1. Accident probability will not be increased because the recirculation spray heat exchanges are used for accident mitigation only.

2. Accident consequences are not affected. The inspection ports are required to ensure that the check valves are closed. A check valve stuck in the open position could divert water from the RSHX. The resultant flow would still meet system design requirements, per calculation ME-0547, but to maintain margin of flow available the check valves are to be inspected. System leakage, should a port fail, would be bound by this calculation.

3. No unique accident possibilities are created. The inspection ports are basically passive components which will only be used when the unit is shutdown. The service water lines affected are only used after a DBA. System design bases are unchanged.

Margin of Safety is maintained because the integrity and reliability of the system are not affected. The margins of safety as described in the bases of the Technical Specifications are not affected.

99-SE-MOD-15

Description

DCP 98-001 describes the reconstitution of fuel assemblies following N2C12 shutdown. Reconstituting fuel assemblies, also known as fuel repair, is the process of removing fuel rods and replacing them with solid stainless steel filler rods or suitable fueled rods. North Anna Vendor Procedure FP-VRA/VGB-FI1 will control the performance of all fuel repair work, and has provisions for fuel inspections that are beyond the scope of DCP 98-001.

Summary

Fuel reconstitution as described in DCP 98-001 and as performed in vendor procedure FP-VRA/VGB-FI1, "Fuel Inspection and Repair for North Anna Units 1 & 2" is the process of replacing fuel rods in a fuel assembly with solid stainless steel filler rods or suitable fueled rods. This will enable the fuel assembly to be considered for use in subsequent reload core designs whereas fuel assemblies with known defective fuel rods, by administrative policy, cannot be used in subsequent reload cores. The in core use of fuel assemblies reconstituted with filler rods is described and allowed in Sections 4.2.1.2, "Design Description," of the North Anna UFSAR and Section 5.3.1, "Fuel Assemblies," of the Technical Specifications for both North Anna Unit 1 and Unit 2.

Sections 4.2.1.2, 15.4.1.5.4 and Table 4.2-2 of the North Anna UFSAR and T.S. 5.3.1 describe the in core use of reconstituted fuel assemblies. However, the description of reconstituted fuel in the UFSAR describes the replacement of only failed fuel rods with filler rods. If deemed necessary, it may be desired to remove a non-failed fuel rod from a fuel assembly. One example of this is that during the reconstitution of an assembly with a broken rod, it may be required to remove an adjacent non-failed fuel rod in order to use a fiberscope to verify broken rod removal tool engagement on the broken rod. It is not prudent to place the non-failed rod back in the fuel assembly, as it may be susceptible to fuel rod fretting as a consequence of inadvertent grid cell damage during use of the fiberscope. Fuel assemblies that have had non-failed or undamaged fuel rods replaced with filler rods are restricted from in core use until such time as the UFSAR changes can be processed. Reload cores using reconstituted fuel assemblies continue to require cycle specific evaluation to confirm that the exact configuration of the reconstituted assemblies do not introduce a change in radial gradients in the flow and enthalpy distribution that could invalidate the applicability of the CHF correlation. It is also confirmed that DNB analysis modeling with a regular fuel assembly bounds the reconstituted fuel assembly. Note that fuel assemblies that have been repaired but continue to contain 264 fueled rods meet the UFSAR and Technical Specifications definitions of fuel assemblies, not reconstituted fuel assemblies.

To perform certain reconstitution operations, temporary replacement of the new fuel elevator basket with a functionally equivalent Westinghouse basket is required. The basket replacement will be implemented separately from this Design Change via a procedurally controlled temporary modification in accordance with VPAP-1403, "Temporary Modifications". Installation, functional checkout, and removal of the replacement basket are controlled by North Anna Vendor Procedure 0-FH-FEB-001, "Temporary New Fuel Elevator Basket Replacement". Safety Evaluation 95-SE-PROC-20 was performed for vendor procedure 0-FH-FEB-001 when the procedure was first used at North Anna.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased as a result of this change. The assumptions used in the analysis for the fuel handling accident in the fuel building remain bounding for handling irradiated fuel in the elevator basket. All fuel handling will be performed in accordance with fuel handling procedures. Fuel repair and inspection procedures will require the mechanical stop to be installed whenever irradiated fuel is inserted in the fuel elevator basket. This will assure that a minimum of 7 feet of water shielding exists over an irradiated fuel assembly in accordance with the criterion from UFSAR Sections 9.1.4.6.4 and 12.1.2.5 with the new fuel elevator in the full up position. The handling of fuel assemblies or individual fuel rods can only be accomplished with the fuel at the approximate height of fuel in the storage racks (new fuel elevator full down). The fuel rod handling procedure requires a sling at least 9 feet long to be rigged to the fuel handling crane hook with the other end attached to the fuel rod handling tool. This limits the upward travel of the fuel rod handling tool in case of an inadvertent lift. Because of these procedural and physical limitations, an individual fuel rod cannot be raised to an unsafe elevation when using the individual fuel rod handling tool.

Reconstituted fuel assemblies meet the same design criteria and requirements and perform the same function as non-reconstituted fuel assemblies with similar operating history. Fuel handling interfaces for reconstituted fuel assemblies do not change as a result of this activity. Therefore, no mechanism is introduced which would increase the probability or consequences of a Chapter 15 accident. The impact of using replacement fuel rods or solid stainless steel filler rods on nuclear performance is assessed during the reload design process to ensure that peaking factor limits are not violated. These assemblies are evaluated in the same manner as non-reconstituted fuel assemblies during the reload design process to ensure that none of the reload limits are violated. Therefore, the use of reconstituted fuel assemblies will not increase the probability or consequences of any of the UFSAR Chapter 15 accidents.

The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not increased. The only equipment that will be used for this change is the fuel handling tool and fuel elevator with the replacement fuel elevator basket. The basket is a temporary replacement and is functionally equivalent to the original equipment basket. Only one fuel assembly will be handled at any one time, thus the consequences of an accident are bounded by the fuel handling accident outside containment as described in Section 15.4.5 of the UFSAR.

The margin of safety as defined in the basis for any Technical Specification is not reduced by the repair process or by the use of repaired fuel assemblies. The safety and design limits will not change as a result of using reconstituted fuel. All safety and design limits will continue to be confirmed as part of the reload safety evaluation process. The replacement basket is functionally equivalent to the original equipment basket with the exception that the replacement basket accommodates fuel assembly reconstitution. The fuel assemblies are designed so that reconstitution is possible. No restrictions preclude the handling of irradiated fuel in the fuel elevator basket, and the basket has been used to hold irradiated fuel in the past. Fuel repair procedures will require the mechanical stop to be installed whenever irradiated fuel is handled in the fuel elevator basket to prevent lifting the fuel assembly too close to the surface of the SFP. All fuel handling will be performed in accordance with site procedures.

99-SE-MOD-17

Description
DCP 99-146 provides automatic redundant isolation for the main feed regulation bypass valves (MFRBV) by installing a trip signal directly to the feedwater pump discharge MOVs, 1-FW-MOV-150A, B, C, 2-FW-MOV-250A, B, C control circuits.

Summary
The feedwater pump motors, 1-FW-P-1A1, 1A2, 1B1, 1B2, 1C1, 1C2, 2-FW-P-1A1, 1A2, 1B1, 1B2, 1C1 & 1C2 receive a trip signal from the Solid State Protection System (SSPS), master relay K621 via slave relay K508 relay. This "Trip Turbine & Main Feedwater Pumps" trip signal is initiated as a result of a "Hi-Hi SG Level" or "SI". The pump motor trip in turn initiates the MOV closure for those valves associated with the operating pumps via the relay control logic based on the breakers changing position (close to open). The valve associated with the pump in standby remains open. This open valve provides a flow path to the steam generators during bypass valve manipulation if the main feedwater regulating bypass valve (MFRBV) is unable to close. This condition had not previously been identified and was incorrectly described in the UFSAR, SDBD and NCRODP. It was incorrectly believed that all three MOVs closed on the feedwater pump trip. Potential Problem Report PPR-99-024 identified this condition and Deviation Report N-99-1415 was submitted. As a result, Standing Order #227 was instituted to have the discharge MOV associated with the pump in Standby closed during bypass valve manipulation.

This modification involves directly providing a closure signal to the MOVs' control circuits from SSPS. The "Trip Turbine & Main Feedwater Pumps" trip signal originates in the SSPS, panel 1/2-EI-CB-47F located in the Instrument Rack Room. The MOV control circuit relay contacts are located in Auxiliary Relay Panel "G" (AR-G), 1/2-EP-CB-28G, also located in the Instrument Rack Room. Relay K621, Train "B", has a spare contact that is internally wired to a terminal block. This is the same relay that provides the trip signal to the feedwater pumps. A cable will be terminated to the field side of this terminal block and routed from 1/2-EI-CB-47B to a spare auxiliary relay in 1-EP-CB-28G and a new relay (to be installed) in 2-EP-CB-28G. From these relays, normally open contacts will be wired into the control (close) circuit of each MOV. A closure signal closes the relay contacts, closing the valves. The existing relay configuration associated with the pump trips is not being modified and will continue to actuate the valves based on motor trip.

This single train signal design for the pump discharge MOVs is consistent with MFRV backup valves, 1/2-FW-MOV-154/254A, B & C which also receive a signal to close. These MOVs receive a Train "A" signal only. The non-safety classification (NSQ) of 1/2-FW-MOV-150/250A, B, C complies with NUREG-0138, Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff. NUREG-0138 states "The staff believes that it is acceptable to rely on these non-safety grade components in the steam and feedwater systems because their design and performance are compatible with the accident conditions for which they are called upon to function. It is the staff position that utilization of these components as a backup to a single failure in safety grade components adequately protects the health and safety of the public."

The introduction of the "Trip Turbine & Main Feedwater Pumps" closure signal into the control circuits of the feedwater pump discharge valves does not increase the probability of an accident. This modification ensures feedwater isolation is achieved for all discharge MOVs upon initiation of the closure signal without dependence on the actions, condition or status of other components.

Integrating a closure signal into the control circuits of the discharge MOVs does not increase the consequences of an accident. This closure signal to the feedwater pump discharge MOVs is providing greater assurance feedwater isolation will be achieved in the event of a rupture of the secondary system downstream of the MFRBV.

Modifying the control circuits of the feedwater pump discharge MOVs to close on a SSPS trip signal does not create the possibility of an accident not previously analyzed. This modification will ensure isolation at the feedwater pump discharge MOVs is achieved in the unlikely event a MFRBV fails to close during a rupture of the secondary system downstream.

The Margin of Safety is maintained or increased as a result of this modification. Prior to implementation of this modification, the possibility of a failed MFRBV in conjunction with a rupture of the secondary system downstream of the valve would have resulted in continued flow to the steam generators via the open discharge valve. This modification will ensure valve closure and system isolation.

99-SE-MOD-21

Description
DCP-99-145 makes permanent a Temporary Modification (TM N2-1128). This involves replacement of buffer amplifier cards with thermocouple amplifier cards for three feedwater temperature computer inputs. DCP 99-148 makes these card changes via DCP, no TM involved.

Summary
This activity does not involve any physical modification to the facility. The new thermocouple amplifier (TC) cards (installed by TM N2-1128) are manufactured by the same company as the buffer amplifier (BA) cards, and they are designed to fit the same slots. Bench testing and the performance since having been installed by TM has shown that the TC card has a more stable output than the BA card. The affected cards send a MFW temperature signal to the plant computer system (PCS) and emergency response facility computer system (ERFCS) only. The signal to the P-250 is not affected. Thus, the P-250 FW flow calorimetric is not affected by this activity.

Operations department calorimetric procedures currently "auctioneer" to the most conservative (or highest power) calorimetric indication. Currently the Unit 1 and Unit 2 calorimetrics using their PCS are the highest, thus they are used as the official indication. Since the accuracy of the calorimetric is in question due to the sensitivity of the BA cards to instrument drift, this condition may be requiring an unnecessary reduction in unit electrical output.

Failure of the activity, for the near term, is bounded by the evaluations performed for the FW flow calorimetric performed under 99-SE-MOD-01. Additionally, the PCS indications of FW temperature or FW flow calorimetric will not be adversely affected. This has been proven empirically by comparing the results obtained with the new cards vice U-1 results using the old (pre-modification) cards. Thus, there is no adverse affect on nuclear safety. No new accidents are created, and consequences of analyzed accidents are not affected. There is no reduction in the margin of safety or ability to mitigate accidents. For these reasons, an unreviewed safety question does not exist.

Since the activity will install amplifier cards in the circuit that are better suited for the application and result in a more accurate FW flow calorimetric, unnecessary reductions in unit electrical output may be eliminated. Therefore, this activity should be allowed.

99-SE-MOD-26

Description
The "C" steam generator main steam flow transmitter, 1-MS-FT-1494 is equipped with high and low side two valve drain manifolds. The valves for both manifolds have become difficult to operate. Each manifold is to be replaced with two isolation valves in series.

Summary
The "C" steam generator main steam flow transmitter, 1-MS-FT-1494 is equipped with high and low side two valve drain manifolds. The valves for both manifolds are difficult to operate. The Hoke manifold and repair parts are no longer available. The existing two valve manifolds are equipped with test taps which are not used. As the test taps are not being used, each manifold will be replaced with two instrument valves.

The flow transmitter is part of the reactor protection system. It provides a trip signal when high main steam flow is detected. There are two flow transmitters for main steam line with reactor trip criteria being a trip signal from one out of the two transmitters coincident either low steam line pressure or low-low T_{avg} and high steam line differential pressure signals. The operability of the reactor protection system and engineered safety feature actuation are addressed in Technical Specifications 3.3.1.1 and 3.3.2.1 which require an inoperable trip circuit to be placed in the trip condition within 1 hour of discovery.

The accident considered to be applicable for the manifold replacement is a major secondary pipe rupture.

UNREVIEWED SAFETY QUESTION ASSESSMENT

1) Accident probability will not be increased. The valves are being installed in accordance with all applicable specifications. The transmitter detects high flow conditions in order to mitigate an accident by tripping the reactor and protecting it from loss of heat sink and does not affect the probability for an accident.

2) Accident consequences are not affected. The new valves meet all of the specifications, requirements, and codes which were required for the original. They will operate in the same manner as the original two valve manifolds.

3) No unique accident possibilities are created. The manifold replacement will not affect the operation of the main steam system or the reactor protection system. System design bases are unchanged.

4) Margin of Safety is maintained because the integrity and reliability of the main steam system is unchanged.

00-SE-MOD-01

Description
On the Fuel-Handling Manipulator Crane in Containment, the Control Panel, the Motor Control Center (MCC) and related cables and controls will be replaced. The new equipment will allow the Control Panel and MCC to be removed from Containment between refueling outages. The new MCC contains a modern speed control (frequency drive). The new Control Panel contains a Programmable Logic Controller (PLC).

Summary
The Fuel Handling Manipulator Crane is used at North Anna to move fuel assemblies inside of the containment. Because of the harsh environment in that location, the existing electrical equipment and wiring have demonstrated significant deterioration and have required regular repairs. This Design Change is to remove the Control Panel and Motor Control Center from the crane and replace them with equivalent, modern electronic devices. The new Control Panel and Motor Control Center will be removable from the containment between refueling outages.

The new Motor Control Center will use a variable frequency drive instead of the thyristor drive circuit in use now. The new Control Panel will use a Programmable Logic Controller. The connecting cables between the two panels and the festoon cable will also be replaced. The purpose of these changes is to improve reliability and reduce mechanical deterioration.

These changes should be allowed.

Modifications to the Fuel Handling Manipulator Crane will not increase the probability or consequences of a fuel-handling accident in containment since no moving parts are being replaced on the trolley or hoist and the control function is the same as before, with improved and tested equipment. Furthermore, the crane is non-safety related (NSQ for Seismic). The electrical power and control circuits being changed by this Design Change will not change the probability or consequences of the postulated fuel-handling accident. The logic for the new system is essentially the same as the existing system. The physical changes to the crane that are proposed here are not reflected in the operation of the crane as it relates to that accident. The integrity of the crane structure (including seismic requirements) is not being adversely affected. Since the Manipulator crane is only used when the reactor is shut down and stable, there are no additional accidents made possible by the design changes proposed here.

The greatest concern during refueling is fuel failure. The changes to the Manipulator Crane proposed here do not have an effect on fuel-cladding failure. The control panel and MCC will be installed when the reactor head is protecting the fuel. The Control Panel and MCC will be seismically restrained during the time the reactor head is removed (although this is not a prerequisite for the work of this DCP since the work is not performed over the reactor). The functions of the crane include detecting failed fuel cladding and moving the fuel. Neither of these functions is affected by this DCP.

The Manipulator Crane will have greater reliability as a result of these changes, but these changes will not have any direct impact on the integrity of the fuel. Thus, they will not cause a failure of equipment not previously analyzed. Only the electrical power and control circuits and related equipment are being replaced, and the structural integrity will not be reduced. The Margin of Safety in the Tech Specs will not be changed by this Design Change. No Tech Spec, UFSAR or Operating License changes are required.

00-SE-MOD-04

Description

The "A" and "B" train reactor head sensor bellows assemblies in the reactor vessel level instrumentation system (RVLIS) will be inverted such that the capillary connections are reoriented from the top to the bottom of the sensor assemblies in order to preclude air intrusion into the sealed tubing system.

Summary

Westinghouse Technical Bulletin TB-101R1 "RVLIS Calibration Anomalies Due to Air Inleakage" reported that at several plants, recalibration of the reactor vessel level instrumentation system during refueling shutdowns have indicated that air inleakage into the sealed portion of the system have caused errors in readings and inaccurate calibrations. In almost all cases, air was found in the section of tubing from the reactor vessel head sensor and the operating deck. Westinghouse determined that when the sensor is disconnected from the reactor for refueling, the sensor bellows is exposed to atmospheric pressure, and the water in the tubing above this elevation is below atmospheric pressure. There are three locations with mechanical connections, having the potential for inleakage: the fill valves at the head connection and operating deck, and the bellows or its o-ring seal in the head sensor.

To prevent possible air inleakage through the sensor bellows, o-ring seal, and reactor head connection fill valve, Westinghouse recommends that the vessel head sensor be inverted so that the capillary tubing connection is on the bottom. During refueling, the bellows and seal would then be exposed to a positive pressure and could be covered with water to block air inleakage. Also, air trapped in the bellows could not reach the tubing connection at the bottom of the bellows. The modification also moves the fill valve at the sensor to a lower elevation, resulting in a positive pressure at this potential leakage location. In order to accomplish the sensor inversion, the existing capillary tubing will be cut and additional tubing added. Westinghouse reports that they have not been advised of any air inleakage problems where the sensors were installed in the inverted position.

The reorientation of the RVLIS reactor head sensor bellows assemblies does not create an unreviewed safety question. The operation and function of the RVLIS system is not affected. The sensor bellows assemblies are mechanical pressure boundary separation devices that are designed to operate in any position. The design and installation of the new tubing extension pieces is consistent with the original system design requirements. Thus, this design change does not affect any previously evaluated accidents or create any new accidents of a different type.

In accordance with Technical Specifications 3.3.3.6, the new reorientation of the RVLIS sensor assemblies will be performed during a refueling outage when the RVLIS system may be removed from service for maintenance.

00-SE-MOD-05

Description
During installation of the reactor cavity seal ring, the seal ring must be manipulated to miss the protrusion of the reactor head vent line. The vent line is oriented to direct discharged fluid towards the fuel transfer canal area. The piping will be modified to provide clearance for seal ring installation and removal. The piping modification will direct discharged fluid to the fuel transfer canal/refueling cavity area without impingement hazard. Modification to a section of non safety related pipe in the reactor head vent line does not affect system operation or performance, does not change any design requirements and maintains vent function capability.

Summary
During installation of the reactor cavity seal ring, the seal ring must be manipulated to miss the protrusion of the reactor head vent line. REA 2000-006 requested to modify the reactor head vent line to allow installation of the seal ring without angling it to miss the extended portion of the vent line.

The reactor head vent line extends out from the seismic platform and is oriented to direct discharged fluid towards the fuel transfer canal area. A short section of non-safety related reactor head vent piping down stream of the double isolation valves will be removed. The piping will be modified to provide clearance for seal ring installation and removal. The piping modification will direct the vent discharge to the fuel transfer canal/refueling cavity area without impingement hazard.

Modification to a section of non safety related pipe in the reactor head vent line does not affect system operation or performance, does not change any design requirements and maintains vent function capability.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

A) Accident probability of occurrence or consequence has not been increased because the design change conforms to system design requirements and the requirements of the applicable codes and standards. The piping upstream of the double isolation valves is within the reactor pressure boundary. Modification of the open-ended non-safety related vent pipe is downstream of the double isolation valves and is not within the reactor coolant pressure boundary. The piping modification does not change the probability of occurrence or consequence of a reactor coolant pressure boundary leak.

B) No unique accident probabilities or malfunctions of a different type are created. The piping upstream of the double isolation valves is within the reactor pressure boundary. Reconfiguration of the open ended non-safety related vent pipe downstream of the double isolation valves does not increase or create the probability for an accident of a different type. The reactor head vent system consists of piping components and solenoid isolation valves. The modification is to piping components only and does not increase in the probability of a malfunction or create a different type of equipment malfunction.

C) Margin of Safety is maintained because the design meets system requirements and the requirements of applicable codes and standards. There is no change in the performance or operation of the reactor head vent system. Reconfiguration of the open ended non-safety related vent pipe downstream of the double isolation valves does not affect or change to the Operating License or Technical Specifications.

00-SE-MOD-06

Description

The reactor coolant bypass flow controller 1-RC-RC-1481A is equipped with two valve drain manifolds. The upstream valve of the low side manifold is leaking through. The manifold is to be replaced with two instrument valves in series.

Summary

The reactor coolant bypass flow controller 1-RC-RC-1481A is equipped with two valve drain manifolds. The upstream valve of the low side manifold is leaking through. The manifold is to be replaced with two instrument valves in series. The upstream manifold valve is leaking through. The existing two valve manifold is no longer made and the test tap is not used. The manifold is not required and will replaced with two valves in series.

The flow controllers measure the flow in the reactor coolant bypass loops which provide part of an interlock to the loop stop valves required by Technical Specification 3.4.1.5. The controller provides a signal preventing the opening of a cold leg loop stop valve unless the hot leg loop stop valve and bypass valve have been fully open and flow has been greater than 125 gpm for 90 minutes. This interlock is to prevent a reactor transient which could occur because the isolated loop boron concentration is lower than that of the operating loops. The controller is only required during unit start up and in modes 1 and 2, the power to the loop stop valves is required to be removed by locking out the breakers.

The instrument valves used to fabricate the manifold will meet all of the specification and requirements of the original equipment and will have welded connections. No unique accident possibilities are created. The manifold replacement will not affect the operation of the loop stop valves or the RCS. System design bases are unchanged.

UNREVIEWED SAFETY QUESTION ASSESSMENT

1) Accident probability will not be increased. The valves are being installed in accordance with all applicable specifications and the probability for leakage is not changing.

2) Accident consequences are not affected. The new valves meet all of the specifications, requirements, and codes which were required for the original. They will operate in the same manner as the original two valve manifold.

3) No unique accident probabilities are created. The design function of the manifold is not changing. The manifold acts as a system pressure boundary only. The operation of the controller is not affected and the interlocks to the loop stop valve will still be provided.

4) Margin of Safety is maintained because the integrity and reliability of the RCS system is unchanged and compliance with Technical Specifications is not affected.

00-SE-MOD-07

Description
The "A" reactor coolant loop flow transmitter, 1-RC-FT-1415 is equipped with two valve drain manifolds. The high side manifold valve, 1-RC-ICV-3199, has a packing leak which can not be repaired. The manifold is to be replaced with two isolation valves in series.

Summary
The "A" reactor coolant loop flow transmitter, 1-RC-FT-1414 is equipped with two valve drain manifolds. The low side manifold shows signs of leakage. The Hoke manifold and repair parts are no longer available. The existing two valve manifold is equipped with a test tap which is not used. As the test tap is not being used, the manifold will be replaced with two instrument valves.

The flow transmitter is part of the reactor protection system. It provides a trip signal when low RCS flow is detected. There are three flow transmitters for each loop with trip criteria being a trip signal from two out of the three transmitters. The operability of the reactor protection system is addressed in Technical Specification 3.3.1.1 which requires an inoperable trip circuit to be placed in the trip condition within 1 hour of discovery. The trips are to be operable in mode 1 with different permissives in effect depending on reactor power levels.

The accidents considered to be applicable for the manifold replacement are partial loss of reactor coolant, loss of forced reactor coolant flow, and single RCP locked rotor. These are the accidents which may result in a trip from low RCS flow. A small break LOCA was also considered as the safety function of the valve manifold is as a system pressure boundary.

UNREVIEWED SAFETY QUESTION ASSESSMENT

1) Accident probability will not be increased. The valves are being installed in accordance with all applicable specifications and the probability for leakage is not changing. The transmitter detects low flow conditions in order to mitigate an accident by tripping the reactor and does not affect the probability for an accident.

2) Accident consequences are not affected. The new valves meet all of the specifications, requirements, and codes which were required for the original. They will operate in the same manner as the original two valve manifold.

3) No unique accident possibilities are created. The manifold replacement will not affect the operation of the loop stop valves or the RCS. System design bases are unchanged.

4) Margin of Safety is maintained because the integrity and reliability of the RCS system is unchanged.

00-SE-MOD-08

Description

The "C" reactor coolant loop flow transmitter, 1-RC-FT-1434 is equipped with two valve drain manifolds. The low side manifold valve, 1-RC-ICV-3027, has a packing leak which can not be repaired. The manifold is to be replaced with two isolation valves in series.

Summary

The "C" reactor coolant loop flow transmitter, 1-RC-FT-1434 is equipped with two valve drain manifolds. The low side manifold has a packing leak that can not be repaired. The Hoke manifold and repair parts are no longer available. The existing two valve manifold is equipped with a test tap which is not used. As the test tap is not being used, the manifold will be replaced with two instrument valves.

The flow transmitter is part of the reactor protection system. It provides a trip signal when low RCS flow is detected. There are three flow transmitters for each loop with trip criteria being a trip signal from two out of the three transmitters. The operability of the reactor protection system is addressed in Technical Specification 3.3.1.1 which requires an inoperable trip circuit to be placed in the trip condition within 1 hour of discovery. The trips are to be operable in mode 1 with different permissives in effect depending on reactor power levels.

The accidents considered to be applicable for the manifold replacement are partial loss of reactor coolant, loss of forced reactor coolant flow, and single RCP locked rotor. These are the accidents which may result in a trip from low RCS flow. A small break LOCA was also considered as the safety function of the valve manifold is as a system pressure boundary.

UNREVIEWED SAFETY QUESTION ASSESSMENT

1) Accident probability will not be increased. The valves are being installed in accordance with all applicable specifications and the probability for leakage is not changing. The transmitter detects low flow conditions in order to mitigate an accident by tripping the reactor and does not affect the probability for an accident.

2) Accident consequences are not affected. The new valves meet all of the specifications, requirements, and codes which were required for the original. They will operate in the same manner as the original two valve manifold.

3) No unique accident possibilities are created. The manifold replacement will not affect the operation of the loop stop valves or the RCS. System design bases are unchanged.

4) Margin of Safety is maintained because the integrity and reliability of the RCS system is unchanged.

SAFETY EVALUATION LOG
TEMPORARY MODIFICATIONS
2000

S.E. #	Unit	Document	System	Description	SNSOC Date
00-SE-TM-01	1,2	N1-1679	LW	Swaps leads at 1-EI-CB-44, TB 22 & 23, and at the local junction box for 1-LW-FT-103, low level waste to clarifier flow, to allow the ground (discovered on high (signal) side conductor of the FT circuit) to be transferred to the low (return) side of the circuit until cable can be replaced	3-10-00
00-SE-TM-02	1	N1-1680	SI, RH	Installs a temporary hose between an SI accumulator vent & a drain off of the RHR relief valve discharge line to provide an additional controlled source of nitrogen to blanket the PRT during RCS draindown.	3-13-00
00-SE-TM-03	1,2	N1-1681	SW	Installs a new relief valve on the SW air compressor receiver tank with a lower lift setpoint (118 psig) to prevent overpressurization of the tank	6-01-00
00-SE-TM-04	1	N1-1682	MS	Returns the reheater power supply output to normal levels to allow removal of the manual override on the reheat FCVs. This TM allows continued operation with any combination of PC1, PC2, and/or PC3 cards removed.	6-21-00
00-SE-TM-05	1	N1-1683 1-TMOP-43.1 1-TOP-43.2		Addresses installation of a mechanical jumper to bypass the blower (1-GM-BLO-1B) for U1 main generator hydrogen dryer 1-GM-D-1B to evaluate if gas flow is achievable if the blower were removed from the hydrogen gas dryer system.	6-22-00
00-SE-TM-06	1,2	N1-1688 N2-1131	BC	Jumpers out the circuitry for the fire protection trip & the vibration switch trip of the bearing cooling fans due to a malfunction within the BC fan trip circuitry associated with the FP systems or vibration detection equipment.	8-17-00
00-SE-TM-07	1	N1-1685	DA	Allows testing of the incore sump high level switch (1-DA-LS-106). Annunciator 1J-C8 is locked in.	8-17-00
00-SE-TM-08	1	N1-1687	DA	Disconnect the patch cord (795, 1J59C) to remove the hi-hi input to the annunciator. The hi-hi alarm is alarming spuriously (1-DA-LS-106). Defeating the hi-hi alarm will restore annunciator 1J-C8 to a condition which will alert the operator to a high level alarm in the incore room sump.	8-17-00
00-SE-TM-09	1,2	TM N1-1689	SW	A ground was discovered on the positive lead (black) of the FT circuit causing indication failure. Rolling leads at 1-EI-CN-16A, TB 13 & 14, and at local junction box for 1-SW-FT-103, will allow the ground to be transferred to the negative (white) side of the circuit.	8-22-00
00-SE-TM-10	1	TM N1-1686		Lifts leads at 1-GM-TS-102B to disable the input to annunciator K-B7 because the temperature switch alarms prior to the setpoint & causes annunciator K-B7 to lock in on hot days.	8-22-00
00-SE-TM-11	1,2	TM N2-1684	BC, CD	Installs a temporary seal water supply to 2-BC-P-5A and 2-BC-P-5B.	8-22-00

SAFETY EVALUATION LOG
TEMPORARY MODIFICATIONS
2000

S.E. #	Unit	Document	System	Description	SNSOC Date
00-SE-TM-12	1,2	TM N1-1690 TM N2-1131 1&2-TMOP-43.1 1&2-TOP-43.2		Installation of a mechanical jumper to bypass the blowers for Units 1 & 2 main generator hydrogen dryers 1-GM-D-1B & 2-GM-D-1B to evaluate gas flow achievable if the blower were removed from the hydrogen gas dryer system.	9-05-00
00-SE-TM-12 **REV. 1**	1,2	TM N1-1690 TM N2-1131 1&2-TMOP-43.1 1&2-TOP-43.2		Installation of a mechanical jumper to bypass the blowers for Units 1 & 2 main generator hydrogen dryers 1-GM-D-1B & 2-GM-D-1B to evaluate gas flow achievable if the blower were removed from the hydrogen gas dryer system. -REV. 1 states that the changes will be made permanent by DCP 95-282 (U1) & DCP 95-283 (U2)	12-28-00
00-SE-TM-13	2	N2-1133	FW	Routes temporary cable from the output of isolation card C3-434 from flow transmitter 2-FW-FT-2477 to test cables in the rack room.	9-14-00
00-SE-TM-14	1,2	N1-1691	BC	Installation of a locking device to maintain 1-BC-MOV-125 and/or 2-BC-MOV-225 in the open position during lake to lake operation.	9-28-00
00-SE-TM-15	1,2	0-MCM-0400-18	BC	Installation of a locking device to maintain BC MOVs open during lake to lake operation.	9-29-00
00-SE-TM-16	1	N1-1692	BC	Installs an electrical jumper to bypass the water level permissive normally required from 1-VP-LS-110A to allow "A" bearing cooling pump to start due to a loss of BC	9-30-00
00-SE-TM-17	2	N2-1134	MS	Allows removal of one or more of the associated system circuit cards (PC1, PC2, PC3) from 2-EI-CB-30 to return the reheater power supply output to normal levels.	10-05-00
00-SE-TM-18	1,2	N2-1135	CD	Installs a temporary RTD on the mechanical chiller oil pump discharge piping to monitor oil temperature until the failing bearing temperature sensor problem is resolved.	11-06-00
00-SE-TM-19	2	N2-1136	VB	Lifts lead from the failing flow switch on vital bus inverter 2-VB-INV-04 to annunciator 2H-A4 to defeat the trouble alarm	11-22-00
00-SE-TM-20	1	N1-1693	GM	Installs a jumper across the bad leg of the transmitter switch for 1-GM-PT-112 to restore hydrogen pressure & purity instrumentation	11-22-00

00-SE-TM-01

Description
Temporary Modification
Swap leads at 1-EI-CB-44 TB 22 and 23 and at local junction box for 1-LW-FT-103, low level waste to clarifier flow.

Summary
A ground was discovered on the high (signal) side conductor of the FT circuit causing indication failure of 1-LW-FT-103, Low Level Waste to Clarifier Flow. The Temporary Modification (TM) involves swapping leads at 1-EI-CB-44 TB 22 and 23 and the local junction box for 1-LW-FT-103. Swapping the leads will allow the ground to be transferred to the low (return) side of the circuit, which is normally grounded. This action will restore the circuit to a functional condition, and allow it to be calibrated and returned to operable status. When calibrated, the transmitter should perform as expected with no spurious signals.

1-LW-103 is not required by VPAP-2103, which describes the technical requirements of the Liquid Waste System. The proposed TM will not affect any of the systems that are described in either the UFSAR or VPAP-2103, nor will it affect the surveillance requirements of the Liquid Waste system.

The TM will be installed and remain in place until the associated cable can be replaced. This long term corrective action will return the cable and circuit to original design conditions. Use of the temporary arrangement is considered acceptable in the short term to restore the channel to operable status. The long term corrective action (cable replacement) will eliminate any degraded condition associated with the existing cable. These short term and long term corrective actions are compliant with Generic Letter 91-18 (Information on resolution of degraded and nonconforming conditions) guidance regarding the treatment of component operability and restoration of qualification. As previously mentioned, the short term action will allow the channel to be returned to a functional condition, calibrated, and returned to operable status.

00-SE-TM-02

Description
Temporary Modification TM-N1-1680
Install a temporary hose between an SI accumulator vent and a drain off of the RHR relief valve discharge line.

Summary
It is desired to provide an additional controlled source of nitrogen to the PRT to provide a slight overpressure to the RCS as part of the normal RCS draindown from 28% to 74 inches. The proposed Temporary Modification will use a hose rated for at least 100 psig to supply nitrogen from a SI accumulator vent to the RHR relief valves discharge line to the PRT. This will allow the control room operator to control RCS overpressure by opening the pressurizer PORVs and controlling the makeup flow of nitrogen to the SI accumulator with its supply HCV.

Personnel safety will be maintained by maintaining the nitrogen supply pressure from accumulator at approximately 50 psi, and the hose will be physically restrained at the connections. In addition, a check valve will be provided on the jumper discharge side. This will prevent the hose from whipping and limit the amount of radioactive gas that could be released from the PRT if the jumper hose were to be cut.

Equipment safety is provided by one PZR PORV blocked open and the PRT rupture disc. The nitrogen pressure to the RCS will be limited too less than or equal to 50 psig. This pressure will provide a back pressure to the RHR relief valves. However, as discussed above, other RCS/RHR overpressure protection is in place.

An unreviewed safety question is not created because:
(1) The probability of an accident or malfunction previously evaluated in the SAR occurring is not increased. The TM does not introduce any accident initiators. The unit is shutdown and will be in Mode 5 while this TM is active.
(2) The consequences of any accident or malfunction previously evaluated in the SAR are not increased. No fission product barriers are compromised by this Temporary Modification (TM). The unit is shutdown and will be in Mode 5 while this TM is active.
(3) The possibility of creating a new accident or malfunction has not increased. The TM will be installed by qualified personnel and using appropriate safety guidelines. The control room operator will have control of the nitrogen supply via the SI accumulator makeup HCV. A jumper hose rupture does not breach the RCS boundary because of the installed check valve.

Because the TM is not an undue risk to personnel safety or reactor safety, and because the TM will help ensure that the outage will not be unduly delayed, (remember that a shorter outage is a safer outage as long as it is properly planned), this Temporary Modification should be allowed.

00-SE-TM-03

Description

Temporary Modification TM-N1-1681

Due to corrosion, the wall thickness of the bottom head of the Service Water Air Compressor Receiver Tank was found to be less than the code allowable for the 150 psig design pressure rating of the air receiver. The maximum allowable pressure in the vessel for the minimum wall thickness found during UT was calculated in accordance with ASME Section VIII to be 119.78 psig. To prevent overpressurizing the vessel, a new relief valve with a relief setpoint of 118 psig will be installed.

Summary

The existing Service Water Air Compressor Receiver Tank relief valve has a setpoint of 150 psig which is the same as the rated design pressure of the tank as stated in UFSAR Table 9.2-4. The new relief valve will have a setpoint of 118 psig. Section 9.2.1.2.4 of the UFSAR states that the SW Air Compressors operate to provide 100 psig air to the receiver tank where it is stored for use by the traveling water screen differential level control system and the SW Reservoir level indicating and alarm system. One compressor starts when the air receiver tank pressure drops to 75 psig and the other starts when the receiver pressure drops to 50 psig. The System Engineer has indicated that the lead compressor actually operates between 75 and 90 psig and the maximum pressure that the compressors provide is 100 psig. Installation of a new relief valve with a setpoint of 118 psig will therefore not affect the operation of the SW Air System. The new relief valve will be the same size as the existing relief valve and have the same relieving capacity. The only difference in the valves will be the spring that controls the relief setpoint of the valve, therefore seismic qualification is not a concern. The relief valve relieves to the atmosphere in the SWPH.

Installation of a relief valve with a lower setpoint will protect the air receiver from possible damage due to overpressurization and will also prevent injury of station personnel. Operation of the SW Air System will be unaffected by the lower relief valve setpoint since the maximum operating pressure of the system is still approximately 20 psig less than the new setpoint. The new setpoint of 118 psig will be less than the ASME VIII code allowable pressure of 119.78 psig based on the minimum wall thickness reading obtained by UT examination.

Based on the above discussion, an unreviewed safety question does not exist for this temporary modification.

00-SE-TM-04

Description
Temporary Modification # N1-1682
The power supply to the reheater temperature control system was found to be drifting low causing 1-MS-FCV-104A, B, C, D to partially close. The valves were placed on manual override to maintain full open position. Troubleshooting by I&C found that the power supply is being excessively loaded by the electronic components associated with the reheat temperature control system which is not normally used at North Anna. It desired to remove one or more of the associated system circuit cards (PC1, PC2, PC3) from 1-EI-CB-30 to return the power supply output to normal.

Summary
Disabling one or more of the automatic reheater temperature control functions by removal of one or more of the PC1, PC2, and PC3 circuit cards from 1-EI-CB-30 will eliminate the problem with the reheat system power supply and return the power supply output to normal levels. This Temporary Modification, by restoring the power supply to normal output, will allow the manual override on the reheat FCVs to be cleared and allow them to function properly during a reactor trip without reliance on manual operator actions. The equipment affected by this Temporary Modification provides a control, not a protection function, and is not used at North Anna with the exception of the LP1 and LP2 turbine inlet temperature indicators.

The LP1 and LP2 temperature indicators on the Reheat Valve Control Panel may be disabled as an effect of this modification. Alternate temperature indications are available and will be used to monitor heatup and cooldown rates by procedural control should the LP1 and LP2 indicators be rendered inoperable by the selected card removal.

The most extensive case of card removal (removal of all three circuit cards - PC1, PC2 and PC3) is considered the most limiting configuration for this Temporary Modification. Therefore, this case is assumed in the body of this Safety Evaluation. Cases involving removal of only one or two of the cards are bounded by this evaluation of the most extensive case.

This Temporary Modification does not constitute an Unreviewed Safety Question for the following reasons:

1. Removing the card(s) from the system does not affect any automatic safety functions. The automatic temperature control aspect of the system has no safety function and is not currently in use at NAPS.

2. The reheat temperature control system is not Safety Related, has no Tech Spec requirements and is not described in the UFSAR. The probability or consequences of an accident are not affected.

3. Even in the case of the removal of all three circuit cards (PC1, PC2, and PC3), all functions of the reheat control system function other than the automatic temperature control function and the LP1 and LP2 inlet temperature indications on the Reheat Control Panel are unaffected. The operation of the RESET button as described in 1-E-0 is unchanged, as is normal closure of the FCVs in the event of a trip.

Therefore, this Temporary Modification does not involve an Unreviewed Safety Question and no changes are required to the Operating License.

00-SE-TM-05

Description

Temporary Operating Procedure 1-TOP-43.2, "1-GM-D-1B, GENERATOR HYDROGEN SYSTEM GAS DRYER OPERATION WITH TEMPORARY FLOW MODIFICATION," Rev. 0
Temporary Maintenance Operating Procedure 1-TMOP-43.1, "1-GM-D-1B, GENERATOR HYDROGEN SYSTEM GAS DRYER TEMPORARY FLOW MODIFICATION," Rev. 0
Temporary Modification 1683
This Safety Evaluation addresses a "Procedurally Controlled Temporary Modification" (ref. VPAP-1403, Section 6.3) to control the installation of a mechanical jumper to bypass the blower (1-GM-BLO-1B) for the Unit 1 main generator hydrogen dryer 1-GM-D-1B. As such, the requirements for the installation of this Procedurally Controlled Temporary Modification (PCTM) are delineated in new Temporary Operating Procedure 1-TOP-43.2 and Temporary Maintenance Operating Procedure 1-TMOP-43.1. A flow path will therefore be established from the main generator through the hydrogen dryer subsystem. It will be necessary to electrically disconnect the blower motor from its power supply to prevent the dryer controls from starting the motor in the temporary configuration. This equipment is located in the basement of the Unit 1 Turbine Building. The result of this action will be that during power operations, the hydrogen gas flow path through the dryer subsystem will bypass the blower with the dryer in service.

Summary

MAJOR ISSUES:

This Safety Evaluation addresses a "Procedurally Controlled Temporary Modification" (ref. VPAP-1403, Section 6.3) to control the installation of a mechanical jumper to bypass the blower (1-GM-BLO-1B) for the Unit 1 main generator hydrogen dryer 1-GM-D-1B. As such, the requirements for the installation of this Procedurally Controlled Temporary Modification (PCTM) are delineated in new Temporary Operating Procedure 1-TOP-43.2 and Temporary Maintenance Operating Procedure 1-TMOP-43.1. A flow path will therefore be established from the main generator through the hydrogen dryer subsystem. It will be necessary to electrically disconnect the blower motor from its power supply to prevent the dryer controls from starting the motor in the temporary configuration. This equipment is located in the basement of the Unit 1 Turbine Building. The result of this action will be that during power operations, the hydrogen gas flow path through the dryer subsystem will bypass the blower with the dryer in service. This change will permit evaluation of the gas flow achievable if the blower were removed from the hydrogen gas dryer system. An Unreviewed Safety Question evaluation is required to assess the modified configuration.

JUSTIFICATION:

This change should be allowed as it will be in compliance with the Technical Specifications, the Safety Analysis Report, and the design basis requirements of the Unit 1 main turbine and all associated plant systems. The SAR does not specifically describe the hydrogen dryers for the main generator in any way. Operation of the turbine generator control and protection systems is not affected by this change. The overall operation of the turbine generator is unchanged and no changes are being made to any protection circuits. The hydrogen dryers may be out of service without adverse impact on any SAR-described operation. The purity of the hydrogen gas for the main generator will be maintained within the specified limits during and following implementation of this change — thus, the performance of the main generator will not be adversely affected by this change. Operation of the main generator is not adversely affected by this activity. The generator control and protection system will remain in its normal configuration during and following implementation of this change. Installed instrumentation will not be affected by this change — thus, the ability of the operators to monitor the plant will not be reduced by this activity. In addition, a flow meter will be installed as part of the PCTM to enhance the ability to assess the operational performance with the blower bypassed.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

1. Condition does not increase the probability of occurrence or the consequences of an accident or malfunctions of equipment important to safety and previously evaluated in the Safety Analysis Report.

This activity is the installation of a mechanical jumper to bypass the blower for the hydrogen dryer and subsequent operation of the dryer. In addition, it will be necessary to electrically disconnect the blower motor from its power supply to prevent the dryer controls from starting the motor in the temporary configuration. Thus, this activity will be implemented per approved station procedures, which are bounded by existing analysis. In addition, this activity does not increase the probability of any turbine or main steam related accidents. Proper turbine protection requires that the turbine be isolated from the steam supply in the event of an accident.

2. Condition does not create a possibility for an accident or malfunction of a different type than was previously evaluated in the Safety Analysis Report.

The operation of the equipment and systems important to nuclear safety are not affected by this activity. The mechanical jumper installation is consistent with existing requirements for components in the GM system and its failure is bounded by the existing analysis. In addition, all accidents that involve the turbine-generator require isolating main steam from the turbine to control and limit the accident. The steam isolation capability of the main turbine has not been affected by this change.

3. Condition does not reduce the margin of safety of any part of the Technical Specifications as described in the bases section.

Technical Specification margin as it relates to the main turbine-generator is concerned with isolation of steam flow from the turbine in the event of a turbine trip or overspeed condition. Neither of these is affected by the installation of a jumper to bypass the blower for the hydrogen dryer. Thus, the changes of the subject Procedurally Controlled Temporary Modification do not reduce the margin of safety of any part of the Technical Specifications as described in the Basis section.

00-SE-TM-06

Description
Temporary Modification # 1688 and 1131
The TM jumpers out the circuitry for the fire protection trip and the vibration switch trip of the Bearing Cooling (BC) fans.

Summary
A temporary modification is being installed which defeats the Fire Protection system and vibration trips of the running Bearing Cooling fans. This is being done due to an existing malfunction within the circuitry.

The Bearing Cooling, affected Fire Protection system, and vibration detection equipment are not safety related nor are they required by Technical Specifications. Installation of the jumper does not affect any safety related systems and would not affect the ability of any safety related systems to perform its intended functions.

There will be a once per shift fire watch while this jumper is installed to provide fire detection capability. Upon notification, actions will be taken to secure BC fans, dispatch the Fire Brigade, and attempt to initiate Tower Cell deluge.

In addition, the vibration trip circuit provides protection to each individual BC fan in the event of elevated vibrations. Predictive Analysis will monitor each of the fans for increasing vibrations to detect conditions that could adversely affect the performance of the fans.

The probability of a fire is not increased nor is the probability of damage to the BC fans. Temporarily defeating the protection circuitry while maintaining BC fan operation has no ability to influence the mechanisms by which a fire or fault is generated. Compensatory actions established will adequately provide the same functions.

The overall operational consequences of a loss of Bearing Cooling from a tower/fan problem remains the same - shutdown of the secondary side of the plant.

This TM does not create the means to cause a new or different malfunction of the BC system or surrounding equipment.

The margin of safety for the station as described in the Technical Specifications Bases is not altered since the BC system and its components are not described in the Technical Specifications.

There is no Unreviewed Safety Question involved with doing this Temporary Modification.

00-SE-TM-07

Description
Temporary Modification (TM) 1685
Allow testing of the incore sump high level switch (1-DA-LS-106).

Summary
The Incore Sump Level alarm (1J-C8) is comprised of a Hi and then a Hi-Hi alarm corresponding to sump levels of 18" and 20" respectively. These two alarms are actuated by two different level electrodes. The 18" high electrode provides the level control for the incore instrumentation sump and at 18" will open contact (B) (11715-TLD-DA-20) in level switch 1-DA-LS-106 to actuate the Hi level alarm and will close contact (A) in the switch to start the incore instrument room sump pump, 1-DA-P-5. The 20" electrode provides the Hi-Hi level alarm only and will close contact (C) in the level switch at 20" to reflash the Hi/Hi-Hi level alarm. The pump stops and the alarm clears when level reaches the setpoint of a third probe (6" electrode).

On 8/11/2000, annunciator 1J-C8 (Incore Inst. Room Sump Hi/Hi-Hi Level alarm) was received and remained locked in. The hathaway system was tested and it was determined that the Hi-Hi portion of the field circuit was causing the alarm. However, it is not clear that an actual high level condition exists since there is no level indication for the incore sump. To ensure the Hi-Hi level alarm is due to an erroneous signal and not actual incore sump level, this Temporary Modification (TM) will install a jumper from contact C01 to C00 (11715-ESK-6GD) to defeat the 1-DA-P-5 high sump level interlock, allowing the pump to be manually started with a sump level less than 18". This jumper will ensure the pump will operate on a valid Hi level signal and will verify the operability of the 18" electrode.

This TM will be installed only long enough to test the validity of the Hi/Hi-Hi level alarm and will then be removed. Prior to installing the jumper, the breaker for 1-DA-P-5 will be opened to ensure the pump does not prematurely start and run longer than necessary. Once the jumper is installed, the breaker will then be closed and the pump started and run until one of the following occurs:

(a) The containment sump level stops increasing (the incore sump pump discharges to the containment sump),
(b) The incore sump Hi/Hi-Hi level alarm (1J-C8) clears, or
(c) 30 seconds passes with no change in the containment sump level

This Temporary Modification should be allowed for the following reasons:

1) The safety significance of this Hi-Hi alarm is minimal. Reactor Coolant leakage limitations are listed in Tech Spec 3.4.6.2. and leakage detection systems required are listed in 3.4.6.1. The leakage detection systems are Containment gaseous and particulate radiation monitors and the containment sump level and discharge flow measurement system. The Incore sump level indication is not a part of the Safety Analysis system required to monitor for increased RCS leakage.

2) The UFSAR also takes no credit for the sump level indication in Section 5.2.4.1, 'Leakage Detection'. The UFSAR credits the following systems for monitoring for RCS leakage: Containment gaseous radiation monitor, Containment particulate radiation monitor, Containment Structure leakage monitoring system, Containment recirculation system cooler heat load, Containment Sump monitoring and the RCS makeup rate. Again, the Incore sump is not included in the group of essential leakage indications.

3) The Incore Sump level alarm is a good indicator to have available and this TM will ensure that the Hi level portion of the alarm which is still operating properly.

No Technical Specifications require change by implementing the TM. This TM is beneficial in that one portion of the alarm is restored for use by the operator. For these reasons no unreviewed safety question is created by this TM for the Hi/Hi-Hi alarm on Incore Sump Level and the TM should be installed.

00-SE-TM-08

Description
Temporary Modification (TM) 1687
The HI-Hi alarm for the Incore Room Sump Level is alarming spuriously (1-DA-LS-106). The high alarm for the Incore Room Sump level remains operable as does the operation of the sump pump.

Summary
The Incore Sump Level alarm is comprised of a Hi and then a Hi-Hi alarm corresponding to sump levels of 18" and 20" respectively. These two alarms are actuated by level switch 1-DA-LS-106-1 and -2. The 106-1 switch is sending an erroneous signal to indicate a sump level of 20". This Temporary Modification will disable this erroneous signal and restore the Control Room annunciator to a non-alarming condition until an actual Hi level of 18 inches is reached in the sump. This Temporary Modification should be allowed for the following reasons:

1) The safety significance of this Hi-Hi alarm is minimal. Reactor Coolant leakage limitations are listed in Tech Spec 3.4.6.2. and leakage detection systems required are listed in 3.4.6.1. The leakage detection systems are Containment gaseous and particulate radiation monitors and the containment sump level and discharge flow measurement system. The Incore sump level indication is not a part of the Safety Analysis system required to monitor for increased RCS leakage.

2) The UFSAR also takes no credit for the sump level indication in Section 5.2.4.1, 'Leakage Detection'. The UFSAR credits the following systems for monitoring for RCS leakage: Containment gaseous radiation monitor, Containment particulate radiation monitor, Containment Structure leakage monitoring system, Containment recirculation system cooler heat load, Containment Sump monitoring and the RCS makeup rate. Again, the Incore sump is not included in the group of essential leakage indications.

3) The Incore Sump level alarm is a good indicator to have available and this TM will restore that portion of the alarm which is still operating properly, the Hi level alarm. By disabling the Hi-Hi alarm, the original intent of the alarm is restored by alerting the OATC to an unusual condition with the audible and visual alarm from Annunciator 1J-C8 when sump level reaches 18". Allowing the HI-Hi alarm to remain in (flashing) with the audible alarm acknowledged establishes a degree of complacency on the part of the operator regarding an alarm which is constantly present and this should be avoided.

No Technical Specifications require change by implementing the TM. This TM is beneficial in that one portion of the alarm is restored for use by the operator. For these reasons no unreviewed safety question is created by this TM for the HI-Hi alarm on Incore Sump Level and the TM should be installed.

00-SE-TM-09

Description
Temporary Modification (TM) N1-1689
Roll leads at 1-EI-CB-16A TB 13 and 14 and at local junction box for 1-SW-FT-103, Service Water Return Header No. 4 Flow Transmitter

Summary
A ground was discovered on the positive lead (black) of the FT circuit causing indication failure of 1-SW-FT-103, Service Water Return Header No. 4 Flow Transmitter. The Temporary Modification (TM) involves rolling leads at 1-EI-CB-16A TB 13 and 14 and the local junction box for 1-SW-FT-103. Swapping the leads will allow the ground to be transferred to the negative (white) side of the circuit, which is normally grounded. This action will restore the circuit to a functional condition, and allow it to be calibrated and returned to operable status. When calibrated, the transmitter should perform as expected with no spurious signals.

The TM will be installed and remain in place until the associated cable can be replaced. The long term corrective action will return the cable and circuit to original design conditions. Use of the temporary arrangement is considered acceptable in the short term to restore the channel to operable status. The long term corrective action (cable replacement) will eliminate any degraded condition associated with the existing cable. These short term and long term corrective actions are compliant with Generic Letter 91-18 (Information on resolution of degraded and nonconforming conditions) guidance regarding the treatment of component operability and restoration of qualification. As previously mentioned, the short term action will allow the channel to be returned to a functional condition, calibrated, and returned to operable status.

UFSAR Section 9.2.1 describes the Service Water System. The Section states that the flow instrumentation for the Service Water headers is accurate for major changes such as those caused by a loss of a pump or erroneous valve line-up. It also states that associated alarms are provided to detect a complete or major loss of Service Water flow. The TM will not change the purpose or function of the flow indication loop. Calibration after the configuration change will ensure accuracy and operability of the indication. The reconfiguration will not cause adverse effects in the parameter indication. The TM is limited to one flow instrument loop. There will be no affect on any other instruments. Channel separation will not be compromised.

The TM does not involve or create an Unreviewed Safety Question. The indication loop itself is used to monitor the status and performance of the Service Water System. The indication is not associated with the initiation of any accident/malfunction or any accident/malfunction precursor. Therefore, the TM will not increase the probability of an accident or malfunction. Since the TM will restore the indication loop to operable status, the instrument will be available to monitor Service Water Header No. 4 return flow. As such, the TM will not increase the consequences of any accident or malfunction. No new equipment, instrument components, or new failure modes are introduced, so no new accidents or malfunctions are created. The function of the instrument will remain the same, so no new T.S. surveillance requirements are required, nor are any License Condition changes necessitated. Therefore, the margin of safety as described in the T.S. Bases for the Service Water System and other related systems is unchanged.

00-SE-TM-10

Description
Temporary Modification (TM) N1-1686
Lift the leads at 1-GM-TS-102B to disable the input to annunciator K-B7 (Generator Leads Cooling Trouble).

Summary
Temperature switch 1-GM-TS-102B, "B" phase bus duct air temperature, is alarming prior to the setpoint of 175 degrees Fahrenheit and periodically causes annunciator K-B7 to lock in on hot days. Since the temperature switch can not be repaired or calibrated on line, and the annunciator has no reflash capability, it is desired to lift the leads from the switch to disable the input to the annunciator until the Winter months. Since there is no reflash associated with this annunciator, other inputs will have no effect on the alarm when it is locked in, and therefore it is beneficial to remove the degraded input to this annunciator.

Although there will be no input to annunciator K-B7 from the "B" phase bus duct temperature, the following inputs will still be available to cause a Generator Leads Cooling Trouble alarm:

1-GM-TS-102A & C, high "A" and "C" phase bus duct air temperature;
1-GM-TS-103A through -103F, high "A", "B", and "C" main transformer low side bus temperature;
1-GM-TS-104, high generator bus duct cooling return air temperature;
1-GM-FS-100, low generator bus duct supply air flow
1-GM-FS-101, loss of water flow to generator leads cooler
1-GM-MS-100, high relative humidity of the cooling air supply

Temperature switch 1-GM-TS-102B provides no other function than providing an input to K-B7. There are numerous inputs to the annunciator that would alert the operator of a problem with the generator leads bus duct air cooling system or the Bearing Cooling Water system which is used to cool the air. In addition to the above mentioned inputs, there are temperature indicators installed in bus ducting, and therefore, temperatures of each phase can be obtained locally.

This Temporary Modification does not constitute an Unreviewed Safety Question for the following reasons:

1. Removing the input from 1-GM-TS-102B to annunciator K-B7 does not affect any automatic safety functions.

2. The temperature switch is not Safety Related, has no Tech Spec requirements and is not described in the UFSAR. The probability or consequences of an accident are not affected.

3. The probability or consequences of an accident or malfunction occurring previously evaluated in the SAR is not increased, nor is the possibility of creating a new accident or malfunction increased as a result of this temporary modification.

Therefore, this Temporary Modification does not involve an Unreviewed Safety Question and no changes are required to the Operating License.

00-SE-TM-11

Description
Temporary Modification N2-1684
2-OP-50.4 (changed to include operating instructions for the temporary seal water supply)
Install temporary seal water supply to 2-BC-P-5A and 2-BC-P-5B (Mechanical Chiller Condenser Pumps). Supply will be from the running pump discharge (discharge PI drain) to the seals. Installation will allow capability to supply seal water from the discharge line of either pump. Previously, seal water was supplied from tubing off the individual pump casing.

Summary
The Temporary Modification (TM) involves the installation of temporary seal water supply to 2-BC-P-5A and 2-BC-P-5B (Mechanical Chiller Condenser Pumps). Supply will be from the running pump discharge (discharge PI drain) to the seals. Installation will allow capability to supply seal water from the discharge line of either pump. Previously, seal water was supplied from tubing off the individual pump casing. It is desired to provide temporary seal water supply for 2-BC-P-5A and 2-BC-P-5B from a more positive pressure source than the pump casing due to frequent air binding of the pumps. Tubing and fittings for the TM will be rated for the application.

UFSAR Section 9.2.2 describes the Chilled Water subsystem. It notes that the subsystem supplies chilled water to the Containment Air Recirculation Cooling Coils, the Gas Stripper Vent Chiller, sampling coolers, the Waste Gas Recombiner Aftercooler (not used), and the RWST coolers (used for initial cooldown of the RWST after completion of refueling). It states that the Chilled Water subsystem does not supply water to equipment that is required to operate to maintain the plant in a safe condition. The Mechanical Chiller is one of the components that can be used to cool Chilled Water. The UFSAR states that the Mechanical Chiller refrigerant is condensed by water from the Bearing Cooling System.

UFSAR Section 10.4.7 describes the Bearing Cooling system. A description is provided for the Mechanical Chiller Condenser Pumps which states that they are installed in parallel with the Unit 2 Bearing Cooling Pumps and associated system components, to permit the condenser pumps to take suction from and discharge back to the same source as the Bearing Cooling Pumps. It further states that the pumps are designed to supply cooling water to the condenser of the Mechanical Chiller unit. A detailed description of the seal water arrangement for the condenser pumps is not included. However, one of the drawings referenced in the UFSAR description does show the seal water supply arrangement from the pump casing.

The TM will change the seal water arrangement shown on the reference drawing (12050-FM-24A/80A) by providing supply from the discharge of the running pump. The existing seal water supply valves at the pump seals will remain to provide the capability to throttle the supply water as required. The ability to isolate the supply tubing in the event of failure is provided by the configuration. The arrangement will not cause any adverse effects to pump operation. It will provide a more positive pressure source of seal water to enhance the sealing of the shaft to eliminate air inleakage from the shaft area. This will aid in maintaining a good prime of the pump(s) and thus, reduce the chance of air binding. The overall function of the pump to provide cooling water to the Mechanical Chiller to condense the refrigerant remains unchanged. Enhanced seal water supply will increase the reliability of the pumps and Mechanical Chiller operation.

The TM does not introduce an Unreviewed Safety Question for the following reasons:

The Mechanical Chiller and its condenser pumps do not contribute to the initiation of any analyzed accidents. The change to the seal water supply arrangement will enhance the availability of pump shaft sealing water to increase the reliability of pump and Mechanical Chiller operation. Thus, the probability of an accident or loss of Chilled Water is not increased.

The change is limited to the Mechanical Chiller Condenser Pumps. The arrangement will not adversely affect the operation of any component used to mitigate the consequences of any accident. It is anticipated that the reliability of the pumps to support operation of the Mechanical Chiller will be enhanced. Even if a loss of the operation of the Mechanical Chiller occurred, the TM would not affect the ability to utilize Service Water as an alternate source of cooling water for normal Containment Cooling. In any event, operation of the Chilled Water subsystem is not required for the mitigation of any analyzed accident, nor is it required to operate to maintain the plant in a safe condition.

Any impact from the TM is limited to the condenser pumps. Since it only involves a reconfiguration of the seal water supply to the pump from a different location in the process flow, no new plant accidents or malfunctions of the Chilled Water or Bearing Cooling systems are created.

The change does not directly impact any T.S. LCO or License Conditions. The T.S. requirements for Containment Average Temperature and Partial Air Pressure do not govern the equipment used for cooling and pressure control. They are parameter based specifications to ensure assumed conditions for accident analysis. In the event of a loss of the Mechanical Chiller, alternate means are available for Containment Cooling Water supply utilizing Service Water or Chilled Water cooled by the Steam Chillers. Compliance with the specifications will be maintained. Thus, the T.S. margin of safety will not be decreased.

00-SE-TM-12

Description

Temporary Operating Procedures 1-TOP-43.2, Rev. 1, and 2-TOP-43.2, Rev. 0
Temporary Maintenance Operating Procedures 1-TMOP-43.1, Rev. 1, and 2-TMOP-43.1, Rev. 0
Temporary Modification No. 1690 (Unit 1) and 1131 (Unit 2)
This Safety Evaluation addresses a "Temporary Modification" (ref. VPAP-1403) to control the installation of a mechanical jumper to bypass the blowers for the Units 1 and 2 main generator hydrogen dryers 1-GM-D-1B and 2-GM-D-1B. The requirements for the installation of this Temporary Modification are delineated in Temporary Operating Procedure TOP-43.2 and Temporary Maintenance Operating Procedure TMOP-43.1. A flow path will therefore be established from the main generator through the hydrogen dryer subsystem. It will be necessary to electrically disconnect the blower motor from its power supply to prevent the dryer controls from starting the motor in the temporary configuration. This equipment is located in the basement of the respective Turbine Building. The result of this action will be that during power operations, the hydrogen gas flow path through the dryer subsystem will bypass the blower with the dryer in service.

Summary

MAJOR ISSUES:

This Safety Evaluation addresses a Temporary Modification to control the installation of a mechanical jumper to bypass the blowers (1-GM-BLO-1B and 2-GM-BLO-1B) for the Units 1 and 2 main generator hydrogen dryers 1-GM-D-1B and 2-GM-D-1B. As such, the requirements for the installation of this Temporary Modification (TM) are delineated in new Temporary Operating Procedures 1-TOP-43.2 and 2-TOP-43.2 and Temporary Maintenance Operating Procedures 1-TMOP-43.1 and 2-TMOP-43.1. A flow path will therefore be established from the main generator through the hydrogen dryer subsystem. It will be necessary to electrically disconnect the blower motor from its power supply to prevent the dryer controls from starting the motor in the temporary configuration. This equipment is located in the basement of the respective Turbine Building. The result of this action will be that during power operations, the hydrogen gas flow path through the dryer subsystem will bypass the blower with the dryer in service. This change will permit evaluation of the gas flow achievable if the blower were removed from the hydrogen gas dryer system. An Unreviewed Safety Question evaluation is required to assess the modified configuration.

JUSTIFICATION:

This change should be allowed as it will be in compliance with the Technical Specifications, the Safety Analysis Report, and the design basis requirements of the Unit 1 main turbine and all associated plant systems. The SAR does not specifically describe the hydrogen dryers for the main generator in any way. Operation of the turbine generator control and protection systems is not affected by this change. The overall operation of the turbine generator is unchanged and no changes are being made to any protection circuits. The hydrogen dryers may be out of service without adverse impact on any SAR-described operation. The purity of the hydrogen gas for the main generator will be maintained within the specified limits during and following implementation of this change — thus, the performance of the main generator will not be adversely affected by this change. Operation of the main generator is not adversely affected by this activity. The generator control and protection system will remain in its normal configuration during and following implementation of this change. Installed instrumentation will not be affected by this change — thus, the ability of the operators to monitor the plant will not be reduced by this activity.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

1. Condition does not increase the probability of occurrence or the consequences of an accident or malfunctions of equipment important to safety and previously evaluated in the Safety Analysis Report.

This activity is the installation of a mechanical jumper to bypass the blower for the hydrogen dryer and subsequent operation of the dryer. In addition, it will be necessary to electrically disconnect the blower motor from its power supply to prevent the dryer controls from starting the motor in the temporary configuration. Thus, this activity will be implemented per approved station procedures, which are bounded by existing analysis. In addition, this activity does not increase the probability of any turbine or main steam related accidents. Proper turbine protection requires that the turbine be isolated from the steam supply in the event of an accident.

2. Condition does not create a possibility for an accident or malfunction of a different type than was previously evaluated in the Safety Analysis Report.

The operation of the equipment and systems important to nuclear safety are not affected by this activity. The mechanical jumper installation is consistent with existing requirements for components in the GM system and its failure is bounded by the existing analysis. In addition, all accidents that involve the turbine-generator require isolating main steam from the turbine to control and limit the accident. The steam isolation capability of the main turbine has not been affected by this change.

3. Condition does not reduce the margin of safety of any part of the Technical Specifications as described in the bases section.

Technical Specification margin as it relates to the main turbine-generator is concerned with isolation of steam flow from the turbine in the event of a turbine trip or overspeed condition. Neither of these is affected by the installation of a jumper to bypass the blower for the hydrogen dryer. Thus, the changes of the subject Temporary Modification do not reduce the margin of safety of any part of the Technical Specifications as described in the Basis section.

00-SE-TM-13

Description
Temporary Modification N2-1133
Temporary cable will be routed from the output of isolation card C3-434 from flow transmitter 2-FW-FT-2477 to test cables in the rack room. The permanently installed test cables in the rack room lead to a junction box in the Control room where temporary cable will be routed to flow indicator 2-FW-FI-2477.

Summary
The conductors of the cable feeding indicator 2-FW-FI-2477 appear to be failing causing the "A" SG Channel III Feedwater Flow indication to be erratic and low. There appears to be abnormally high resistance in this existing cable. This temporary modification involves routing new temporary cable from the output of isolation card C3-434 from 2-FW-FT-2477 to permanently installed test cable going from junction box JB-2701-2 in the rack room to junction box JB-2704-2 in the control room. From the junction box in the control room, new temporary cable will be routed to 2-FW-FI-2477 on the vertical board. This action will restore the flow indicator to operable status and allow the operator to monitor this channel of feed flow.

The TM will be remain in place until new permanent cable can be installed, most likely next Unit 2 refueling outage. Use of the temporary arrangement is considered acceptable to restore the channel to operable status. The long term corrective action will eliminate the degraded condition of the existing cable. These short and long term corrective actions are in compliance with GL 91-18 (Information on Resolution of Degraded and Nonconforming conditions) guidance regarding treatment of component operability and restoration of qualification.

UFSAR Section 10.4.3 describes the Feedwater and Condensate systems. A failure in the feedwater control system could lead to either an abnormal increase or decrease in the SG water level. An abnormal increase is terminated by the SG Hi - Hi level function (P14) and an abnormal decrease is terminated by either a SG Low level coincident with a steam flow feed flow mismatch or a SG Low - Low level. Auxiliary feedwater would automatically start and flow would be provided to the SG on a SG Low - Low level Reactor Trip. This temporary modification will not change the function of the indicating loop and will bypass the portion of the existing cable going to the flow indicator allowing the flow indicator to function properly. The existing fault in the cable will not be eliminated by this TM. Properly installed this TM will not affect the feedwater flow input to the SG Water Level Control circuit. The feed reg. valves will continue to function as designed. The new temporary cable used to perform this modification will be, as a minimum, the same gauge as the existing cable. Calibration after the TM is installed will ensure the accuracy and operability of the feed flow indication. This TM is limited to one channel and will not affect any other trains or channels. Channel separation will not be compromised. This TM will not affect the protection function of the instrument loop since it is isolated from the control circuit through the isolation card.

This TM does not involve or create an unreviewed safety question. The indicating loop will be returned to an operable status and both the protection and control circuits will be unaffected by this TM. The TM will not increase the probability of an accident or malfunction. No new equipment, other than the temporary bypass cable, is being introduced so no new accidents or malfunctions are created. The function of the instrument will remain the same. No new T. S. surveillance requirements are required nor are any License Conditions required to be changed. The margin of safety as described in the T. S. Bases for the feedwater and other related systems is unchanged.

00-SE-TM-14

Description

Temporary Modification N1-1691
This Safety Evaluation addresses the installation of a locking device to maintain 1-BC-MOV-125 and/or 2-BC-MOV-225, the Unit 1 & 2 Bearing Cooling to Circulating Water Discharge Tunnel Isolation Valves, in the open position during Lake to Lake operation. The valves are currently oscillating in the flow stream. 1-BC-MOV-125 and/or 2-BC-MOV-225 will be repaired after the BC system is returned to Tower to Tower operation. Valve repair will be controlled by the completion of their respective work orders.

Summary

This Temporary Modification (TM) will install clamping devices on the valve stems of 1-BC-MOV-125 and/or 2-BC-MOV-225, the Unit 1 & 2 Bearing Cooling to Circulating Water Discharge Tunnel Isolation Valves, to maintain them open and to prevent oscillation of the valves in the flow stream during Lake to Lake operation. These valves are located approx. 10 feet above the basement floor in the Turbine Building. The clamping device is required to prevent closure and oscillation of the valves since the worm gears are suspected of being worn. The valves will be tagged open with no power to the valves until after the clamping devices are removed. The Unit 1 & 2 discharge lines to the Circulating Water tunnel can be isolated if necessary by either removing the clamp and manually operating the valve; removing the clamp, clearing tags then operating the valve electrically; or by manually increasing air pressure to 1-BC-PCV-116 or 2-BC-PCV-216, respectively. The configuration and design of the clamping devices is such that accidental bumping or removal of the clamps is highly unlikely. If the clamping device should fall off the valve will remain in the open position. Removal of the temporary clamping devices will be coordinated with returning the BC system to Tower to Tower operation at which time tags will be lifted and the valves closed electrically. Valve repair will be performed after the valves are closed and the Bearing Cooling system is returned to the Tower to Tower mode of operation.

The BC system is not a Tech Spec system. The Bearing Cooling system is described in section 10.4.7 of the UFSAR. The UFSAR describes operation of the Bearing Cooling system in either the Tower to Tower or Lake to Lake mode of operation though Tower to Tower is the preferred mode of operation. Though not a safety related function, auto transfer capability to/from Tower to Tower from/to Lake to Lake will be impaired with the TM installed. The TM will be removed prior to any auto-transfer attempt. Operation of the Bearing Cooling system will continue to conform to UFSAR requirements. There is no adverse environmental impact as a result of this temporary modification.

The overall function and operation of the BC system will remain unchanged as a result of this activity. BC will still be available, as necessary, to support inservice loads as plant conditions warrant. The Bearing Cooling system is non safety related and is not required for obtaining or maintaining safe shutdown. Therefore, the temporary installation of these clamping devices on 1-BC-MOV-125 and/or 2-BC-MOV-225 does not create an unreviewed safety question.

00-SE-TM-15

Description
Temporary Modification N1-1691, controlled by 0-MCM-0400-18
This Safety Evaluation addresses the installation of a locking device to maintain Bearing Cooling MOV's, which are normally closed during Tower to Tower operation, open during Lake to Lake operation. The valves are currently oscillating in the flow stream causing system pressures to fluctuate. The valves with locking devices will be repaired after the BC system is returned to Tower to Tower operation. Valve repair will be controlled by the completion of their respective work orders.

Summary
This Temporary Modification (TM) will install clamping devices on the valve stems of the Bearing Cooling System MOV's, to maintain them open and to prevent oscillation of the valves in the flow stream during Lake to Lake operation. The clamping device is required to prevent closure and oscillation of the valves since the worm gears are suspected of being worn. The valves will be tagged open with no power to the valves until after the clamping devices are removed. The lines on which these valves are installed can be isolated, if necessary, by either removing the clamp and manually operating the valve; removing the clamp, clearing tags then operating the valve electrically; or by isolating the line by the use of other valves in the system, as applicable. The configuration and design of the clamping devices is such that accidental bumping or removal of the clamps is highly unlikely. If the clamping device should fall off the valve will remain tagged in the open position. Removal of the temporary clamping devices will be coordinated with returning the BC system to Tower to Tower operation at which time tags will be lifted and the valves closed electrically. Valve repair will be performed after the valves are closed and the Bearing Cooling system is returned to the Tower to Tower mode of operation.

The BC system is not a Tech Spec system. The Bearing Cooling system is described in section 10.4.7 of the UFSAR. The UFSAR describes operation of the Bearing Cooling system in either the Tower to Tower or Lake to Lake mode of operation though Tower to Tower is the preferred mode of operation. Though not a safety related function, auto transfer capability to/from Tower to Tower from/to Lake to Lake will be impaired with the TM installed. The TM will be removed prior to any auto-transfer attempt. Operation of the Bearing Cooling system will continue to conform to UFSAR requirements. There is no adverse environmental impact as a result of this temporary modification.

The overall function and operation of the BC system will remain unchanged as a result of this activity. BC will still be available, as necessary, to support inservice loads as plant conditions warrant. The Bearing Cooling system is non safety related and is not required for obtaining or maintaining safe shutdown. Therefore, the temporary installation of these clamping devices on these Bearing Cooling MOV's does not create an unreviewed safety question.

00-SE-TM-16

Description
Temporary Modification N1-1692
A jumper will be installed which will bypass the water level permissive normally required from 1-VP-LS-110A to allow the "A" Bearing Cooling Pump to start due to a loss of Bearing Cooling.

Summary
An electrical jumper is to be installed to bypass the start permissive normally required from Vacuum Priming Level Switch 1-VP-LS-110A. The purpose of this start permissive to ensure that there is sufficient water level in the Bearing Cooling Pump and system to prevent cavitation and/or air binding of the Bearing Cooling Pump during a start. The "A" Bearing Cooling Pump could be damaged if the pump is started with little or no water in the pump casing.

Failure to perform this modification could potentially cause very serious damage to a number of secondary plant component including the Main Generator if Bearing Cooling was lost and the "A" Bearing Cooling Pump could not be started during the maintenance scheduled on 1-VP-ARV-107A. Maintenance cannot be performed on ARV 107A without isolating the level switch from the Bearing Cooling System. Damage to the "A" Bearing Cooling Pump would be of minor consequence if Bearing Cooling was lost to the secondary plant while at power.

The jumper will be installed in breaker cubicle 15B8 between terminal AJ-2 (Lead No. 1BCSA01C03) and terminal AJ-8 (Lead No. 1BCSA01C01). This jumper has been successfully installed on previous occasions, in accordance with Attachment 6 of 1-ECM-2303-01, for uncoupled test runs.

A 1-LOG-14 is currently being performed twice per shift by operations to verify that water is leaking from the 1-BC-P-1A pump seals which provides reasonable assurance that air is not present in the pump casing.

Bearing Cooling is not a Tech. Spec. system. The Bearing Cooling system is described in section 10.4.7 of the UFSAR. The UFSAR describes operation of the Bearing Cooling system in either the Tower to Tower or Lake to Lake operation. Currently we are running in a Lake to Lake configuration. This TM will be removed when the repairs on the ARV are completed. Operation will continue to conform to UFSAR requirements. There is no adverse environmental impact as a result of this TM.

The overall function and operation of the Bearing Cooling system will remain unchanged as a result of this activity. This TM will provide additional assurance that BC will still be available, as necessary, to support inservice loads as plant conditions warrant. The Bearing Cooling System is non-safety related and is not required for obtaining or maintaining safe shutdown. Therefore, this TM does not create an unreviewed safety question.

00-SE-TM-17

Description
Temporary Modification # U2 1134
The power supply to the reheater temperature control system was found to be drifting low causing 2-MS-FCV-204A, B, C, D to partially close. The valves were placed on manual override to maintain full open position. Troubleshooting by I&C found that the power supply is being excessively loaded by the electronic components associated with the reheat temperature control system which is not normally used at North Anna. It desired to remove one or more of the associated system circuit cards (PC1, PC2, PC3) from 2-EI-CB-30 to return the power supply output to normal.

Summary
Disabling one or more of the automatic reheater temperature control functions by removal of one or more of the PC1, PC2, and PC3 circuit cards from 2-EI-CB-30 will eliminate the problem with the reheat system power supply and return the power supply output to normal levels. This Temporary Modification, by restoring the power supply to normal output, will allow the manual override on the reheat FCVs to be cleared and allow them to function properly during a reactor trip without reliance on manual operator actions. The equipment affected by this Temporary Modification provides a control, not a protection function, and is not used at North Anna with the exception of the LP1 and LP2 turbine inlet temperature indicators.

The LP1 and LP2 temperature indicators on the Reheat Valve Control Panel may be disabled as an effect of this modification. Alternate temperature indications are available and will be used to monitor heatup and cooldown rates by procedural control should the LP1 and LP2 indicators be rendered inoperable by the selected card removal.

The most extensive case of card removal (removal of all three circuit cards - PC1, PC2 and PC3) is considered the most limiting configuration for this Temporary Modification. Therefore, this case is assumed in the body of this Safety Evaluation. Cases involving removal of only one or two of the cards are bounded by this evaluation of the most extensive case.

This Temporary Modification does not constitute an Unreviewed Safety Question for the following reasons:

1. Removing the card(s) from the system does not affect any automatic safety functions. The automatic temperature control aspect of the system has no safety function and is not currently in use at NAPS.

2. The reheat temperature control system is not Safety Related, has no Tech Spec requirements and is not described in the UFSAR. The probability or consequences of an accident are not affected.

3. Even in the case of the removal of all three circuit cards (PC1, PC2, and PC3), all functions of the reheat control system function other than the automatic temperature control function and the LP1 and LP2 inlet temperature indications on the Reheat Control Panel are unaffected. The operation of the RESET button as described in 2-E-0 is unchanged, as is normal closure of the FCVs in the event of a trip.

Therefore, this Temporary Modification does not involve an Unreviewed Safety Question and no changes are required to the Operating License.

00-SE-TM-18

Description
Temporary Modification 1135
This Temporary Modification will substitute a temporary RTD on the Mechanical Chiller bearing which presently has a defective RTD installed. Since this malfunctioning RTD is initiating a shutdown signal to the Chiller, another means of monitoring temperature is necessary in order to return the Chiller to service.

Summary
This temporary modification creates a means to insert the output from a temporary RTD installed on the Mechanical Chiller oil pump discharge piping into the point where the chiller bearing temperature RTD normally provides input to the shutdown circuit. The failed thrust assembly bearing temperature RTD (contained within the freon enclosure of the chiller) will have its leads lifted and taped and thus be out of service until the scheduled maintenance period for the chiller. The thrust assembly bearing is part of the shaft drive going to the freon pump impellers. Oil supplied to this bearing drains directly into the sump. The oil pump, which is located in the sump, pumps oil from the sump through an oil cooler, a filter and back to the thrust assembly bearing. The substitute RTD will be installed on the discharge of the oil pump between the pump and the oil cooler. The setpoint of the circuit is 205 degrees F and is preset. Adjustment of the setpoint could not be performed without further modifications to the protective circuits. Therefore, the setpoint will remain unchanged.

The temperature of the oil discharge piping where the substitute RTD will be installed is expected to be very similar to the existing RTD embedded in the thrust assembly bearing. The time delay is expected to be approx. 5 minutes from the time the temperature increases in the thrust assembly bearing to the time the oil discharge piping experiences the same temperature. The setpoint of 205 degrees F is indicative of bearing degradation. Degradation of the thrust assembly bearing is not expected to be so rapid that a 5 minute time delay would cause any additional significant damage to the bearing or the Mech. Chiller. The consequences of the bearing failure would be an increase in chilled water temperature, containment partial air pressure decreasing and containment temperature increasing. The CARF's would have to be swapped to service water.

Tech. Spec. 3.6.1.4 requires the containment partial internal air pressure to be maintained greater than or equal to 9.0 psia and Tech. Spec. 3.6.1.5 requires that containment average temperature be maintained greater than of equal to 86 degrees F and less than or equal to 120 degrees F.

The primary load on the Mech. Chiller are the Cont. Air Recirculation Cooling Coils, the gas stripper vent chillers, sampling coolers and the waste gas recombiner after cooler, as necessary.

Loss of chilled water to the CARF's is addressed by abnormal procedure AP-35.

00-SE-TM-19

Description
Temporary Modification # N2-1136
The lead from the flow switch on Vital Bus Inverter 2-VB-INV-04 to annunciator 2H-A4 will be lifted to defeat the alarm.

Summary
The flow switch on Vital Bus Inverter 2-VB-INV-04 (2-IV Inverter) is failing causing spurious alarms on main annunciator panel 2H-A4. The alarm is a common Vital Bus Inverter Trouble alarm that also gets inputs for low voltage (AC and DC), loss of sync. and ground. The purpose of the flow switch is to warn the operator of low air flow which is a sign of fan degradation or failure. The appropriate lead(s) will be lifted and taped to disable the input to the annunciator. Taping the lead(s) will provide reasonable assurance that the lead(s) will not come in contact with any other circuits.

The 2-IV Inverter which is rated for 15 KVA contains a fan that is located near the top of the cabinet. The fans normally draw air through louvers in the bottom of the cabinet and exhaust at the top. Loss of air flow due to the failure of the fan is alarmed in the Main Control Room. Loss of air flow will cause inverter temperatures to rise. However, the rise in temperature is not expected to be sufficient to cause failure of the inverter and the inverter could operate indefinitely without forced air flow. The Field Engineer from the manufacturer (Solid State Controls Inc.) has stated that inverters rated for less than 20 KVA normally are not supplied with fans for cooling. The inverters at NAPS have cooling fans only since they were requested by the buyer at the time of purchase. Component Engineering has stated that the Silicone Control Rectifiers will continue to function properly until the their temperature reaches approximately 200 degrees F. If the fan in the 2-IV Inverter failed, this temperature could be verified to be less than 200 degrees F by the use of thermography.

The control room alarm is currently an annoyance to the operator and could potentially distract the Operator at the Controls (OATC) from other more significant alarms. It may also prevent the crew from being aware of other real problems that may cause loss of the 2-IV inverter such as low voltage, loss of synchronization, and ground. The performance of the 2-VI Inverter will not be affected by this temporary modification (TM) and therefore the 2-IV Vital Bus will also be unaffected. A Non-Routine surveillance (1-LOG-14) or safeguards logs will be used when the TM is implemented to monitor the fan operation twice a shift. This will ensure the Control Room Operators are notified upon a failure of the fan. The fan and the flow switch are scheduled to be replaced with a different style assembly under Work Order 437371-04.

This Temporary Modification does not constitute an Unreviewed Safety Question for the following reasons:

1. Removing the lead to the low flow alarm input to the Vital Bus 2-IV Inverter Trouble annunciator (2H-A4) does not affect any automatic safety functions.

2. The low flow alarm input to the inverter trouble annunciator is not described in the UFSAR, only the inputs from input undervoltage, output undervoltage, synchronization failure, and input ground detection are mentioned. Since this TM is disabling the low flow input to the annunciator only, the probability or consequences of an accident are not affected.

3. The change does not directly impact any T.S. LCO or License Conditions. The T.S. requires that the 120 volt AC Vital Bus # 2-IV be energized from its associated inverter connected to the DC Bus # 2-IV. Disabling the low flow input to the Vital Bus 2-IV Inverter Trouble annunciator (2H-A4) will not affect the operation of the Inverter and thus, the T.S. margin of safety will not be decreased.

Based on the above discussion, an unreviewed safety question does not exist. This TM will enhance the Operator at the Controls ability to monitor the plant and not adversely affect the 2-IV Vital Bus.

00-SE-TM-20

Description
Temporary Modification N1-1693
A jumper will be installed across the bad leg of the transmitter switch for 1-GM-PT-112 to restore hydrogen pressure and purity instrumentation.

Summary
Hydrogen pressure, purity indication and alarm functions have been lost due to a bad leg on transmitter switch (SW-20) for 1-GM-PT-112. A point to point jumper will be installed across the bad leg of the switch in order to restore indication and restore annunciation.

The jumper will be installed on the bottom of cabinet 1-EP-CB-29 to bypass the bad leg of the switch. The gauge of the jumper wire will be sufficient to handle the minor electrical loading to restore gauge indication. This switch's function is primarily to disconnect the transmitter from the indicators during maintenance. Bypassing the switch will not affect the function of the indication during normal operation. Currently the indicators for both hydrogen pressure and purity have failed. The jumper will not interface with any other circuits other than those associated with 1-GM-PS-112 and will not affect loading on the electrical system.

The UFSAR requires that hydrogen gas control be maintained in the generator housing within predetermined limits of purity, pressure and temperature. Failure to perform this jumper could prevent operator action if the limits for purity and pressure are exceeded. The risk of affecting any other plant equipment is remote.

Based on the above discussion, an unreviewed safety question does not exist for this TM since this TM will restore indication of plant parameters necessary for maintaining the main generator in a safe operating condition.

SAFETY EVALUATION LOG
PROCEDURES
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S.E. #	Unit	Document	System	Description	SNSOC Date
99-SE-JCO-02 **REV. 1**	1,2	JCO-C-99-02 (R. 1) 1&2-E-0 1&2-ES-1.3 0-AP-36	HV	Primary ventilation Alignment Following a CDA to assure Filtration of ECCS Leakage - Revised to allow use of the 14-day AOT with 1 unit shutdown	3-15-00
00-SE-PROC-01	1,2	0-PT-82.14 (Rev. 9)		(1) Changes the jumper used to start the SBO DG from a wire to a wire with a switch; (2) the ACC sequence relay timers are removed from the acceptance criteria as per ET SE 99-075, Rev. 0; (3) reformatted procedure for usability	1-05-00
00-SE-PROC-02	1,2	0-MOP-21.50 (R. 0)		Aux. Bldg. Central Exhaust Fans 1-HV-F-8A, 8B, 8C -Removes fans 1-HV-F-8A, 8B, &/or 8C and/or their associated dampers from service for maintenance	2-17-00
00-SE-PROC-02 **REV. 1**	1,2	0-MOP-21.50 (Rev. 0 – P2)		Aux. Bldg. Central Exhaust Fans 1-HV-F-8A, 8B, 8C---- Removes fans 1-HV-F-8A, 8B, &/or 8C and/or their associated dampers from service for maintenance - The revision states that dampers AOD-103-3 & AOD-103-4 will be worked one at a time and AOD-103-1 & AOD-103-2 may be worked concurrent with each other.	2-25-00
00-SE-PROC-03	1,2	0-GOP-5.4 (Rev. 0)		Installation & Removal of Temporary Flanges on Turbine Building Low Volume Sump Pump Discharge Lines	2-29-00
00-SE-PROC-04	1	1-MOP-26.92		Adds installation of a jumper from 1-EP-MCC-1C1-1 to power reactor containment lighting panel 1RC4.	3-02-00
00-SE-PROC-05	1,2	1&2-PT-83.1 1&2-PT-83.2		Simulated Loss of Offsite Power Tests – A test switch will be used instead of the aux. relay cabinet	3-09-00
00-SE-PROC-06	1	1-OP-48.2 (Rev. 28) (OTO1)	CW	Adds steps to jumper around the Unit 1 elbow vacuum priming level switches on a CW pump start failure	3-13-00
00-SE-PROC-07	1	1-PT-210.19 (R. 8)		Revised to perform Cv testing on the accumulator discharge check valves	3-14-00
00-SE-PROC-08	1	1-TOP-14.1 (Rev. 0)		Returning PDTT Water to Cavity through RHR	3-17-00
00-SE-PROC-09	1,2	1&2-OP-8.6 UFSAR FN 2000-009	CH	Procedures control installation of a mechanical jumper between the VCT gas space & the primary sample sink to allow the VCT to be purged to the process vent. Also controls purging the VCT directly to the primary sample sink	3-18-00
00-SE-PROC-10	1	1-TOP-14.1 (Rev 0-P1)		Allows recovery of loop stop valve leakage from the PDTT pump suction to the RP system *00-SE-PROC-08 still applies to original version of procedure*	3-23-00
00-SE-PROC-11	1,2	1&2-PT-94.0		Changes the Startup Physics Testing measurement of integral rod worth using the rod swap technique	3-23-00

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S.E. #	Unit	Document	System	Description	SNSOC Date
00-SE-PROC-12	1,2	0-GOP-21.1 (R. 0) JCO C-2000-01 (R. 0) S.O. #229 (R. 0)	HV	Safeguards Ventilation Discharge Damper Temporary Air Supply	4-01-00
00-SE-PROC-13	1,2	1&2-OP-15.1		Attachment 6 is being revised to support a possible balance shot on the main turbine. S/G pressure control (& consequently reactor power control) will be transferred to the S/G PORVs instead of condenser steam dumps. Also allows power level at 2-5% with a maximum of 8% with OMOC permission instead of 1%	4-07-00
00-SE-PROC-14	1	1-MOP-8.32 (Rev. 2, P2)		Allows maintenance to be performed on the alternate charging header with ECCS leakage greater than the design basis leakage limits as long as the leakage can be isolated within 25 minutes.	4-15-00
00-SE-PROC-15	1,2	0-GOP-22.1 (R. 0)		"Operations Support for Removing Sludge from the Auxiliary Building Sump"	7-06-00
00-SE-PROC-16	1,2	1-MCM-1910-01		"Installation & Removal of Temporary Drinking Water Fountains in Radiologically Controlled Areas"	7-12-00
00-SE-PROC-17	1,2	1-EPM-B-1817-01 1-EPM-B-1817-02 2-EPM-B-1817-01 2-EPM-B-1817-02		Allows bypassing the contact on the interlock portion of the control circuit for the respective pump that is being tested so that the contacts on the opposite bus' charging pump will remain in the configuration to receive an auto start signal if no charging pumps are running on that particular unit	7-13-00
00-SE-PROC-18	2	2-PT-61.3 (Rev. 26 – P1-OTO1)		Addresses installation & testing of a blind flange in the Unit 2 high radiation sample system line to the Unit 2 containment sump to allow maintenance on 2-DA-TV-203A	7-20-00
00-SE-PROC-19	1	1-OP-5.7		Installs a temporary hose between an SI accumulator vent & a drain off of the RHR relief valve discharge line to provide an additional controlled source of nitrogen to blanket the RPT during RCS draindown.	8-31-00
00-SE-PROC-20	1,2	1-MOP-49.41 (R. 0)		1-CC-E-1B SW Side Chemical Cleaning	9-14-00
00-SE-PROC-21	1,2	1&2-MOP-5.97 (Rev. 0)		Allows isolated & drained reactor coolant loops to be returned to service by backfilling through the loop stops from the active portion of the RCS.	9-26-00
00-SE-PROC-22	1,2	0-MOP-50.2 (R. 1)		Allows filling the BC tower basin using the fire protection system when it is not desired to use the backfill method.	10-20-00
00-SE-PROC-23	1,2	1-OP-53.1 (R. 26-OTO1)		Adds instructions for pumping 1-EG-TK-2B to barrels to remove debris clogging the tank fill connection piping	11-14-00
00-SE-PROC-24	1,2	2-OP-50.2 (Rev. 11-OTO1)		Installs a jumper to bypass the low temperature interlock associated with the BC return valves to the BC tower, 1-BC-MOV-126 & 2-BC-MOV-226, to ensure these valves will open even if the tower outlet temperature is below 70°.	11-17-00

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PROCEDURES
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S.E. #	Unit	Document	System	Description	SNSOC Date
00-SE-PROC-25	1,2	0-OP-22.17 (R. 2)		Blowdown of the SW reservoir using the installed blowdown piping from the SW to LW systems.	11-21-00
00-SE-PROC-26	1,2	1&2-AP-19		Adds an attachment to the procedures that jumpers out the circuitry for the fire protection trip & the vibration switch trip of the bearing cooling fans. This prevents the BC fans from tripping due to a malfunction within the BC fan trip circuitry.	11-30-00
00-SE-PROC-27	1,2	1-WP-G99175 2-WP-G99176		New procedures to implement DCPs 99-175 & 99-176 to replace the containment air recirculation air-operated dampers with backdraft type dampers.	12-12-00
00-SE-PROC-28	1,2	0-MCM-1910-01 (Rev. 0)		Addresses installation & removal of temporary drinking water fountains inside the auxiliary building, fuel building, & containment buildings. Also addresses routing of temporary piping / tubing for domestic water supply to the fountains.	12-14-00

99-SE-PROC-13 Rev 1

Description

ET N 00-039, REV. 0, 0-GOP-5.5 (New Procedure)

In an effort to reduce the seasonal heat stress on various equipment in the EDG rooms, it is desired to lower the water stand-by temperatures for the Emergency Diesel Generator Engines during warm weather months, but maintain a temperature level that will meet the Tech Spec and UFSAR start requirements. This ET discusses the acceptability of reducing these temperatures from the normal temperature range of 130-135°F for the jacket water, air cooler and lube oil during standby condition. The Temporary Modification will lift a lead to the jacket coolant low temperature alarm, if required.

Summary

The two temperatures listed under Part A.7 will be monitored hourly until a new equilibrium temperature has been reached. After the temperatures have stabilized, and remain constant +/- 2 degF for 8 hours, the frequency can be moved to twice daily during the normal Safeguards Operator rounds per 1 / 2-LOG-6F. 0-GOP-5.5 (New Procedure), and 1 / 2-LOG-6F will be the formal tracking mechanism that will ensure the above temperature requirements are met per the Required Actions section of ET N 00-039, REV. 0.

This should only be initiated on one Emergency Diesel Generator at a time. After the above temperatures have stabilized above the specified limits, this may be performed on the remaining Emergency Diesel Generators, one at a time. This may be performed during the months of June through September.

If the jacket coolant low temperature alarm actuates (setpoint is 115 degF), the temporary modification may be performed to lift the lead to prevent a locked in Diesel Trouble Annunciator in the Main Control Room during the entire time the jacket coolant keepwarm pump and heater are secured. One lead may be lifted for the temporary modification, which will disable the jacket coolant low temperature alarm. To ensure the proper lead is lifted, a simultaneous verification will be performed. After relanding the lead, the function of the alarm will be verified.

Loss of Offsite Power to the station, section 15.2.9, is the only applicable accident analyzed per the SAR. This activity will not affect the ability to mitigate a Loss of Offsite Power to the station, or any other accidents as described in the UFSAR. Reducing the standby temperatures and lifting the lead on the jacket coolant alarm circuit per the TM, will have no effect on the ability of the Emergency Diesel Generator to perform its design function. The temperature changes and the TM will not impact the Emergency Diesel Generator's ability to start and pick up the required emergency loads, within the required 10 second time period under all accident conditions.

The jacket coolant temperature reduction during the warm weather months will only delay the changing of the AMOT valve position. The AMOT valves determine if coolant passes through the radiators or recirculates though the engine until design control temperature is reached in the jacket coolant system and the air cooler system. This will not adversely impact the combustion process or damage any diesel engine components as long as ambient temperature is maintained above 70 degF. This will also have no impact on the required start time of the engine, or the ability for the Emergency Diesel Generator to accept the required emergency loads.

The resulting reduction in lube oil temperature, caused by the reduction in the heat added by the jacket coolant system will not adversely affect the lubrication properties of the lube oil or damage any diesel engine components, as long as, the temperature to the engine is maintained at or above 110 degF. Also, the lube oil viscosity will be low enough, so that proper drainage back to the sump can occur through the small clearances in the lower crankshaft bearings. If the lube oil viscosity is not maintained low enough, the oil could migrate to the upper crankshaft, which in turn potentially cause exhaust stack fires or worse, hydraulic lockup. Maintaining lube oil temperature to the engine at or above 110 degF will have no impact on the required start time of the engine, or the ability for the Emergency Diesel Generator to accept the required emergency loads.

The Emergency Diesel Generator Keepwarm System and Low Jacket Coolant Alarm are not addressed in the Technical Specifications. As long as the above temperature limits are maintained, the Emergency

Diesel Generator reliability, and design function will be unaffected; therefore, an unreviewed safety question does not exist.

00-SE-PROC-01

Description
0-PT-82.14, SBO Diesel Generator Test (Start by Simulated LORSS Power), Rev. 9
Three changes are proposed by this revision.
The jumper used to start the SBO DG is changed from a wire to a wire with a switch. The AAC sequence relay timers are removed from the acceptance criteria in accordance with ET SE 99-075, Rev. 0. Procedure steps and formatting are reworked to improve usability.

Summary
Three changes are proposed to 0-PT-82.14, SBO Diesel Generator Test, by this revision. The jumper used to start the SBO DG is changed from a wire to a wire with a switch. The AAC sequence relay timers are removed from the acceptance criteria in accordance with ET SE 99-075, Rev. 0. Procedure steps and formatting are reworked to improve usability.

Implementation of these procedure changes has no effect on any existing accident precursors. Implementation of these procedure changes during performance of the PT will not affect any precursor events for equipment failure. Therefore, implementation of this revision will not increase the probability of occurrence of previously analyzed accidents or malfunctions of equipment.

Implementation of these procedure changes will not change the method in which SBO DG testing is performed. Use of a switch to control initiation of the test signal allows for finer control of the test and has no effect on the consequences of Chapter 15 accidents. Implementation of these procedure changes will have no effect on the failure of a SBO DG. Should the SBO DG fail, it would be no worse off with a switch failure, also. Therefore, implementation of this revision will not increase the consequences of previously analyzed accidents or malfunctions of equipment.

By implementing these procedure changes, a switch will be used to initiate the SBO DG test. Use of this switch within the previously evaluated jumper will not create a condition that could cause an accident of any type. Use of a switch to control the test initiation signal will create no other potential for equipment malfunction. Failure or misoperation of the switch will affect the test sequence only. Therefore, implementation of this revision will not create the possibility for an accident or malfunction of equipment not previously analyzed.

The margin of safety as described in the bases of the Technical Specifications will not be reduced by implementation of this revision.

Therefore an unreviewed safety question does not exist for either unit.

00-SE-PROC-02

Description

0-MOP-21.50, Auxiliary Building Central Exhaust Fans, 1-HV-F-8A, 1-HV-F-8B, and 1-HV-F-8C. Remove 1-HV-F-8A, 1-HV-F-8B, and/or 1-HV-F-8C and/or their associated dampers from service for maintenance.

Summary

The UFSAR describes the operation of each of the primary ventilation areas to include the ability to selectively pass the exhaust systems through the Iodine Filter Loops as it becomes necessary for decontamination. This includes exhaust from the Auxiliary Building (Central and General Exhaust), Decontamination and Waste Solids Building, Fuel Building, Containment (Shutdown Unit) and Engineered Safety Features Areas. During a design basis accident, the areas that require filtration include the Auxiliary Building Central, and the Engineered Safety Features Areas.

EOP procedures require alignment of the Auxiliary Building Central and Safeguards Area ventilation exhaust from these areas through the Filter Loops during a design basis LOCA. Two Safeguards and one Auxiliary Building Central fan will then be in operation. For other areas, the associated ventilation fans will be secured during the accident, and their dampers will be placed in the Bypass position. This alignment and associated leakage paths was previously evaluated in JCO-C-99-02.

0-MOP-21.50 is a new procedure that is being developed to allow maintenance of the Auxiliary Building Central Exhaust Fans and/or associated system dampers. This affects the charcoal filtration and Appendix R Central Exhaust systems.

During the maintenance, it will not be desirable to align any ventilation system through the charcoal filters. The maintenance evolution has the potential to affect Safeguards Ventilation due to leakage around the damper being worked. This same leakage path would also affect Containment Purge, Fuel Building, and Decontamination and Waste Solids Building, but they will not be in service during this activity. It is assumed that no active leakage will be in progress in the Safeguards area until recirculation mode is established.

In the event of a Loss of Coolant Accident that results in a transfer to containment sump recirculation, the Auxiliary Building Central Exhaust must be aligned to filtration mode within 29 minutes of the initiation of the event. This will prevent the unfiltered release of radiation due to ECCS leakage. This time limit has been previously evaluated as being sufficient to ensure filtration is available and aligned prior to the transfer to Containment Sump Recirculation.

The administrative controls established in the procedure provide assurance that the system may be returned to normal within 29 minutes of the initiation of a design basis accident. Only one damper may be worked at a time. A minimum of 1000 scfm ventilation flow will be verified for each running charging pump after the Appendix R hatches are opened and the Central Exhaust fans are secured. Calculation ME-0591, Charging Pump Cubicle Exhaust Flow Rate During Post-Accident Conditions, determined that 1000 scfm ventilation flow at a supply temperature of 120 degrees F is adequate for design basis operation of the charging pumps. If cubicle temperatures will exceed 115 degrees F, then ventilation from the Central Exhaust fans will be restored before the cubicles exceed 120 degrees F.

An Appendix R fire will require removal of the exhaust duct hatches on the running charging pumps and routing of the emergency supply duct. The MOP requires the removal of the duct hatches for additional cooling air flow and ambient temperature monitoring while the maintenance is in progress.

Maintenance activities on the central exhaust dampers and fans can not increase the probability of any accidents. This type of maintenance is not an accident precursor. By implementing this procedure, an unreviewed safety question does not exist.

00-SE-PROC-02, Rev. 1

Description
O-MOP-21.50, Auxiliary Building Central Exhaust Fans, 1-HV-F-8A, 1-HV-F-8B, and 1-HV-F-8C. Remove 1-HV-F-8A, 1-HV-F-8B, and/or 1-HV-F-8C and/or their associated dampers from service for maintenance. The revision states that dampers AOD-103-3 and AOD-103-4 will be worked one at a time and AOD-103-1 and AOD-103-2 may be worked concurrent with each other.

Summary
The UFSAR describes the operation of each of the primary ventilation areas to include the ability to selectively pass the exhaust systems through the Iodine Filter Loops as it becomes necessary for decontamination. This includes exhaust from the Auxiliary Building (Central and General Exhaust), Decontamination and Waste Solids Building, Fuel Building, Containment (Shutdown Unit) and Engineered Safety Features Areas. During a design basis accident, the areas that require filtration include the Auxiliary Building Central, and the Engineered Safety Features Areas.

EOP procedures require alignment of the Auxiliary Building Central and Safeguards Area ventilation exhaust from these areas through the Filter Loops during a design basis LOCA. Two Safeguards and one Auxiliary Building Central fan will then be in operation. For other areas, the associated ventilation fans will be secured during the accident, and their dampers will be placed in the Bypass position. This alignment and associated leakage paths was previously evaluated in JCO-C-99-02.

0-MOP-21.50 is a new procedure that is being developed to allow maintenance of the Auxiliary Building Central Exhaust Fans and/or associated system dampers. This affects the charcoal filtration and Appendix R Central Exhaust systems.

During the maintenance, it will not be desirable to align any ventilation system through the charcoal filters. The maintenance evolution has the potential to affect Safeguards Ventilation due to leakage around the damper being worked. This same leakage path would also affect Containment Purge, Fuel Building, and Decontamination and Waste Solids Building, but they will not be in service during this activity. It is assumed that no active leakage will be in progress in the Safeguards area until recirculation mode is established.

In the event of a Loss of Coolant Accident that results in a transfer to containment sump recirculation, the Auxiliary Building Central Exhaust must be aligned to filtration mode within 29 minutes of the initiation of the event. This will prevent the unfiltered release of radiation due to ECCS leakage. This time limit has been previously evaluated as being sufficient to ensure filtration is available and aligned prior to the transfer to Containment Sump Recirculation.

The administrative controls established in the procedure provide assurance that the system may be returned to normal within 29 minutes of the initiation of a design basis accident. Only one filter in service damper (AOD-103-3 and AOD-103-4) may be worked at a time. The filter bypass dampers (AOD-103-1 and AOD-103-2) may be worked concurrently A minimum of 1000 scfm ventilation flow will be verified for each running charging pump after the Appendix R hatches are opened and the Central Exhaust fans are secured. Calculation ME-0591, Charging Pump Cubicle Exhaust Flow Rate During Post-Accident Conditions, determined that 1000 scfm ventilation flow at a supply temperature of 120 degrees F is adequate for design basis operation of the charging pumps. If cubicle temperatures will exceed 115 degrees F, then ventilation from the Central Exhaust fans will be restored before the cubicles exceed 120 degrees F.

An Appendix R fire will require removal of the exhaust duct hatches on the running charging pumps and routing of the emergency supply duct. The MOP requires the removal of the duct hatches for additional cooling air flow and ambient temperature monitoring while the maintenance is in progress.

Maintenance activities on the central exhaust dampers and fans can not increase the probability of any accidents. This type of maintenance is not an accident precursor. By implementing this procedure, an unreviewed safety question does not exist.

00-SE-PROC-03

Description

0-GOP-5.4 Installation and Removal of Temporary Flanges on Turbine Building Low Volume Sump Pump Discharge Lines

Temporary flanges will be installed and removed on the Turbine Building low volume sump pump discharge lines. This will allow the installed piping to be used with a temporary pump to dewater the sump for outage activities.

Summary

Reduction of the amount of water in the Turbine Building sumps is desired for routine outage activities. This facilitates maintenance and allows periodic cleaning when the sump is not needed to support plant operations.

Therefore, a flange with a valve and hose fitting will be installed on a blank flange located just downstream of the installed sump pumps discharge check valves. This is considered a procedurally controlled temporary modification. A temporary pump, powered from the Service Air system, will remove water from the associated sump and flow through this temporary flange into the installed piping. This will eliminate the need to run temporary hoses across the Turbine Building floor to the next sump.

The fitting and flange connections will be functionally tested by the initial flow from the sump to the installed piping. If there is leakage, then the temporary pump will be secured and corrective actions will be initiated. Restoration of the system will be via the work order process.

The Turbine Building Sump Pumps and the associated piping system are not safety related, nor are they required by Technical Specifications or described in the Bases. Abnormal Procedures AP-23, Oil or Hazardous Spill, and AP-24, Steam generator Tube Leakage provide instructions for securing all Turbine Building Sump Pumps in order to prevent an unwanted discharge to the discharge canal. By performing the actions contained within 0-GOP-5.4, preventing unwanted releases will not be adversely effected.

All existing instrumentation, controls and flow paths will be unaffected. The associated pump discharge check valve will prevent the flow from recirculating to the sump and from possibly causing damage to the installed pumps.

All activities will take place outside of any radiologically controlled areas. No known radioactive substances will be involved. Levels of radiation or airborne radioactivity will not increase.

Steps are contained within 0-GOP-5.4 to control the temporary modification and to ensure proper removal and testing.

Dewatering the Turbine Building sumps via a temporary pump and into the installed discharge piping does not increase the probability of, nor increase the consequences of, nor create the possibility of accidents different than, any Chapter 15 accidents. No new flow path to the environment is created nor is any existing flow path adversely impacted by this activity.

Therefore, no unreviewed safety question exists and the activity should be allowed.

00-SE-PROC-04

Description
1-MOP-26.92
The revision to 1-MOP-26.92 adds the installation of a jumper from 1-EP-MCC-1C1-1 to power Reactor Containment lighting panel 1RC4.

Summary
1-MOP-26.92 is being revised to allow the installation of a jumper from 1-EP-MCC-1C1-1 to power Reactor Containment lighting panel 1RC4 while 1-EP-MCC-1A1-1 is removed from service for maintenance. The normal electrical protection of the affected breaker and the bus are not altered by the temporary modification. No permanent modifications are being made and there is no effect on the environment as a result of this procedure implementation.

Both 1-EP-MCC-1C1-1 and 1-EP-MCC-1A1-1 are non safety related, station service 480 volt MCCs. The procedure is written such that it may be performed in any mode. Initial conditions of the MOP ensure plant conditions can support removal of 1-EP-MCC-1A1-1 for maintenance. Procedure controls are in place to ensure all Technical Specification and Technical Requirement Manual requirements are satisfied. The impact of the loss of specific loads, including 1-CC-MOV-100A and 1-MS-NRV-101A, has been previously evaluated. This revision includes the temporary modification to power containment lighting from another station service bus.

The procedure revision does not increase the probability of occurrence nor the consequences of an accident or malfunction previously addressed in the UFSAR and no new accident or malfunction is created.

00-SE-PROC-05

Description
1-PT-83.1, 2-PT-83.1, 1-PT-83.2, 2-PT-83.2
Jumpers 1 and 2 are for the SW spray and bypass MOVs and jumpers 8 and 9 are for the breakers 15/25H11 and 15/25J11. Instead of going from the Auxiliary Relay Cabinet to the SSP Cabinet, a test switch will be used to accomplish the same goal. The test switch will be similar to that used in 1/2-PT-82.8.

Summary
The Simulated Loss of Offsite Power With ESF Actuation test procedures are being modified by ET SE 99-077, Streamlining 1/2-PT-83.1/83.2 and 1/2-PT-71.4. The ET no longer requires that jumpers 1, 2, 8 and 9, on Attachment 3 of the procedures, go from the Auxiliary Relay Cabinet to the Solid State Protection Cabinet for the Loss of Reserve Station Service Power (LORSSP) section. In place of these current, approved jumpers, the test switch will be used to perform the same function, in conjunction with the same contacts within the Solid State Protection Cabinet. The test switch is similar to that used in Attachment 3 of 1/2-PT-82.8, Simultaneous Diesel Start Test. Using the test switch will save significantly in test set-up time and increase personnel safety by eliminating the need to enter the energized Auxiliary Relay Cabinet. The only technical change that the test switch will cause is the elimination of the 7-second delay that occurs when initiating the LORSSP portion of the test. This one change has no impact on equipment performance or operability and the 7-second delay will be tested in 1/2-PT-71.4. All plant equipment affected by the test will operate within their design parameters.

The Design Basis Accident considered during this review was the Loss of Offsite Power. The test switch will be used instead of the current jumpers during the LORSSP functional testing. The test switch will be used to verify that the effected equipment will operate properly during a LORSSP and will not increase the probability that such an accident will occur. The use of the test switch will not increase consequences of a LORSSP accident either. No new accident or malfunction will be created because all of the equipment used during the test will be operated within its normal design capabilities. There will be no reduction in the margin of safety as described in the T.S. basis section. Therefore, no unreviewed safety question exists for the use of the test switch in either Unit's Simulated Loss of Offsite Power With ESF Actuation test procedures.

00-SE-PROC-06

Description
1-OP-48.2, "Operation of Circulating Water System," Rev. 28
This procedure change adds steps to jumper around the Unit 1 Elbow Vacuum Priming Level Switches on a Circulating Water Pump start failure.

Summary
This procedure change adds steps to jumper around the Unit 1 Elbow Vacuum Priming Level Switches on a Circulating Water Pump start failure. This jumper will bypass elbow vacuum priming switches that have failed. The procedure verifies that the CW elbow is full by checking the affected indicating light prior to continuing with start of the CW pump. The CW pump will not be started, whether or not the jumper is used, if the indicating light is not lit.

Use of a jumper to manually start a Circulating Water pump is not part of any accident precursor. Installation of this jumper will only affect the manual start of a Circulating Water pump. Either other pumps will be running to provide a heat sink, or the condenser will not be available due to insufficient Circulating Water pumps running. Therefore, implementation of this procedure change will not increase the probability of occurrence of accidents or malfunctions of equipment that have been previously evaluated.

No mitigating equipment is affected by the use of a jumper to bypass the Elbow Vacuum Priming switch on a manual Circulating Water pump start. This jumper will only affect the ability to manually start a Circulating Water pump. The inability to start a Circulating Water pump will not affect the consequences of a loss of Condenser vacuum. Therefore, implementation of this procedure change will not increase the consequences of an accident or malfunction of equipment that has been previously evaluated.

Installation of this jumper has no effect on any equipment or systems that could create an accident. This jumper only affects the ability to manually start a Circulating Water pump. No other equipment is affected. Therefore, implementation of this procedure change will not increase the probability of occurrence of an accident or malfunction of a different type than previously evaluated.

No changes are proposed to the bases of the Tech Specs.

No changes are proposed to the Operating License or Tech Specs.

Therefore, no Unreviewed Safety Question exists.

00-SE-PROC-07

Description
1-PT-210.19 Rev 8 "Inservice Inspection SI Accumulator Discharge Check Valves Full Open Test"
1-PT-210.19 is being modified to perform CV testing on the accumulator discharge check valves. The following instrument loops will be removed from service to support installation of temporary test equipment: 1-SI-P-1921, 1-SI-P-1925, 1-SI-P-1929, 1-SI-L-1920, 1-SI-L-1924, 1-SI-L-1928, and 1-RS-P-156A. This test will also require the installation of temporary test instruments to 1-RS-PT-156A. This process will involve the removal of the cover for 1-RS-PT-156A, an EQ pressure transmitter.

Summary
The original issue of 1-PT-210.19 was evaluated by Safety Evaluation 95-SE-PROC-31 as "Not constituting an Unreviewed Safety Question."

The change proposed by this document is an enhancement to the original procedure that utilizes special test instrumentation to provide a different means of evaluating the accumulator check valve operability. While the method of initiating accumulator flow remains the same, the operability of the check valves is judged based on the flow rate per unit time that occurs when the accumulator isolation valves are opened. The new method, known as the "CV Test" uses measurement of accumulator pressure change and level change to determine flow from the accumulator. Measured flows are compared with accident analysis flows and if the analysis flow is exceeded that it follows that the check valves are operable.

To provide the necessary data for the CV method, the procedure will take one level channel and one pressure channel for each accumulator and install the necessary test transmitters. These test transmitters will be "jumpered" into the field wiring of the channels used to transmit the data to the Primary Process Racks where the data is collected by a high-speed recorder. The installation and removal of the test instrumentation and the jumpering and recovery of the field wiring is controlled by the procedure. This ensures that the instrument channels used in the testing are returned to service and operable when the test is completed.

During testing the control room has indication of accumulator status by means of the redundant level and pressure channels that remain operable.

None of the testing by this procedure overlaps or compromises the operators' capability to maintain the essential needs of the reactor for cooling. Since the testing is conducted into the refueling cavity in Mode 6 the test in fact complements the RCS inventory.

The instrumentation added by the procedure change being evaluated does not change the basic scope or method of testing; it does however provide a more detailed picture of the test.

Since the main test scenario is not changed and since none of the test instrumentation interferes with mode 6 operations, the probability of an accident or malfunction previously analyzed is not increased.

The accumulator dump testing is within the scope of operations in the Mode 6 environment and the instrumentation added does not change the testing scope; therefore there is no possibility of the occurrence of an accident different from those analyzed.

Based on a previous analysis of this testing sequence the margin of safety as defined in the basis for Technical Specifications for Mode 6 operations is not affected.

Based on the above, the proposed changes to 1-PT-210.19 do not constitute an Unresolved Safety Question.

00-SE-PROC-08

Description
1-TOP-14.1
A new TOP is being developed to allow recovery of loop stop valve leakage from the PDTT pump suction to RHR. This procedure will allow the installation of a hose, an air pump and a check valve between the suction of the PDTT pump and an LMC valve at the RHR pump suction.

Summary
A temporary modification is to be added to procedure 1-TOP-14.1 as an alternate method for loop stop valve leakage recovery. This procedure will allow the installation of a hose, an air pump and a check valve between the suction of the PDTT pump and a LMC valve at the RHR pump suction.

The temporary modification will be leak checked when placed in service. Failure of the hose would result in water from the PDTT being pumped on to the containment floor until the leak is terminated. The Loop Stop Valves will be closed during the period that this temporary modification is installed which will limit any leakage to the PDTT. Water from the RHR system will be preserved by the check valve that is to be installed near where this Temporary Modification ties into the RHR system. This procedure will only be used during a de-fueled condition, so the safety significance is negligible.

Failure of the temporary check valve could cause a reduction in Refueling Cavity and Spent Fuel Pit level; however, this Temporary Modification will normally be used with the transfer canal gate valve closed. Maintaining the transfer canal gate valve closed is not a procedural requirement and its configuration does not affect this evaluation. This TM will only be installed when the unit is de-fueled and it will be removed prior to core on-load. The transfer canal gate valve is normally closed in this condition.

Implementation of this TM will not increase the probability of occurrence of an accident or malfunction of equipment previously analyzed.

Failure of the TM will not affect equipment and systems used to respond to the considered accidents. The ability to provide makeup to the RCS and cavity are not reduced by implementing this TM. Implementation of this TM has no effect on systems or equipment required to provide backup cooling to the reactor vessel or spent fuel pit. Therefore, implementation of this TM will not increase the consequences of an accident or malfunction of equipment previously analyzed.

The TM will be installed with no fuel in the Reactor Vessel, when the core cooling function of RHR is not required. Catastrophic failure of the TM could result in a loss of Cavity inventory; however, even if the transfer canal gate valve were open to the Spent Fuel Pool, the leakage would be detected locally or remotely from MCR indications and would be isolated locally prior to the development of any adverse inventory condition. The TM will not interface with other systems that are required for any safety function. Therefore, implementation of this TM will not create the possibility of an accident or malfunction of equipment not previously analyzed.

Implementation of this jumper has no effect on the basis section of the Tech Specs. Therefore, the margin of safety as defined in the bases to the Tech Specs is not reduced.

For these reasons, an Unreviewed Safety Question does not exist.

00-SE-PROC-09

Description
1/2-OP-8.6 Volume Control Tank Operation
UFSAR Change FN 2000-009
This procedure controls the installation of a mechanical jumper between the VCT gas space and the Primary Sample Sink to allow the VCT to be purged to the Process Vent System. The procedure also controls purging of the VCT directly to the Primary Sample Sink.

Summary
1/2-OP-8.6 controls the operation of each unit's Volume Control Tank (VCT). Sections are being added to each procedure to allow purging the VCT gas space to the Process Vent System via the Primary Sample Sink or to the sample sink directly. Purging to the Process Vent System is accomplished by installation of a mechanical jumper which is simple in nature. Purging of the VCT is normally provided by an installed pressure control valve to the Gaseous Waste system (i.e. Waste Gas Decay Tanks). This jumper allows the VCT to be purged to the Process Vent system in order to minimize the oxygen addition to the Waste Gas Decay Tanks.

This purge operation will remove gaseous material from the VCT while taking its pressure from approximately 30 psig to 16 psig. If a VCT rupture would occur, it would remove the motive force for the purge and the purge would essentially stop. Relatively low pressures and temperatures are used with this mechanical jumper. By maintaining the specified VCT pressure range, no adverse affect will be generated on other plant systems.

This jumper uses operator controls versus the installed pressure control valve for maintaining minimum VCT pressure greater than log specifications. This is an acceptable practice for this application. The minimum VCT pressure is required for keeping adequate back pressure on the Reactor Coolant Pump (RCP) seals. The necessary back pressure will be maintained.

A flex hose will be connected between 1-SS-15 and 1-SS-159 for Unit 1 and between 2-SS-145 and 2-SS-97 for Unit 2. The hose and its connections are rated for the pressures of the VCT. The gas purge will be contained within systems designed for the process parameters. Installed radiation monitors for the Process Vent system will monitor the purge and provide indication if the purge should be terminated. High radiation alarms will be handled in accordance with existing plant procedures.

For purging to directly to the sample sink, the "A" Ventilation Stack Gaseous Radiation monitor, 1-VG-RM-104, will be monitored and will provide indication if the purge should be terminated.

Implementation of these procedure changes has no effect on any existing accident precursors nor any precursors for equipment failure. Therefore, implementation of these procedure changes will not increase the probability of occurrence of previously analyzed accidents or malfunctions of equipment.

Implementation of these procedure changes will not increase the consequences of previously analyzed accidents or malfunctions of equipment. All accidents remain bounded by the accident analyses.

Neither the VCT nor the primary sample sink is required by Technical Specifications. Therefore, the margin of safety as defined in the bases of Technical Specifications will not be reduced by implementation of these changes.

Therefore, an unreviewed safety question does not exist for either unit and these changes should be allowed.

00-SE-PROC-10

Description
1-TOP-14.1 Revision 0, P1
This TOP change is being developed to allow recovery of loop stop valve leakage from the PDTT pump suction to the RP system. This procedure will allow the installation of a hose, an air pump and a check valve between the suction of the PDTT pump and a vent valve on the RP system back to the Refueling Cavity.

Summary
A temporary modification is to be added to procedure 1-TOP-14.1 as an alternate method for loop stop valve leakage recovery. This procedure will allow the installation of a hose, an air pump and a check valve between the suction of the PDTT pump and a vent valve on the RP system to the Refueling Cavity. The original procedure evaluated returning the leakage through the RHR system. This change will allow recovery of the leakage when the RHR system is unavailable. Only the changes to use the RP system versus the RHR suction LMC are evaluated in this Safety Evaluation. Safety Evaluation 00-SE-PROC-08 still applies to the original version of this procedure.

The temporary modification will be leak checked when placed in service. Failure of the hose would result in water from the PDTT being pumped on to the containment floor until the leak is terminated. The Loop Stop Valves will be closed during the period that this temporary modification is installed which will limit any leakage to the PDTT. Water from the RP system will be preserved by the check valve that is to be installed near where this Temporary Modification ties into the RP system. This procedure will only be used during a de-fueled condition, so the safety significance is negligible.

Failure of the temporary check valve could cause a reduction in Refueling Cavity and Spent Fuel Pit level; however, this Temporary Modification will normally be used with the transfer canal gate valve closed. Maintaining the transfer canal gate valve closed is not a procedural requirement and its configuration does not affect this evaluation. This TM will only be installed when the unit is de-fueled and it will be removed prior to core on-load. The transfer canal gate valve is normally closed in this condition.

Implementation of this TM will not increase the probability of occurrence of an accident or malfunction of equipment previously analyzed.

Failure of the TM will not affect equipment and systems used to respond to the considered accidents. The ability to provide makeup to the RCS and cavity are not reduced by implementing this TM. Implementation of this TM has no effect on systems or equipment required to provide backup cooling to the reactor vessel or spent fuel pit. The design function of the RP system will not be adversely affected by this TM. Therefore, implementation of this TM will not increase the consequences of an accident or malfunction of equipment previously analyzed.

The TM will be installed with no fuel in the Reactor Vessel, when the core cooling function of RHR is not required. Catastrophic failure of the TM could result in a loss of Cavity inventory; however, even if the transfer canal gate valve were open to the Spent Fuel Pool, the leakage would be detected locally or remotely from MCR indications and would be isolated locally prior to the development of any adverse inventory condition. The TM will not interface with other systems that are required for any safety function. Therefore, implementation of this TM will not create the possibility of an accident or malfunction of equipment not previously analyzed.

Implementation of this jumper has no effect on the basis section of the Tech Specs. Therefore, the margin of safety as defined in the bases to the Tech Specs is not reduced.

For these reasons, an Unreviewed Safety Question does not exist.

00-SE-PROC-11

Description
1-PT-94.0, 2-PT-94.0
Change the Startup Physics Testing measurement of integral rod worth using the rod swap technique as follows:

1) After determining the worth of a test bank (i.e., the test bank is fully inserted and the reference bank at the measured critical position), swap the test bank out of the core using the next test bank. Currently this swap is done using the reference bank. When the previous test bank is fully withdrawn, use the reference bank to complete the swap of the current test bank into the core.
2) Calculate the boron concentration drift over the entire test. Currently, the boron concentration drift is calculated over each test bank measurement.

Summary
A safety evaluation has been performed to determine whether an unreviewed safety question will result from a change to the Startup Physics Testing measurement of integral rod worth using the rod swap technique.

The process currently used is as follows:

1. The control rod bank with the largest predicted integral rod worth is defined as the reference bank. The reference bank's worth is determined by diluting the reference bank into the core from an all-rods-out configuration.
2. The initial state point for the measurement of integral rod worth using rod swap is the reference bank fully inserted and the test bank (the control rod bank to be measured) fully withdrawn.
3. Swap the test bank with the reference bank until the test bank is fully inserted and the reference bank is at the measured critical position (MCP).
4. Calculate the integral worth of a test bank by determining the measured integral worth of the reference bank between the reference bank fully inserted and the reference bank at the measured critical position.
5. Swap the test bank with the reference bank until the test bank is fully withdrawn and the reference bank fully inserted (this is the final state point for this test bank and the initial state point for the subsequent test bank).
6. Confirm that boron drift (the difference in reactivity between the initial and final state points after adjustment for temperature drift) is less than 10 pcm (consequence of failing test is to repeat rod swap test for this bank).
7. Repeat steps 3 through 6 for each remaining test bank.

Note that all of the above steps, with the exception of how boron drift is addressed, is documented in the Virginia Power Rod Swap Topical Report but <u>not</u> specified in the Topical Report SER. The proposed changes to this process modify steps 5, 6, and 7 as follows:

5. Swap the test bank with the subsequent test bank (preferably a higher worth bank) resulting in a final state point where the test bank is fully withdrawn, the subsequent test bank is fully inserted, and the reference bank is at a new measured critical position for the subsequent test bank.
6. Repeat steps 4 and 5 for each remaining test bank
7. Confirm that boron drift (the difference in reactivity between the first test bank's initial state point and the last test bank's final state point after adjustment for temperature drift) for the measurement of the integral rod worth for all test banks is less than 25 pcm (consequence of failing test is to repeat rod swap test for all banks).

The occurrence of boron drift during rod swap is highly unusual (a review of data for the four most recent North Anna Startup Physics Tests revealed essentially zero boron drift) and the worst consequence would be the loss of time due to having to repeat the entire rod swap test.

This change does not represent an unreviewed safety question because:

- There is no increase in the probability of occurrence of an accident. Potential causes of rod accidents include reactor control and control rod drive systems, and operator error. The process of swapping a test bank out of the core, while inserting another test bank rather than the reference bank, does not impact the reactor control and control rod drive systems, or increase the likelihood of operator error. Therefore, there is no increase in the probability of occurrence of these accidents.
- There is no increase in the consequences of an accident. The process of swapping a test bank out of the core, while inserting another test bank rather than the reference bank, has no impact on shutdown margin. Therefore, the ability to achieve and maintain safe shutdown is unchanged.
- The possibility of a new or different type of accident, other than those already analyzed, is not created. The process by which the integral rod worth of a control rod bank is measured only involves the movement of control rod banks. Currently analyzed control rod accidents bound the possible occurrences.
- There is no reduction in the operator's ability to control and monitor the plant. Close monitoring of flux and reactivity is monitored throughout Startup Physics Testing.
- The margin of safety is not affected by this change. The limiting conditions for operation, as defined in the Technical Specifications that apply to Startup Physics Testing, are not affected.

00-SE-PROC-12

Description

JCO C-2000-01, Standing Order #229 - Rev. 0, and 0-GOP-21.1

This evaluation concludes that even with a loss of air supply to the Safeguards exhaust fan discharge dampers, compensatory measures are adequate to monitor the condition and promptly restore Safeguards Area exhaust. Temporary loss of Safeguards Area ventilation will not cause temperatures to exceed the equipment qualification limits.

Summary

Background

The LOCA design basis accident assumes that ventilation from areas with ECCS leakage will be aligned to the charcoal filters for 30 days following the accident (UFSAR 15.4.1.7). Following a LOCA concurrent with loss of Instrument Air, air bottles are installed to hold the Safeguards Area exhaust fan discharge dampers open for 30 days. Plant Issue N-1999-3043 identified the fact that surveillance procedures did not ensure the system could perform this function, so procedures 1-PT-77.12A & 12B, and 2-PT-77.12A & 12B were developed accordingly.

1-PT-77.12A (on 1-IA-TK-5A) performed on 3/23/00, and 1-PT-77.12B (on 1-IA-TK-5B) performed on 3/26/00, were both unsatisfactory. Based on past surveillance history, it is suspected that neither of the air supplies to the Unit 2 Safeguards Area exhaust fan discharge dampers would satisfy the acceptance criteria of their respective Periodic Test Procedures.

Safety and Regulatory Issues

The Safeguards Area exhaust fans provide two safety-related functions: a) cooling to the safety-related pumps in Safeguards Area, and b) ensure radioactive materials leaking form ECCS equipment are filtered prior to being released to the environment. Temporary failure of the fan discharge dampers could prevent these two functions from being met. Compensatory measures are provided to promptly restore ventilation airflow. The small amount of heat build-up that occurs during the time that compensatory measures are being taken is bounded by temperature calculations reflected in the EZD. (The EZD, Zone SFGD-1, reflects higher Safeguards Area temperatures allowable for 24 hours following a LOCA concurrent with loss of normal ventilation.) The manual actions of this compensatory measure are consistent with the design basis for the Instrument Air and Ventilation systems. The IA system design includes manual actions to restore IA during an accident (UFSAR 9.3-4). Also, the ventilation system design includes manual actions to align filtration during an accident (UFSAR 15.4.1.7).

Evaluation of Mitigating Actions

Compensatory measures will be established to ensure that a motive force (air) is available to the Safeguards Area Exhaust Fan Discharge Dampers. Manual Operator Action may be required (based on plant conditions following a LOCA) to provide this motive force after a DBA. Guidance has been provided to the Operations staff (Standing Order # 299) on how to assess plant data and what manual actions are needed to establish backup air to the Safeguards Area Exhaust Fan Discharge Dampers. The manual Operator actions are not considered heroic since;

1. The guidance is in a procedure (Standing Order # 229 and 0-GOP-21.1).
2. The doses in the area are not excessive after 24 hours (conservative calculation). Manual actions can be accomplished within a short duration.
   - 2 hours = 210 Rem/hour
   - 8 hours = 81 Rem/hour
   - 24 Hours = 32 Rem/hour
   - 96 hours = 12 Rem/hour
3. The actions are considered "skill of the craft" for Operations personnel.
4. The actions do not need to be established in a hurried manner (> 24 hours after a DBA).
5. The actions are only required in the very unlikely event of a LOCA concurrent with IA System failure.

Unreviewed Safety Question Determination

Compensatory measures as proposed in this JCO create no unique precursors or impacts on Chapter 15 accidents. The consequences of an accident are not increased since normal ventilation will be promptly restored. The ESF Area ventilation system will continue to receive its automatic swap-over to the HEPA/charcoal filter banks. Operators will monitor indications of potential excessive temperatures in the Safeguards Area, and establish backup air to the Safeguards Area Exhaust Fan Discharge Dampers if needed in accordance with proposed procedures to mitigate high temperature conditions and prevent environmental challenges to equipment. The margin of safety as defined in the TS bases is not reduced. For these reasons, an unreviewed safety question does not exist.

00-SE-PROC-13

Description

1/2-OP-15.1, Attachment 6, Balance Shot of the Main Turbine

Attachment 6 of 1/2-OP-15.1 is being revised to support a possible balance shot on the main turbine. Steam Generator pressure control (and consequently Reactor power control) will be transferred to the Steam Generator PORV's instead of Condenser Steam Dumps.

Summary

Attachment 6 of 1/2-OP-15.1 is being revised to support a possible balance shot on the main turbine. Steam Generator pressure control (and consequently Reactor power control) will be transferred to the Steam Generator PORV's instead of Condenser Steam Dumps. In its initial form, Attachment 6 of 1/2-OP-15.1 called for reactor power to be stabilized at approximately 1 percent. This revision increases that to a nominal 2-5 percent, with a maximum of 8 percent (with Operations Manager On Call permission), in order to obtain more stable plant operation.

Each PORV is rated at 425,244 lbm/hr of saturated steam at 1025 psig, which is approximately 10 percent of the Required Total Capacity of each bank of five Safety Valves, 4,255,542 lbm/hr. Eight percent reactor power represents a significant fraction of the total capacity of the atmospheric steam dumps. The UFSAR does not specifically address operation of the main steam PORV's as a means of stable plant operation. The UFSAR states that the PORV's are set to avoid opening the main steam safety valves and they are used to achieve a controlled cooldown of the reactor in the event condenser steam dumps are not available. As specifically stated in the UFSAR, the decay heat release valve can be used during reactor physics testing and unit hot standby conditions. The main steam PORV's perform a safety related function of cooling down the reactor coolant system in preparation for RCS depressurization to recover from a steam generator tube rupture. The emergency operating procedures rely on the PORV's in the event that a loss of offsite power or other upset condition renders the condenser steam dumps unavailable. As stated in the Basis section of the Technical Requirements Manual, the design basis of the Steam Generator PORV's is established by the capability to cool the unit to RHR entry conditions. Taken together, these statements would imply that the atmospheric steam dumps are not normally used to maintain stable plant conditions.

Each PORV is rated at approximately 10 percent of steam flow at rated power. The procedure restricts reactor power to a maximum of 8 percent. This margin to rated capacity should prevent the inadvertent opening of a main steam safety valve and postulated failure to re-close. The inadvertent opening of a main steam safety valve would occur as a result of unstable steam generator control, leading to required steam flow from one PORV exceeding its capacity. Power is procedurally limited to a maximum of eight percent, which is less than the rated capacity of the atmospheric steam dumps. A power level of approximately 2-5 percent is chosen to enhance stable plant operation. Operators are trained to avoid a power increase in the event power decreases to below the point of adding nuclear heat. Therefore, implementation of this change will not increase the probability of occurrence of accidents or malfunctions of equipment previously analyzed.

The worst case postulated failure for this evolution is the inadvertent opening of a main steam safety valve, with a subsequent failure to re-close. This is well within the bounds of the associated accident analysis. Consequences are not affected in any way. All protective equipment will be operable during this evolution. Therefore, implementation of this change will not increase the consequences of an accident or malfunction of equipment previously analyzed.

As stated previously, the most credible failure is the inadvertent opening of a main steam safety valve. This can happen in the event of unstable steam generator control. The only possible accidents associated with this activity are the inadvertent depressurization of the main steam system or an unanticipated reduction in reactor power, followed by a subsequent power escalation. These are within the bounds of plant design. Therefore, implementation of this change will not increase the probability of occurrence or consequences of accidents or malfunctions of equipment not previously analyzed.

All required systems will remain fully operable. The atmospheric steam dumps will remain fully operable as required by TRM Section 3. Since all equipment remains operable, the margin of safety is not affected in any way.

For these reasons, an Unreviewed Safety Question does not exist.

00-SE-PROC-14

Description
1-MOP-8.32, Alternate Charging Header Maintenance
The primary reason for this change is to allow maintenance to be done on the alternate charging header with ECCS leakage greater than the Design Basis Leakage Limits as long as the leakage can be isolated within 25 minutes.

Summary
To facilitate repair to the Alt. Charging Header where isolation cannot be maintained and GDC-19 criteria may come into question, the procedure 1-MOP-8.32, Alternate Charging Header Maintenance, has been PAR'ed. The procedure change to 1-MOP-8.32 is to allow repairs to continue on the alternate header even if the Design Basis Leakage Limit of 12,000 cc/hr is exceeded as long as there is the capability to isolate the leakage with 25 minutes.

The current conservative criterion for ECCS leakage limits is as follows:

-600 cc/hr--The operational ECCS limit of leakage in unfiltered areas during normal operation.
-900 cc/hr--The combined operational ECCS limit of leakage in filtered plus unfiltered areas during normal operation.
-1200 cc/hr--During LOCA conditions, it is maintained that unfiltered leakage will remain under this limit in
order to maintain the control room habitability during this accident.
-12,000 cc/hr--During LOCA conditions, it is maintained that filtered leakage will remain under this limit in
order to maintain the control room habitability during this accident.

These limits have been established for everyday operation and LOCA conditions. The operational ECCS limits are established for normal day-to-day operations and does not require any manual operator action to ensure GDC-19 limits are met for control room habitability. Likewise, for LOCA conditions the control room personnel are protected if these limits are sustained. It is upheld that during the maintenance of the alternate header that ECCS leakage criterion can be readily established to meet these leakage limits within 25 minutes of a LOCA.

During the maintenance activities, a potential leakage path to the aux. building atmosphere will exist due to potential leakby of the isolation valves. Such a leakage path will not create adverse conditions in a non-LOCA situation. However, the leakage path would be a concern if it existed during the Post-LOCA recirculation mode of operation, since the leakage may be greater than the 1200 cc/hr, and allow the release of containment sump water to the aux. building atmosphere placing the control room habitability and GDC-19 into question. Therefore a dedicated operator in direct communication with the Control Room will be at the Alternate Charging Header vent valve or any drains valves at all times when any leakage path exists from the CVCS system to the atmosphere so that any leakage path can be isolated within 25 minutes. In addition, Maintenance will be in contact with the Control Room and will effect repairs within the 25 minutes. This will ensure that external ESF component leakage will not exceed 1200 cc/hr, and the current LOCA dose limits remain bounding. The manual actions were evaluated to determine the feasibility of their performance. NAF calculations show that the minimum time to recirculation mode operation is 29.3 minutes. With a limit of 25 minutes to perform the actions to isolate the leakage, the actions can be initiated from the occurrence of a LOCA and completed prior to the initiation of recirculation mode operation with an additional 4 minutes to exit the area thus protecting control room habitability before recirculation mode is initiated.

System Engineering and Maintenance reviews have concluded that repairs can be effected within the 25 minute time frame.

There is also reasonable assurance that the maintenance will be successful. But if maintenance does fail and the ECCS leakage limit is greater than the 1200 cc/hr limit in unfiltered areas, an evaluation of the applicability of TS 3.0.3 needs to be done due to the potential inoperability of all equipment needed to maintain control room habitability per GDC-19.

Compensatory actions to close any open leakage paths in the event of a LOCA ensures the probability of occurrence of an accident is not increased and the possibility of a different type accident is not created. As long as all leakage paths are isolated within the 25 minutes there is no increase in consequences. The activity will not affect the operability of the remaining HHSI pumps. Therefore, the activity does not create an unreviewed safety question.

00-SE-PROC-15

Description

0-GOP-22.1, Operations Support for Removing Sludge from the Auxiliary Building Sump
Create a new procedure to install a flange with a cam-lok or equivalent coupling replacing the bonnet of 1-DA-28 isolation valve in the discharge line of 1-DA-P-3B. A temporary hose and pump will then be used to remove water from the Auxiliary Building Sump, reducing the level below the installed pumps, and pumping the water to its normal destination, the High Level Liquid Waste Tanks, via the fitting into 1-DA-28.

Summary

It is desired to reduce the amount of water in the Auxiliary Building Sump before the sludge in the bottom of the sump is removed and prepared for shipment offsite. Reduction of the water before sludge removal is desired rather than afterward, to minimize radwaste.

Therefore, a flange with a cam-lok or equivalent coupling will be installed in place of the bonnet of 1-DA-28 (isolation valve in the discharge line of 1-DA-P-3B). A temporary hose and pump (powered from either Service Air or a non-Class 1E power supply) will then be used to remove water from the Auxiliary Building Sump, and pumping the water to its normal destination, the High Level Liquid Waste Tanks, as described in UFSAR Section 9.3.3, via the fitting into 1-DA-28. Procedure 0-GOP-22.1 requires that the temporary pump discharge valve be closed when the temporary pumps are secured to prevent short cycling through the temporary pump in the event an Auxiliary Building Sump Pump starts.

The fitting connection will be functionally tested by the initial flow through the temporary flow path. If there is leakage from the connection or the temporary flow path, that is not directed back into the sump, the temporary pump will be secured to minimize potential contamination until the condition can be corrected. 1-DA-28 will also be leak checked after it has been reassembled following removal of the temporary flange and connection. The fittings/components used to will be required to have a minimum pressure rating of 125 psig. This is 5 times more than the shut off head of the Aux. Building Sump Pumps and equivalent to the shut off head of the temporary pump. If the Auxiliary Building Sump Pumps were to start, the pressure rating of the temporary components will not be exceeded. If the temporary pump was inadvertently operated with a discharge valve closed, the pressure rating of the temporary components wouldn't be exceeded.

The Auxiliary Building Sump Pumps and the associated DA system are not safety related, nor are they required by Tech Specs or described in the Tech Spec Bases.

All existing instrumentation, controls, and flow paths will be unaffected after installation of the temporary modification. The existing sump pumps 1-DA-P-3A and 3B will still be in service ready to pump the sump in auto if sump inleakage should increase above the ability of the temporary pump. The flange connected to the bonnet of 1-DA-28 does not affect the flow path of 1-DA-P-3B. The valve is a Grinnell valve. The diaphragm "internal" will come off with the bonnet, thus ensuring the open flow path for 1-DA-P-3B if it starts. The discharge check valves for 1-DA-P-3A and 3B will prevent flow from the temporary pump from short cycling back into the sump.

Just as in other dewatering activities, it is not expected that levels of radiation or airborne radioactivity will increase. The activity does take place in the RCA and the usual radiological controls and practices will be in place.

Dewatering activities in the Auxiliary Building neither increase the probability of, nor increase the consequences of, nor create the possibility of accidents different than, any Chapter 15 accidents. No new flow path to the environment is created nor is any existing flow path adversely impacted by this activity. Therefore, an unreviewed safety question does not exist.

00-SE-PROC-16

Description
1-MCM-1910-01 Installation and Removal Of Temporary Drinking Water Fountains in Radiologically Controlled Areas
This Safety Evaluation addresses the installation of a temporary Domestic Water line from 1-DW-216 located in the Decontamination Building for a water fountain in the Fuel Building.

Summary
A temporary Domestic Water line is being installed in the Decontamination Building to supply drinking water to a fountain on the top floor of the Fuel Building for personnel doing fuel handling activities.

Domestic Water is being tapped off of an existing line at 1-DW-216 at elevation 271 in the Decontamination Building. The line will be routed through the Fuel Building basement to the top floor of the Fuel Building. The line will be routed in the overhead areas when possible. The fountain dispenser will have a foot pedal to allow workers to obtain potable water without using their potentially contaminated hands. The fountain will facilitate consumption of domestic water by personnel working in a high heat area without the need to exit the work area. Paper drinking cups will not be provided.

All components will be selected to convey potable water below 200 degrees F, at normal operating pressures of the Domestic Water system, which range between 46 and 58 pig. The line will be routed in overhead areas that do not interfere with any operable safety related equipment. Seismic restraint and distance requirements (ref. VPAP-0312) will be satisfied with the installation of the temporary line and the fountain. Any leakage from the line will be directed to the Fuel Building sump. Operators would be alerted to any leakage in excess of the sump pump capacity by a sump high level alarm. In addition, excessive DW leakage would bring in the domestic water tank low level alarm.

For personnel safety, the drinking fountain will be tested prior to use/human consumption to ensure potable water is achieved.

This change will not affect the operability of the remaining DW system. The change does not alter the equipment or operation of the power station and its safety systems. The ability of all systems to perform as designed is not impacted. Therefore, the probability of occurrence of analyzed accidents and malfunctions has not been increased. Likewise, the consequences of any analyzed accidents and malfunctions have not been changed. There is no potential for the creation of a new or different accident or malfunction. Additionally, the Technical Specifications and Operating License are not affected. Therefore, this change does not constitute an unreviewed safety question and should be allowed.

00-SE-PROC-17

Description

1-EPM-B-1817-01, Functional Testing of Interlocks from Control Circuits for Breaker 15H6, Charging Pump 1-CH-P-1A, 1-EPM-B-1817-02, Functional Testing of Interlocks from Control Circuits for Breaker 15J6, Charging Pump 1-CH-P-1B, 2-EPM-B-1817-01, Functional Testing of Interlocks from Control Circuits for Breaker 25H6, Charging Pump 2-CH-P-1A, 2-EPM-B-1817-02, Functional Testing of Interlocks from Control Circuits for Breaker 25J6, Charging Pump 2-CH-P-1B

This procedurally controlled temporary modification will allow bypassing the contact on the interlock portion of the control circuit for the respective pump that is being tested so that the contacts on the opposite bus' charging pump will remain in the configuration to receive an auto start signal if no charging pumps are running on that particular unit.

Summary

This procedurally controlled temporary modification will allow bypassing the contact on the interlock portion of the control circuit for the respective pump that is being tested so that the contacts on the opposite bus' charging pump will remain in the configuration to receive an auto start signal if no charging pumps are running on that particular unit.

If, during the Functional Testing of Interlocks from Control Circuits for Breakers for the "A" or "B" charging pumps, the "C" charging pump is running and subsequently trips during the testing of the interlocks for the "A" or "B" pump without the jumper in place, the charging pump with auto-start capability on the same bus which is being tested will not be available to run. The pump on the opposite bus may not auto start during the test if required under all conditions.

Without the jumper installed, the assurance of an automatic start due to breaker logic is not guaranteed. However, an automatic start will occur due to low header pressure. This start would not be immediate since there is a finite amount of time associated with the pressure decay of the charging header.

This procedurally controlled temporary modification will only be implemented if the "C" charging pump is in service and the opposite bus' charging pump is operable with auto-start capability.

This jumper will allow the charging pump to auto-start on the no pumps running interlock. Therefore, neither the probability nor the consequences of an accident are increased by use of this procedurally controlled TM.

No component or system as described in the SAR will be altered. The operable charging pump(s) are still available for accident conditions with respect to the LCO of Technical Specification 3.5.2. No accidents of a different type are created. The use of this procedurally controlled temporary modification will not challenge any concepts discussed in the Technical Specification basis, and there will be no violation of any LCO's by the implementation of this jumper. The TM will not reduce the margin of safety of any part of the Technical Specifications.

For these reasons an unreviewed safety question does not exist.

00-SE-PROC-18

Description
2-PT-61.3 Revision 26-P1-OTO1
1) Install and Type C test a blind flange (pancake) in piping inside containment isolating containment penetration 111D.
2) Determine as-found leakage of 2-DA-TV-203A.
3) Move the containment isolation boundary to the blind flange inside containment.
4) Repair 2-DA-TV-203A.
5) Remove blind flange from penetration piping.

Summary
This Safety Evaluation addresses a Procedurally Controlled Temporary Modification (Reference: VPAP-1403, Section 6.3) to control the installation and testing of a blind flange in the Unit 2 High Radiation Sample System line to the Unit 2 Containment sump. This is being done to allow maintenance to be performed on containment trip valve 2-DA-TV-203A. This valve experienced a mid position indication the last time it was stroked and external repairs were unsuccessful.

Both trip valves, 2-DA-TV-203A and 203B, for this Type C boundary are located physically outside of containment with the "A" valve being closest to the containment. The "B" valve is currently tagged closed to comply with Technical Specification actions. The proposed repair will add a blind flange in the non-safety related piping inside containment which is connected to this penetration. The 1-hour action condition for Technical Specification 3.6.1.1 Containment Integrity will be entered during testing of the blind flanged piping. Under the action condition for Technical Specification 3.6.3.1, the containment penetration isolation boundary will be moved from the "B" containment trip valve to the blind flange. Only one isolation barrier, i.e. the blind flanged piping, is required to maintain containment isolation under Technical Specification 3.6.3.1 actions.

Based on Technical Specification 3.6.3.1 actions for an inoperable containment isolation valve and a review of the design basis of the piping connected to penetration 111D, the non-safety related blind flanged piping together with the safety-related piping connected to check valve 2-DA-48 is adequate for use as the containment boundary during the on-line repair of 2-DA-TV-203A. During an accident, the configuration provides two independent automatic passive barriers between the inside and outside containment atmospheres. However, check valve 2-DA-48 is not Type C leakage tested. The piping is seismically designed, protected from missiles and high energy line breaks. The piping will be verified leak tight during Type C leakage testing. The blind flange will be acceptably Type C leakage tested prior to declaring it the containment pressure boundary.

Justification:

This change should be allowed as it will be in compliance with the Technical Specifications, the Safety Analysis Report, and the design basis for the affected systems. The SAR does not prohibit the use of a blind flange for this purpose. This evolution will allow maintenance to be completed on the associated trip valve and will allow the system to be returned to its original configuration. Type C testing will be performed on the blind flange following installation to ensure Containment Isolation is met to allow maintenance on 2-DA-TV-203A.

Unreviewed Safety Question Assessment:

This activity will install a blind flange in piping connected to containment penetration 111D to allow maintenance to be completed on the penetration's trip valve. It will be implemented with approved station procedures, which are bounded by existing analyses. The activity does not increase the probability of any analyzed accidents, does not create a new or different type of accident, and does not reduce the margin of safety as defined in the Technical Specifications bases.

Therefore, this activity should be allowed.

00-SE-PROC-19

Description
1-OP-5.7, Operation of the Pressurizer Relief Tank (PRT)
Install a temporary hose between a SI accumulator vent and a drain off of the RHR relief valve discharge line.

Summary
It is desired to provide an additional controlled source of nitrogen to the PRT to provide a slight overpressure to the RCS as part of the normal RCS draindown from 28% to 74 inches. The proposed procedure change will use a hose rated for at least 100 psig to supply nitrogen from a SI accumulator vent to the RHR relief valves discharge line to the PRT. This will allow the control room operator to control RCS overpressure by opening the pressurizer PORVs and controlling the makeup flow of nitrogen to the SI accumulator with its supply HCV.

Personnel safety will be maintained by maintaining the nitrogen supply pressure from accumulator at approximately 50 psi, and the hose will be physically restrained at the connections. In addition, a check valve will be provided on the jumper discharge side. This will prevent the hose from whipping and limit the amount of radioactive gas that could be released from the PRT if the jumper hose were to be cut.

Equipment safety is provided by one PZR PORV blocked open and the PRT rupture disc. The nitrogen pressure to the RCS will be limited to less than or equal to 50 psig. This pressure will provide a back pressure to the RHR relief valves. However, as discussed above, other RCS/RHR overpressure protection is in place.

An unreviewed safety question is not created because:
(1) The probability of an accident or malfunction previously evaluated in the SAR occurring is not increased. The change does not introduce any accident initiators. The unit is shutdown and will be in Mode 5 while this change is active.
(2) The consequences of any accident or malfunction previously evaluated in the SAR are not increased. No fission product barriers are compromised by this change. The unit is shutdown and will be in Mode 5 while this change is active.
(3) The possibility of creating a new accident or malfunction has not increased. The change will be installed by qualified personnel and using appropriate safety guidelines. The control room operator will have control of the nitrogen supply via the SI accumulator makeup HCV. A jumper hose rupture does not breach the RCS boundary because of the installed check valve.

Because the change is not an undue risk to personnel safety or reactor safety, and because the change will help ensure that the outage will not be unduly delayed, (remember that a shorter outage is a safer outage as long as it is properly planned), this procedure change should be allowed.

00-SE-PROC-20

Description
New Procedure to 1-MOP-49.41 to allow chemical cleaning of 1-CC-E-1B
Chemical Cleaning of 1-CC-E-1B

Summary
The 3000 Unit 1 CCHX tubes are made of seamless, type 316 SS and have been installed for a little over 3 years since the Unit 1 CCHXs were re-tubed in 1997. The Unit 1 CCHX tubes are more susceptible to Manganese or MIC pitting than the Unit 2 CCHX tubes, which are constructed of Titanium. 1-CC-E-1A was mechanically cleaned earlier this summer. Visual inspection of 1-CC-E-1A, utilizing a boroscope, showed that mechanical cleaning of the heat exchanger was not highly effective in removal of a hard scale, which has previously been tested to be high in Manganese. System Engineering, working with Component Engineering and Chemistry, developed a plan that would determine whether or not a significant company asset was at risk to Manganese and/or MIC pitting.

The action plan consists of mechanically cleaning the tubes utilizing brushes, chemically cleaning the tubes to remove the Manganese, and then eddy current testing 1000 tubes to determine if the tubes are pitted. 1-CC-E-1B was selected as the heat exchanger to perform the testing on since it had not been cleaned yet this year.

Chemical cleaning would consist of adding sodium meta-bisulfite as the reducing agent to place Manganese in solution, EDTA as a chelating agent to keep it in solution, and sulfuric acid until the pH of the 4000 gallon solution was 4.0 (increasing the effectiveness of the chemical bath). Once the solution had recirculated for at least 3 hours, the solution would be neutralized with addition of soda ash and pumped to the SW discharge header.

Since the quantities of chemicals are small and the chemicals are not volatile, this activity does not result in a chemical release adverse to any equipment or personnel. However, as an enhancement, procedural guidance is provided to prevent lining up the Auxiliary Building ventilation to the Filters. Since the process vents are always lined up to the Auxiliary Building, and can't be isolated, the procedural enhancement can not be performed. This is satisfactory since the low volatility ensures that any vapor from the chemical cleaning would be harmless to charcoal filters.

This evaluation on chemical cleaning 1-CC-E-1B has determined that there is no Unreviewed Safety Question (USQ) since the probability and consequences for any accident previously evaluated in the SAR remains unchanged. Additionally, no other accident is credible as a result of chemically cleaning the CCHX since the chemicals used are not overly aggressive, have been tested on materials that they will come in contact with, and have a short contact time. Finally, the probability of occurrence or consequences of the malfunctions described in the SAR of either a loss of a CCHX tube or loss of control room habitability due to charcoal filter ingestion of chemical vapors is not increased by this activity. It should be noted As such, this activity can be safely performed with no adverse effect on nuclear safety.

Although this evaluation was performed for 1-CC-E-1B, it would be applicable to a procedure for 1-CC-E-1A, as well. Additionally, procedures for the Unit 2 CCHXs could be written once the heat exchanger and tube materials were qualified via an engineering evaluation.

00-SE-PROC-21

Description

1/2-MOP-5.97 Rev. 0, Returning One or More Reactor Coolant Loops to Service Following Maintenance Using Backfill Method.

This new procedure allows isolated and drained reactor coolant loops to be returned to service using by backfilling through the loop stops from the active portion of the RCS. Installation of temporary modifications to bypass the loop stop valve interlocks is included in this procedure.

Summary

Technical Specification Amendments 223 (Unit 1) and 204 (Unit 2) have been received to allow a vacuum assisted backfill technique when returning an isolated and drained reactor coolant loop to service. 1/2-MOP-5.97 Rev. 0 will be the procedures controlling this evolution. The procedures provide the necessary controls for temperature and boron concentration of the isolated loop to ensure the required shutdown margin is maintained.

Specifically, the procedures ensure the following conditions are maintained: a) Seal injection may be supplied to the RCP if the loop has been verified drained (using PDTT inleakage rate), and the boron concentration of the seal injection water is above the TS 3.9.1 or 3.1.1.2 requirements for the applicable mode. b) After defeating the loop stop valve interlocks via jumper installation, the applicable cold leg loop stop valve may be opened provided that the loop is drained, the pressurizer contains at least 450 cubic feet of water (32% cold cal level), and a source range neutron flux monitor is operable. c) Backfilling of the loop may proceed if the pressurizer level is maintained above 32 % cold cal level, the source range neutron flux count rate is no more than a factor of 2 above the initial count rate, and seal injection is maintained above the required boron concentration. d) When the isolated loop is full, the loop stop valves can be fully opened when the boron concentration of the loop is in spec and no more than two hours have passed since the loop was backfilled. This backfill technique was previously evaluated under 99-SE-OT-32, and these required conditions are properly controlled by the proposed procedures 1/2-MOP-5.97 Rev. 0. This evaluation concentrates on the temporary modifications that will be required to defeat the loop stop valve interlocks.

RCS Loop Stop Valve interlocks are designed to ensure that an accidental startup of an undrained, unborated and/or cold, isolated reactor coolant loop results only in a relatively slow reactivity insertion rate. The interlocks perform a protective function using two independent limit switches to verify that the hot leg loop stop valve is open, two independent limit switches to verify that the cold leg loop stop valve is full closed, and two independent flow switches to verify that bypass flow around the cold leg loop stop valve is greater than 125 gpm for 90 minutes. (The flow verifies that the pump is running, the bypass line is not blocked, and the valves in the bypass line are open). Additionally, the hot leg loop stop valve is prevented from opening unless the cold leg valve in the same loop is fully closed.

00-SE-PROC-22

Description
0-MOP-50.2 Rev. 1 BC Cooling Tower Basin
An additional attachment is being added to this procedure for controlling a temporary modification which is composed of a mechanical jumper between the Fire Protection System and the Bearing Cooling Water Tower Basin. Two 2 1/2 inch hoses will be installed from Hose House "L" to the tower basin to allow refilling of the basin.

Summary
0-MOP-50.2 is an approved Maintenance Operating Procedure for the BC Cooling Tower Basin. It is being revised to include steps to refill the basin using either a backfill method or the fire protection system. The use of the fire protection system for makeup to the BC Tower Basin was previously approved for use in Operating Procedure 1-OP-50.2 using one 1 1/2 inch hose. The main difference between this safety evaluation and the safety evaluation performed for 1-OP-50.2 will be the volume of water being removed from the FP system and whether this jeopardizes the FP system integrity.

An additional attachment is being added to 0-MOP-50.2 for controlling a temporary modification which is composed of a mechanical jumper between the Fire Protection System and the Bearing Cooling Water Tower basin. Two 2 1/2 inch hoses will be installed from Hose House "L" to the tower basin to allow refilling of the basin from the fire protection system following maintenance on the tower.

The safety classification of the Bearing Cooling Water (BC) system is non safety related and the Fire Protection (FP) system NSQ. The FP system does have regulatory requirements associated with Technical Requirements Manual (TRM) Section 7.1.8, Fire Suppression System Impairments.

Removing water from the FP system via two 2 1/2 inch hoses will not jeopardize the FP system design basis. The volume of water supplied to BC will reduce the water volume available to mitigate a fire; however, sufficient capacity exists to ensure the FP system performs its design function. In addition, the procedure provides compensatory actions to be taken if the Station Fire Alarm sounds. If the fire alarm sounds the procedure requires the Shift Supervisor to be contacted to determine if isolation of this jumper is required. One fire pump, preferably 1-FP-P-1, will be started and run for the duration of the makeup. In the event that a fire should occur during this time, the remainder of the FP system including the diesel driven fire pump will be available and the makeup can be terminated as required. In the event of a rupture in the hose, FP system capacity/pressure is maintained by a pressure maintenance system consisting of a jockey pump, a hydropneumatic tank with an air compressor and accessories. In addition if system pressure drops below preset values (90 psig for the motor driven fire pump and 52 psig for the diesel driven fire pump) then the fire pumps will automatically start. The rupture would eventually be discovered and could easily be isolated by closing valves 1-FP-236, 1-FP-237 and/or 1-FP-238. The fire protection system will remain operable.

The Supervisor of Safety And Loss Prevention has been contacted concerning this proposed activity. He has stated that using Hose House L as a makeup source of water to BC will not render that hose house inoperable, provided the hoses normally present in the hose house are not used. 0-MOP-50.2 specifically requires that hoses not designated as fire fighting equipment be used for this activity.

The status controls of a Procedurally Controlled Temporary Modification ensure that the hose house status is documented and communicated with the on-duty shift. If the TM is installed for more than one shift, the TM status is required to be documented in the Temporary Modification Log and in the Equipment Status System. Prior to assuming the watch, the Scene Leader reviews all Abnormal Status entries that affect Fire Protection. In addition, the remainder of the shift team is required to review Abnormal Status entries that might affect their watch station, this includes the Shift Supervisor and Unit Supervisors.

During this evolution, the BC and FP systems will be operated within their respective design capabilities. Performance of this procedurally-controlled evolution will neither increase the probability that a previously analyzed accident or malfunction will occur, nor will it increase the possibility that a unique accident or malfunction will happen. The FP System, as described in the TRM, is not adversely affected.

An Unreviewed Safety Question is therefore not created by this Temporary Modification.

00-SE-PROC-23

Description
1-OP-53.1, Rev. 26, OTO1 (Emergency Diesel Generator Fuel Oil Storage and Transfer System)
A section is added to the procedure to pump 1-EG-TK-2B ("B" Underground Fuel Oil Storage Tank) to barrels to remove debris clogging the tank fill connection piping. This will be done by installing a pump at the truck fill connection and pumping to a barrel to draw out debris from the tank fill piping.

Summary
It is desired to add a section to 1-OP-53.1 to remove debris that is clogging the 1-EG-TK-2B ("B" Underground Fuel Oil Storage Tank) fill piping. The new section of the procedure directs the installation of a temporary hose and pump at the truck fill hose connection for 1-EG-TK-2B in order to draw any sediment or debris that is clogged in the fill piping to a barrel. Installation of the hose is considered a procedurally controlled Temporary Modification (TM).

UFSAR Section 9.5.4 provides a description of the EDG Fuel Oil Storage and Transfer System. The section includes a description of the Underground Fuel Oil Storage Tanks (UGFOSTs) and a description of the truck fill lines. The lines are emergency, Seismic Category I, tornado, missile, and flood-protected truck fill line connections, and are used to fill the UGFOSTs in the event that the non-seismic AGFOST is not available. The fill line for 1-EG-TK-2B consists of a strainer, isolation valve and a 3 inch truck fill connection. The line is connected to line 2"-FOF-151-S, downstream of the 2B UGFOST inlet check valve, 1-FO-218 and inlet isolation valve, 1-FO-217. The new section of the procedure will provide the guidance to isolate the tank inlet piping by closing 1-FO-217, remove the strainer internals on the truck fill line, and install a hose with a pump to the truck fill connection. The discharge of the pump will be directed to a 55 gallon barrel. The fill line isolation valve will then be opened and the pump started in attempt to draw debris or sediment through the affected section of tank fill line to the 55 gallon barrel. The pump will be stopped if any condition is met:

(1) A vacuum is being drawn on 1-EG-TK-2B or the associated inlet piping as indicated on the pump suction gauge, or
(2) 1-EG-TK-2B tank level of 9.5 feet is reached, or
(3) Debris has been thoroughly removed from fill piping.

The UFSAR states that there is no special provision made in the design of the fuel oil storage fill system to minimize the creation of turbulence of the sediment in the bottom of the storage tank. However, it is not acceptable to clear any sediment or debris in the fill line by blowing into the tank. Therefore this procedure is suitable for removing the debris in the affected portion of the fill line. Use of the truck fill line without the strainer installed is acceptable since fuel oil will be pumped from the tank and not into it. Procedural guidance is also provided for the maximum amount that can be drained to ensure the tank remains above the Tech. Spec. minimum level.

The 55 gallon barrel(s) will be in the Fuel Oil Pump House. The barrel(s) will be located so as to prevent spilled oil from reaching navigable waters. The guidance of VPAP-2203 will be used to store and transfer or dispose of the pumped fuel oil. If a spill were to occur, it would be limited to inside the Fuel Oil Pump House in the vicinity of the evolution and the pump would be promptly stopped to contain the spill inside the Pump House, and the actions of 0-AP-23 would be taken. It is judged that the evolution has no worse consequences than the UFSAR described method of fill from a Fuel Oil Truck.

The pumping evolution will have no adverse impact on the UGFOSTs. The ability to transfer oil from the UGFOST to the EDG Day Tanks will not be adversely affected. In addition, the ability to transfer fuel oil to the SBO Diesel Day Tank will be maintained. The transfer hose connections are located at an elevation above the UGFOST, so any break or opening in the hose or connections will not cause draining of the UGFOST. No T.S. LCOs will be challenged by the evolution. In addition, the License Conditions related to the EDG Fuel Oil Transfer System will not be impacted.

An Unreviewed Safety Question is not created by the new fill method for the following reasons. The evolution is limited to the Fuel Oil Transfer System. It does not impact any other systems or components, so it does not create the means to change or introduce any existing or new accident precursors. It will not cross-tie any safety related portions of the system. It will not adversely affect the ability to supply fuel oil to the EDG or SBO Diesel Day Tanks, nor affect the design function and performance of the emergency power systems. Therefore, the evolution will not increase the consequences of any accident. The evolution involves components that would normally be used during tank filling methods, therefore, no new related equipment malfunctions are created. The easy access to the temporary pump connected to the truck fill connection for termination of the evolution provide assurance that the consequences of a spill are not increased. Since the T.S. LCOs and License Conditions related to the system will be not be adversely affected by the evolution, the margin of safety provided by those requirements will not be reduced.

00-SE-PROC-24

Description

2-OP-50.2, Rev. 11-OTO1, "Operation of the Bearing Cooling Water System"
A jumper will be installed to bypass the low temperature interlock associated with the Bearing Cooling (BC) return valves to the BC Tower, 1-BC-MOV-126 and 2-BC-MOV-226. This TM will ensure that these valves will open even if the Tower outlet temperature is below 70 degrees. The jumper will only be installed on the Unit 2 temperature element (TE), however the jumper will serve to open the Unit 1 valve because the temperature for the interlock for the Unit 1 valve can come from either units TE.

Summary

The purpose of this Temporary Modification (TM) is to ensure the Bearing Cooling (BC) return valve, 1-BC-MOV-126 and 2-BC-MOV-226, to the BC Tower can be opened with the Tower bypass valves, 1-BC-MOV-127 and 2-BC-MOV-227, closed AND with a tower outlet temperature below 70 degrees. The Bearing Cooling return valves are interlocked to open only if either the Tower bypass valve is FULLY open OR the BC Tower outlet temperature below 70 degrees. It is not desired to open the Tower bypass valves since there is corrosion in the downstream piping and it is not desired to introduce any foreign material into the BC Tower. This condition may have an adverse impact on the flowpath of the Bearing Cooling water to the pumps.

The Bearing Cooling system is currently operating in the Lake-to-Lake cooling mode while the Bearing Cooling Tower is being cleaned. The Tower bypass piping (Unit 1 and Unit 2) could not be cleaned and an inspection identified corrosion and foreign material downstream of the Tower bypass valves, 1-BC-MOV-127 and 2-BC-MOV-227. Due to the system load, the bypass lines are not used nor is it desired to open them at any point during operation.

It is desired to switch from the "Lake-to-Lake" mode to the "Tower-to-Tower" mode of operation upon completion of the BC Tower cleaning. The Bearing Cooling return valves to the Tower will only open when the following conditions are met: (1) Master Control Switch (for the respective units Bearing Cooling Control Panel) is in the TOWER position, and (2) The Tower bypass valve is FULLY open OR Tower outlet temp. is greater than 70 degrees.

Without the jumper the BC Tower return valves may not open with the Tower bypass valves remaining closed, and since temperature element 2-BC-TE-220 may be reading less than 70 degrees because there is currently no flow through this portion of the system in the "Lake-to-Lake" mode and the temperature of the water is close to ambient.

The "One-Time-Only" procedure revision will lift a lead at 2-BC-TE-220 so that the temperature condition will be met to open 2-BC-MOV-226 while maintaining the Tower bypass valves closed. The Bearing Cooling Tower outlet temperature as read on the Unit 2 Bearing Cooling Control Panel will be reading failed high with the lead pulled. Once Bearing Cooling flow is initiated in the Tower-to-Tower mode from the Unit 2 Bearing Cooling pumps, the Unit 1 Bearing Cooling system will be switched to Tower-to-Tower mode.

This TM also satisfies the requirements for the Unit 1 Bearing Cooling water return valve to the Tower, 1-BC-MOV-126, to be opened with Tower outlet temperature less than 70 degrees (jumpering either Unit 1's or Unit 2's temp. element will defeat the low temperature interlock for both MOVs). Tower outlet temperature will be indicated (from 1-BC-TE-120) on the Unit 1 Bearing Cooling Control Panel once flow is initiated on the Unit 1 Bearing Cooling pump suction header.

The temperature requirements for the interlock is in place so that there are no icing concerns associated with the BC Tower. The Bearing Cooling Tower will be visually inspected for icing before switching from the "Lake-to-Lake" mode to the "Tower-to-Tower" mode.

Bearing Cooling is not a Tech. Spec. system. The Bearing Cooling system is described in section 10.4.7 of the UFSAR. The UFSAR describes operation of the Bearing Cooling system in either the "Tower-to-

Tower" or "Lake-to-Lake" operation. Currently we are running in the "Lake-to-Lake" configuration. This TM will be removed when the Bearing Cooling system is returned to the "Tower-to-Tower" mode. Operation will continue to conform to UFSAR requirements. There is no adverse environmental impact as a result of this TM.

The overall function and operation of the Bearing Cooling system will remain unchanged as a result of this activity. The Bearing Cooling System is non-safety related and is not required for obtaining or maintaining safe shutdown. Therefore, this TM does not create an unreviewed safety question.

00-SE-PROC-25

Description
0-OP-22.17, Rev. 2
Blowdown of the SW Reservoir using the installed blowdown piping from the SW to LW Systems. The SW System blowdown flow will be directed to either the Discharge Tunnel or the Clarifier for release to the Waste Heat Treatment Facility.

Summary
It is desired to utilize the originally installed SW to LW line to blowdown the SW Reservoir in a controlled manner to improve the quality of the water in the SW System. The SW System blowdown flow will be directed to either the Discharge Tunnel or the Clarifier for release to the Waste Heat Treatment Facility. Existing lines and valves were provided in the original design of the systems to support this type of blowdown evolution. However, evaluation of this evolution is required due to concerns regarding the impact of the blowdown on SW Reservoir inventory requirements and the non-seismic condition of the cross-tie line from the SW to LW Systems.

The SW Reservoir and Lake Anna are provided by design to function as the Ultimate Heat Sinks for operation of the station. As described in UFSAR Section 9.2.1 and 9.2.5, the SW Reservoir is designed to provide a 30-day cooling water supply to its supplied safety-related equipment, without inventory make-up, during normal and accident conditions. T.S. 3.7.5.1 provides the minimum level (313 ft. elevation) and maximum temperature (average water temperature at SW Pump discharge of no greater than 95 degrees F) requirements to support the SW Reservoir design function. There is no documented evidence that SW System blowdown flow was included as an assumed inventory loss for the calculation of the required minimum reservoir level to support the 30-day cooling water inventory (with no make-up) design requirement.

The SW system piping up to and including the SW blowdown line isolation valve (1-SW-15) is seismically qualified. LW piping downstream of the SW isolation valve (1-SW-15) is not seismically qualified. The isolation valves and a significant portion of the blowdown piping are located in the Aux. Building basement. For evaluation purposes, failure of the non-seismic portion of the blowdown line must be assumed during a seismic event. If in service, failure of the line would cause leakage into the Aux. Building from the SW System. Though the Auxiliary Building is provided with a sump and sump pumps, excessive leakage could cause a flood concern. UFSAR Section 3C.5.4.6.3 provides a discussion of flooding effects within the Aux. Building. The section states that the minimum height above floor level of equipment essential to safety is 15 inches. It further states that the water volume required to reach that height is approximately 132,000 gallons.

For the purposes of this evaluation, 3 different ranges of blowdown flow were considered; 0 - 3 GPM, 3 - 71 GPM and > 71 GPM. The range of the flow indication for the blowdown line (1-LW-FI-124) is 0-400 gpm. The general procedural limitation on clarifier discharge flow is 125-300 gpm. It can be reasonably assumed that actual flow will be controlled at less than 400 gpm.

To preserve the assumed SW Reservoir inventory for its design function during accident conditions, the SW blowdown flow must be terminated near the beginning of the accident. The SW blowdown isolation valve (1-SW-15) is located in the Aux. Building basement. During the Design Basis Accident (Large Break LOCA), the dose rates are acceptable for human occupancy for this area until transfer of the ECCS to Cold Leg Recirculation. Afterward, the estimated dose rates for the area are in excess of allowable values (dose rate values are based on conservative calculations performed for equipment EQ qualification purposes). The minimum time to reach ECCS transfer to Cold Leg Recirculation has previously been calculated to be no less than 29 minutes. Therefore, it will be required to isolate the SW blowdown line within 29 minutes of the onset of accident conditions. The effect of the loss of inventory from the SW System during this period is considered to be negligible. It will constitute less than 0.1% of the total assumed volume of the reservoir at the minimum T.S. level of 313 ft. elevation. In addition, it is not expected to challenge SW system performance or pump run-out.

To alleviate concern of Aux. Building flooding, the SW blowdown flow must be terminated shortly after a seismic event. The pipe size of the blowdown line at the SW isolation valve (1-SW-15) is 3 inch diameter. A conservative calculation of break flow from this size line using Bernoulli's Equation (assuming no head loss) with a delta P of 100 psi across the break, renders a value of approximately 1900 gpm. With this flow rate, the critical Aux. Building flood level would not be reached until 69.5 minutes (assuming no operation of the Aux. Building Sump Pumps). Actual flow is expected to be less. Therefore, isolation of the line within 1 hour following a seismic event would reasonably ensure flood protection of equipment important to safety. Isolation of the line within 29 minutes of the onset of accident conditions, described above, would eliminate any flood concerns from the line due to a subsequent seismic event.

To permit use of the SW blowdown line with flow greater than 3 gpm, manual action will be required to locally isolate the line using 1-SW-15. Isolation must occur within 29 minutes of the onset of accident conditions for either unit. Isolation must occur following a seismic event, if the line was not already isolated following the onset of an accident. These limitations will be provided in 0-OP-22.17, Rev. 2.

Use of manual action as a compensatory action for the use of the SW blowdown line is judged to be acceptable. The action is a specific non-complicated task limited to one component and location. The action does not involve or require any advanced diagnostic tasks or skills. The valve involved can be readily accessed and closed well within the required time limits (including notification and travel time from most areas of the Protected Area). Failure to satisfactorily complete the task is considered remote.

The procedure revision does not increase the probability of occurrence nor the consequences of an accident or malfunction previously addressed in the UFSAR and no new accident or malfunction is created. By implementing this procedure to blowdown SW, neither the probability nor the consequences of an accident are increased by having a small amount of SW flow through the LW system. All accidents remain bounded by the accident analyses.

Implementation of these procedure changes has no effect on any existing accident precursors nor any precursors for equipment failure. Therefore, implementation of these procedure changes will not increase the probability of occurrence of previously analyzed accidents or malfunctions of equipment.

This modification will not challenge any concepts discussed in the Technical Specification basis, and there will be no violation of any LCO's by the implementation of this procedure. The diverting of a small amount of SW will not reduce the margin of safety of any part of the Technical Specifications.

For these reasons an unreviewed safety question does not exist.

00-SE-PROC-26

Description

Abnormal Procedures 1/2-AP-19

This change adds an attachment to 1/2-AP-19 that jumpers out the circuitry for the fire protection trip and the vibration switch trip of the Bearing Cooling (BC) fans.

Summary

A procedurally controlled temporary modification (TM) is being included in an Attachment to Abnormal Procedures 1/2-AP-19 which defeats the Fire Protection System and vibration trips of the running Bearing Cooling fans. This is being done to maintain the Bearing Cooling fans and Bearing Cooling if there is a trip due to a malfunction within the circuitry. The procedurally controlled TM is the same for both units.

The Bearing Cooling, affected Fire Protection system, and vibration detection equipment are not safety related nor are they required by Technical Specifications. Installation of the jumper does not affect any safety related systems and would not affect the ability of any safety related systems to perform its intended functions.

There will be a once per shift fire watch while this jumper is installed to provide fire detection capability. Upon notification, actions will be taken to secure BC fans, dispatch the Fire Brigade, and initiate Tower Cell deluge.

In addition, the vibration trip circuit provides protection to each individual BC fan in the event of elevated vibrations. Predictive Analysis will monitor each of the fans for increasing vibrations to detect conditions that could adversely affect the performance of the fans.

The probability of a fire is not increased nor is the probability of damage to the BC fans. Temporarily defeating the protection circuitry while maintaining BC fan operation has no ability to influence the mechanisms by which a fire or fault is generated. Compensatory actions established will adequately provide the same functions.

The overall operational consequences of a loss of Bearing Cooling from a tower/fan problem remains the same - shutdown of the secondary side of the plant.

This procedurally controlled TM does not create the means to cause a new or different malfunction of the BC system or surrounding equipment.

The margin of safety for the station as described in the Technical Specifications Bases is not altered since the BC system and its components are not described in the Technical Specifications. Failure to install this jumper if required could cause damage to plant equipment and potentially require the plant be ramped down in power or off-line due to a loss of Bearing Cooling.

Based on the above discussion, there is no Unreviewed Safety Question involved with installing this procedurally controlled TM.

00-SE-PROC-27

Description

NSS implementing Procedure 1-WP-G99175

NSS implementing Procedure 2-WP-G99176

Design Change 99-175 and 99-176 replace the Containment Air Recirculation air-operated dampers with backdraft type dampers. During implementation when a damper is removed for replacement the discharge duct of the applicable containment air recirculation fan will be covered to prevent flow through the opening.

Summary

The Containment Air Recirculation System cools the containment during normal operation and provides an air sample source for Containment Gaseous & Particulate Monitoring. The CARF also provides environment cooling for personnel accommodation during shut down and ability for air sampling during mode 6. Design Change 99-175 and 99-176 replace the CARF discharge air operated dampers with backdraft type dampers. The dampers prevent air back flow through a non-operating containment recirculation fan. During implementation with the non-affected fan(s) running, removal of a damper for replacement will allow air to flow through the open duct. To preclude this air flow, covering will be placed across the fan discharge opening in the ring duct for the damper(s) being worked. Covering the duct opening performs the same function as the CARF damper. The remainder of the containment air recirculation system will be maintained available.

Per Technical Specification

UNREVIEWED SAFETY QUESTION ASSESSMENT:

1) Accident probability has not changed because the installation of covering across the fan discharge opening in the ring duct does not permanently alter the ventilation system. Covering the opening performs the same function of the removed damper by preventing air flow through the duct opening. The installation and removal process will be performed when the ventilation system is not required (except for personnel accommodation), shut down and de-fueled, or when the purge and exhaust penetrations are closed. During damper replacement activities the containment air recirculation system will be maintained and available. The closure of the duct opening with the damper removed does not increase the probability of occurrence of an accident because the ventilation system function and operation will be the same. The CARF Ventilation System can not initiate an accident.

2) Accident consequences are not increased. The installation of covering over the discharge duct opening provides the same function as a CARF damper by preventing air flow from the ring duct back towards the non-running fan(s). The activity does not change the ability for operation of the system in performance of its intended function.

3) No unique accident probabilities/possibilities are created.

4) Margin of Safety is unchanged. The CARF ventilation system is maintained and available to provide a ventilation flow path.

Thus installation and removal of CARF discharge duct opening covers creates no Unreviewed Safety Question.

00-SE-PROC-28

Description
0-MCM-1910-01 Installation and Removal Of Temporary Drinking Water Fountains in Radiologically Controlled Areas.
This Safety Evaluation addresses the installation and removal of temporary drinking water fountains inside of the Auxiliary Building, Fuel Building, and Containment Buildings. It also addresses routing of temporary piping/tubing for Domestic Water supply to the fountains.

Summary
Temporary Domestic Water lines (piping/tubing) and water fountains will be installed by 0-MCM-1910-01 when desired to supply drinking water to the Auxiliary Building, Fuel Building, and the Containment Buildings for personnel refreshment during work activities.

Domestic Water will be tapped off of existing lines and valves (drain valves or hose connection isolation valves) provided by the system design. The supply valves are chosen based on the availability and proximity of the valves to the desired service area. The temporary water fountain supply lines (piping/tubing) will be routed through adjacent areas to the service area. The lines will be routed in overhead areas when possible. The lines will be routed in a manner that will ensure that they will be sufficiently secure when filled with water. Should it be necessary to route the lines along or across any floor areas, the lines will be adequately secured to prevent a tripping hazard. The temporary lines and water fountains will be installed in accordance with the seismic restraint or separation distance requirements of VPAP-0312 (Seismic Housekeeping and Temporary Structures and Trailers Inside the Protected Area). Following installation and removal of the temporary supply line associated with service inside the Auxiliary Building, the penetration in the pipe tunnel between the Turbine Building and Auxiliary Building will be verified to be sealed to provide a proper barrier. The temporary supply line associated with service inside the Containment Building will be properly routed through the temporary equipment hatch plate in accordance with DCP 93-181-3, Rev. 5, which designated a penetration check valve and manual isolation valve to be used to tie in a temporary water line to the Containment for water fountain use. Routing of the supply line in this manner will ensure that the line will not adversely affect the Containment boundary isolation (Containment Integrity) when required (for more details see 94-SE-MOD-050, Rev. 1). The fountain dispenser will have a foot pedal to allow workers to obtain potable water without using their potentially contaminated hands. The fountain will facilitate consumption of domestic water by personnel working in a high heat area without the need to exit the work area. Paper drinking cups will not be provided.

All components will be selected to convey potable water below 200 degrees F, at normal operating pressures of the Domestic Water system, which range between 46 and 58 pig. Following installation, the supply lines will be checked for leakage and corrected as required. In addition, the water fountains will be operated to verify proper flow through the fountain drain line to an appropriate disposal drain or sump. Any subsequent leakage from the installation will be directed to the associated area sump. Leakage from lines routed in outside areas will drain to the yard or the storm drains and will pose no adverse environmental effects. Leakage inside of RCA areas will be considered potentially contaminated and will be properly processed via the respective area sump as liquid waste. Operators would be alerted to any leakage in excess of the respective sump pump capacity by a sump high level alarm. In addition, excessive DW leakage would bring in the domestic water tank low level alarm.

For personnel safety, the drinking fountain(s) will be tested prior to use/human consumption to ensure potable water is achieved.

This change will not affect the operability of the remaining DW system. The change does not alter the equipment or operation of the power station and its safety systems. The ability of all systems to perform as designed is not impacted. Therefore, the probability of occurrence of analyzed accidents and malfunctions has not been increased. Likewise, the consequences of any analyzed accidents and malfunctions have not been changed. There is no potential for the creation of a new or different accident or malfunction.

Additionally, the Technical Specifications and Operating License are not affected. Therefore, this change does not constitute an unreviewed safety question and should be allowed.

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S.E. #	Unit	Document	System	Description	SNSOC Date
96-SE-OT-61 REV. 2	1,2	DCP 91-012 (F. C. 1) Request for temporary license amendment for SW	SW	Clarifies minimum SW pump operating pressure with Only one SW pump operation on the header where CCHXs are aligned. Also clarifies that there is a need for flood protection measures on charging pump cubicles when the pump is operable	1-20-00
97-SE-OT-38 REV. 2	1,2	0-OP-4.24(Rev. 8)		Rev. 2 changes the spent fuel gate slings at the crane hook from wire rope to any manufactured sling material to allow use of alternate materials such as synthetic slings	12-12-00
98-SE-OT-48 REV. 1	1,2	TS CHG 350 UFSAR FN 97-053A		Supersedes UFSAR FN 97-53 by adding back into the document the specific piezometer numbers. A NRC letter denied the deletion of these numbers (SER for TS Amendments 220 & 201, dated 12-29-99).	2-22-00
99-SE-OT-48 REV. 1	1,2	TS CHG 373 UFSAR FN 99-055 TRM CHG 41		Revision 1 adds reference to the TRM & makes minor clarifications.	3-30-00
99-SE-OT-55 REV. 1	1,2	ET CEE 99-0014 TS CHG #374, Rev1 DR N-99-1526, PPR 99-027 DR N-99-1723 & PPR 99-033		Station Battery Charger Sizing Evaluation, Rev. 0 Rev. 1 states that the battery charging current specified in TS Table 4.8-3 requires correction from 12 amps to 2 amps	2-17-00
00-SE-OT-01	1,2	UFSAR FN 99-052		Incorporates new design basis calc for minimum delivered AFW flow (ME-0579, Rev. 2).	1-06-00
00-SE-OT-02	1,2	UFSAR FN 99-045		Clarifications & changes to UFSAR sections that discuss NAPS RHR systems as a result of the Configuration Mgmt review	1-06-00
00-SE-OT-03	1,2	Design review for release of safety monitor models N7E & N7F safety monitor shutdown model		The safety monitor will be implemented at NAPS for shutdown risk evaluations as required by 10 CFR 50.65	1-06-00
00-SE-OT-04	1,2	Design review for release of ORAM		The ORAM Code will be implemented at NAPS for outage safety evaluations as per VPAP-2805	1-06-00
00-SE-OT-05	1	Tech Rpt NE-1167 (Rev. 2)		Implements a temperature coastdown at end of cycle for N1C14, followed by a power coastdown, as an alternative to the usual EOC power coastdown operation. The Tavg coast will be no more than 5°F with a ±2°F operating band as evaluated in RSE Tech Rpt NE-1167, Rev. 2	1-06-00
00-SE-OT-06	1,2	TRM Chg #35		Clarifies fire protection systems & equipment criteria & ensures adequate compensatory measures will be implemented if equipment is found to be inoperable. Also clarifies wording & action descriptions to avoid misinterpretation.	1-11-00

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S.E. #	Unit	Document	System	Description	SNSOC Date
00-SE-OT-07	1,2	UFSAR FN 99-068 Section 9.3.2.1.5		Clarifies where secondary chemistry conductivity monitors may be recorded & where OLCMS alarms will occur	1-11-00
00-SE-OT-08	1,2	UFSAR FN 99-042		Clarifications & changes to UFSAR sections that discuss NAPS vital bus power system & station service system as a result of the Configuration Management review	1-20-00
00-SE-OT-09	1,2	UFSAR FN 2000-01 Section 4.2.1.4.3		Eliminates postshipment inspections that have been performed during preshipment inspections. Also deletes the details of drag testing of fuel & insert components (covered in fuel dept. procedures).	1-27-00
00-SE-OT-10	1,2	UFSAR FN 99-043		Changes UFSAR sections that discuss NAPS boron recovery / waste disposal systems as a result of Conf. Mgmt review	2-01-00
00-SE-OT-11	1,2	UFSAR FN 99-074 0-GOP-7.1 (R. 0)		Deletes detailed design information from the UFSAR associated with the filers in the chemical volume & control system & refueling purification system. A procedure is being generated to provide guidance and strategy for filter changeout.	2-08-00
00-SE-OT-12	1,2	UFSAR FN 99-033		Provides discussion on why the manual Halon system installed in the ESGRs meets the requirements of a fixed fire protection system as per 10CFR50, App. R, Section III.G.3, in contrast to Section III.G.w, which requires an automatic suppression system.	2-08-00
00-SE-OT-13	1,2	-QA Topical Rpt FN 2000-04 -UFSAR Chapter 17.2		Redefines records retention requirements for operating phase records, clarifies definition of "QA Record", & defines "Lifetime" as a record retention period	2-17-00
00-SE-OT-13 REV. 1	1,2	QA Topical / UFSAR FN 00-04A		Rev. 1 incorporates NRC comments that includes (1) appropriate definition of lifetime with each record type, (2) re-stating the retention requirements for training materials, & (3) changing 3 years to 3 cycles as the retention period for requalification / retraining records	831-00
00-SE-OT-14	1,2	UFSAR FN 2000-03		Changes UFSAR sections that discuss North Anna's instrument air system as a result of the Config. Mgmt review	2-22-00
00-SE-OT-15	1,2	TRM CHG #38		Converts the existing TRM Microsoft Word documents into a FrameMaker document that will be controlled & maintained by the Configuration Management group	2-24-00
00-SE-OT-16	1,2	UFSAR FN 99-067		Makes changes to UFSAR sections that discuss North Anna's fuel handling & storage systems as a result of the Configuration Management review	3-02-00
00-SE-OT-17	1,2	Chemistry Special Order #00-002		Allows the use of Calgon Biocide – EVAC in the bearing cooling system to alleviate the clam activity.	3-09-00
00-SE-OT-18	1	RSE Tech Rpt NE-1229, Rev. 0		Refueling & operation of North Anna Unit 1, Cycle 15 at a shift average core power not exceeding 2893 MWt.	3-09-00

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S.E. #	Unit	Document	System	Description	SNSOC Date
00-SE-OT-19	1,2	UFSAR FN 99-072		Changes UFSAR sections that discuss North Anna's high energy lines as a result of the Conf. Mgmt. review	3-14-00
00-SE-OT-20	1,2	UFSAR FN 00-002		Changes UFSAR sections that discuss North Anna's electrical & communication systems as a result of the Configuration Management review	3-14-00
00-SE-OT-21	1,2	UFSAR FN 2000-07 Section 5.5		Corrects the previous assumption that the thermal barrier will cool the bearing & seal water to an acceptable level (for extended operation) during a loss of seal injection.	4-13-00
00-SE-OT-22	1,2	TS CHG 378 PI N-2000-0490-E2		1) Documents the ability of the current control room bottled air pressurization system to meet its intended design function. 2) Supports TS Change request #378 – which will revise surveillance requirement 4.7.7.2.a to increase the minimum number of compressed air bottles from 84 to 102 bottles to maintain its design function, based on results of Engineering Study 87-08	4-13-00
00-SE-OT-23	1,2	UFSAR FN 99-075 Topical Rpt Change to Section 17.2.17		Allows retention of quality records in various electronic formats	4-18-00
00-SE-OT-24	ISFSI	TN-32 Dry Storage Cask TSAR, R. 9A NAPS ISFSI SAR		Allows use of TN-32, Cask 21 at NAPS. The inner shell weld does not meet requirements of ASME Section V, article 2, paragraph T-282.2. Two NDE Level III inspectors have examined radiographs & deemed weld acceptable.	4-20-00
00-SE-OT-25	1,2	UFSAR FN 99-062		Changes UFSAR sections that discuss North Anna's fire protection system as a result of the Conf. Mgmt review.	4-20-00
00-SE-OT-26	1,2	UFSAR FN 99-069		Changes UFSAR sections that discuss North Anna's fuel pit cooling, chilled water, turbine generator, & bearing cooling water systems as a result of Conf. Mgmt review	4-25-00
00-SE-OT-27	1,2	UFSAR FN 2000-19		Changes UFSAR sections that discuss North Anna's electrical – instrumentation and plant computer system (EI system) as a result of the Configuration Mgmt review.	4-27-00
00-SE-OT-28	1,2	UFSAR FN 99-044		Changes UFSAR sections that discuss North Anna's ventilation system as a result of the Conf. Mgmt. review	4-27-00
00-SE-OT-29	1,2	TS CHG 376 UFSAR FN 2000-16		Extends the cumulative core burnup applicability limits for NAPS U1&2 to 32.3 EFPY & 34.3 EFPY, respectively.	5-16-00
00-SE-OT-30	1,2	UFSAR FN-2000-21		Changes Section 6.2.1.3.2.5 to state that S/G & pressurizer cubicle blowout panels are steel sheet metal vs. stainless steel. Will also state that a raised steel dome with holes is over the 6"" refueling cavity drain opening vs. a raised wire basket. Section 6.2.2 is being changed to state that the containment recirculation sump first stage coarse mesh screen opening in U2 is 0.558" vs. 0.615".	5-11-00

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S.E. #	Unit	Document	System	Description	SNSOC Date
00-SE-OT-31	1,2	TS CHG 375 UFSAR FN-00-27		Increases the boron concentration in the RWST, casing cooling tank, & spent fuel pool from current TS limits of 2300-2400 ppm to 2600 – 2800 ppm and for the safety injection accumulators, from 2200-2400 ppm to 2500 – 2800 ppm.	5-23-00
00-SE-OT-32	1,2	TS CHG 369A		Establishes additional reactivity controls & surveillance for the vacuum-assisted backfill method of returning a loop to service. Also modifies the boron concentration requirements for an isolated-filled loop to eliminate unnecessary boration.	5-23-00
00-SE-OT-33	1,2	TS CHG 379 UFSAR FN 00 26 1&2-PT-17.2		Eliminates seismic allowance adjustment from RCCA (rod) drop time surveillance criteria, replaces existing rod drop time test criteria with test limits based on safety analysis limits & design uncertainties, monitors & trends rod drop time data to identify & evaluate any adverse trends, & removes reference to seismic adjustment from TS ¾.1.3.4 & UFSAR Section 4.2.3.4.2.	5-23-00
00-SE-OT-34	1,2	UFSAR FN 00-029 Section 3.8.4.8		Revises UFSAR methodology for calculating SW reservoir water losses to agree with 0-PT-75.8 (which uses simple mass balancing calculations).	6-06-00
00-SE-OT-35	1,2	UFSAR FN 99-063		Reflects a change in the allowable ECCS leakage from 900 cc/hr to 4800 cc/hr. Also reflects allowable leakage from the filtered & unfiltered areas.	6-06-00
00-SE-OT-36	1,2	UFSAR FN 00-025		Provides a description of the licensing basis for feedwater isolation for the rupture of a main steam pipe accident in Chapter 15 & makes editorial clarifications in Table 3C-3	6-22-00
00-SE-OT-37	1,2	UFSAR FN 00-031		Revises Section 6.2.1.2.5 to indicate that the NS surge tank & associated supports will maintain their structural integrity during a design basis earthquake & will not impact the integrity of SR components.	6-27-00
00-SE-OT-38	1,2	UFSAR FN 00-013		Changes UFSAR sections that discuss NAPS reactor mechanical design system as a result of the Configuration Management review	7-06-00
00-SE-OT-39	1,2	UFSAR FN 99-076 TRM Chg #39		Removes references that imply that the fuel transfer canal manual isolation valves are containment isolation valves	7-13-00
00-SE-OT-40	1,2	UFSAR FN 00-010		Explains the environmental impact of a turbine building high energy line break on the control room envelope and the EDG rooms	7-18-00
00-SE-OT-41	ISFSI	NCR 781 NAPS ISFSI SAR		Allows use of the TN-32 Cask 19 at NAPS – PCC had failed to perform a post-heat treatment inspection on the weld repair	7-20-00
00-SE-OT-42	1,2	UFSAR / ISFSI Chg FN 2000-034 Topical Rpt Chg dated 7-25-2000		Deletes the title of "Radwaste Team Leader" from Chapter 17, Section 17.2.1.2.e & Figure 17.2.1-2 of the UFSAR. This is a Surry only position & has no impact on North Anna. However, the UFSAR requires this change.	7-27-00

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S.E. #	Unit	Document	System	Description	SNSOC Date
00-SE-OT-43	1,2	NAPS App. R Report Chg Request #1999-N-015		Changes Sections of the App. R. Report that discuss NAPS fire protection system as a result of the Configuration Management review.	8-01-00
00-SE-OT-44	1,2	UFSAR FN 00-033 ISFSI SAR IN 2000-002 PI N-2000-1212		Removes the fence & gate symbols from the documents since there is no longer a fence & gate around the SW pump house since it is not part of the protected area.	8-10-00
00-SE-OT-45	1,2	TS Chg 339 UFSAR FN 00-037		- Adds new TS requirements ¾.7.14 & TS 3/4.7.15 for spent fuel pool soluble boron concentration & fuel assembly loading restriction based on burnup & enrichment. This will permit elimination of the Boraflex credit from the spent fuel pool criticality calculations. - Also increases the maximum fuel enrichment of the reload fuel from 4.3 to 4.6 weight percent U-235.	8-17-00
00-SE-OT-46	1,2	00-TSR-041		Installs, uses, & removes a temporary radiation shield of borated poly-panels on the top of a loaded spent fuel cask in the decon bldg north bay	8-23-00
00-SE-OT-47	1,2	TRM #42	FP	Enhances Fire Protection Program to reflect results from ET SE-99-073, Rev. 0 (TRM Table 7.2-2 Revision) in response to PI N-1999-2748 and ET NAF-96-173, Rev. 2 (Allowed Duration for EQ Door Breaches). Provides consistency with VPAP-2401	8-29-00
00-SE-OT-48	1,2	UFSAR FN 99-037		Changes UFSAR sections that discuss North Anna's nuclear design system as a result of the Configuration Management review	8-29-00
00-SE-OT-49	1,2	ET CEP 00-0024 (Rev. 1)		Revises Exemption Request 27 by deleting reference in the exemption to excore neutron flux monitoring. Nuclear instrumentation is not required for a containment fire.	9-19-00
00-SE-OT-50	1,2	00-TSR-042		Temp. shielding request to install temporary lead blanket shielding on operable / operating 6" diameter safety related CH piping on the inlet to the charging pump to provide protection to workers installing DCP 99-010 for replacement of SW lines in the charging pump cubicles.	9-27-00
00-SE-OT-51	1	TS CHG 381		Involves a "cleanup" of the operating license for Unit 1 by removing inappropriate or unnecessary requirements as a precursor to license renewal & conversion to ITS	10-12-00
00-SE-OT-52	2	UFSAR FN 00-041		Updates UFSAR to include revised 10-CFR-50.61 pressurized thermal shock (PTS) screening calculation results for NAPS U2 based on currently available reactor vessel materials surveillance data, including that associated with recently-analyzed NAPS U2 Capsule W.	10-17-00
00-SE-OT-53	1,2	UFSAR FN 00-023		Makes corrections & clarifications to the Reactor Thermal & Hydraulic Design System as a result of the Configuration Management review.	10-26-00

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S.E. #	Unit	Document	System	Description	SNSOC Date
00-SE-OT-54	1,2	NAPS submittal to NRC for ITS conversion		NAPS submittal to NRC for conversion of current Tech Specs to Improved Tech Specs (ITS)	11-06-00
00-SE-OT-55	2	TS CHG #382		Removes completed, redundant, expired, or otherwise non-applicable license conditions from the Facility Operating License for North Anna Unit 2.	11-08-00
00-SE-OT-56	1.2	UFSAR FN 99-066		Changes UFSAR sections that discuss North Anna's plant safety analyses as a result of the Configuration Management review.	11-08-00
00-SE-OT-57	1,2	TS CHG 378 ET N 00-057 (R. 1)		Decreases the maximum control room emergency ventilation system differential pressure limit from < 6 inches water gauge across the HEPA filter & charcoal absorber assembly to < 4 inches water gauge across the demister filter, HEPA filter, & charcoal absorber & will increase the minimum number of compressed air bottles from 84 bottles to 102 bottles.	11-14-00
00-SE-OT-58	1,2	UFSAR FN 00-022		Makes corrections & clarifications to the reactor design – miscellaneous system as a result of the Configuration Management review.	11-14-00
00-SE-OT-59	1,2	00-TSR-037		Installs temporary lead blanket shielding on operable / operating 2" diameter safety related DG piping to provide protection during DCP 99-010 work for replacement of SW lines in the aux. bldg. basement.	11-30-00
00-SE-OT-60	1,2	TS CHG 376A UFSAR FN 00-048 (supersedes FN 00-016)		Incorporates the effects of instrument uncertainties into the proposed revised design basis RCS P/T limits, LTOPS setpoints, & LTOPS enable temperatures for NAPS Units 1 & 2. The revised analysis bases extend the cumulative core burnup applicability limits to 32.3 EFPY for Unit 1 & 34.3 EFPY for Unit 2. This 50.59 also supports a reduction in the Units 1 & 2 reactor vessel head bolt-up temperatures from 90°F to 60°F.	12-19-00

96-SE-OT-61 Rev 2

Description

Request for Temporary License Amendment for Service Water Preservation - Phase I (Repair/Replacement of Concrete - Encased Piping to/from CCHXs to be performed under DCP-91-012).

The SW piping is experiencing internal wall degradation due to general corrosion and relatively rapid wall loss in localized areas (pits) due to microbiologically influenced corrosion (MIC). Investigation on encased in concrete pipes under DCPs 91-009, 10, and 11 shows significant deterioration of the SW piping. In order to perform repair/replacement of 24" SW headers to CCHXs from their connection with 36" headers up to isolation valves within the Auxiliary Building (note that exposed piping from the isolation valves to CCHXs is being replaced under DCP-94-010), a one time amendment of the TS will be required to allow 35 day isolation of one of the two 24" SW headers to the CCHXs (two periods for 35 days, one for each header). Also, NRC approval is required for the temporary crossconnect installations and defeating the automatic closing of SW MOVs to CCHXs on a CDA signal during the time of CCHX operation from one SW header. Availability of four SW pumps in the pre-accident condition and throttling of the bypass valve will substitute for the automatic closing of SW motor operated valves to the CCHXs of the affected unit on a CDA signal and the required SW flow will be delivered to the RSHXs of the affected unit (Calculation ME-0420). Note that a similar amendment was approved by the NRC in October of 1995 for repair/replacement of exposed SW portion of the above piping (DCP-94-010) which is a continuation of the concrete-encased piping.

Summary

The SW piping is experiencing internal wall degradation due to general corrosion and relatively rapid wall loss in localized areas (pits) due to microbiologically influenced corrosion (MIC). Investigation on encased in concrete pipes under DCPs 91-009 10 11 shows significant deterioration of the SW piping. Under this Design Change incased in concrete 24" SW headers to/from CCHXs from their connection with 36" headers up to isolation valves within the Auxiliary Building will be repaired (note that piping from the isolation valves to CCHXs is being repaired under DCP-94-010). A one time amendment of the TS will be required to allow 35 day isolation of one of the two 24" SW headers to the CCHXs (two periods for 35 days one for each header). Note that similar amendment was approved by the NRC in October 1995 for repair/replacement of exposed portions of the above piping (DCP-94-010).

This DC does not involve an unreviewed safety question:

Repair of the concrete encased SW lines to/from CCHXs will temporarily limit SW supply to CCHXs since while one SW header is out of service, all CCHXs will be supplied from one SW header. Note that during this repair only CCHXs will be supplied from one SW header since within first 168 hour Section 3/4.7.4.1.d TS AS of isolation of the header which should be repaired, temporary 10" diameter SW lines (one supply and one return) will be installed to supply the SW to the charging pumps, instrument air compressors, Unit 2 CR chillers, and spent fuel pool (SFP). These temporary lines will be routed from operating part of the 36" SW headers while 24" headers to CCHXs are being repaired. The CC water system (CCWS) is an intermediate cooling system which transfers heat from heat exchangers containing reactor coolant or other radioactive liquids to the SW system. The design basis of the CC water system is a fast cooldown of one unit while maintaining normal loads on the other unit. The CCWS is not a system which functions to mitigate a DBA or presents a challenge to the integrity of a fission product barrier. Calculation ME-0420 Rev.1 Add. B through J were performed to verify operation of three and four CCHXs on one SW header. During the repair/replacement work, there will be two SW pumps on one SW header to which three or four heat exchangers are connected. Calculations show that to satisfy design basis requirement for the component cooling system (fast cooldown of one unit while maintaining normal loads on the other unit), SW temperature in the reservoir should not be above 75°F due to SW flow limitation. This factor limits implementation of the SW piping replacement around CCHXs to time frame between October and April when temperature in the SW reservoir can be maintained below 75°F. If SW temperature is between 75°F and 78.5°F CCHX will not be able to supply cooling water of sufficiently low temperature to the RHR heat exchangers to meet the fast cooldown requirements of one unit while other unit is in operation. If the SW temperature exceeds 78.5°F three SW pumps should be aligned to the header supplying the CCHXs while one SW pump operates on another header. This realignment is required only during RHR operation i.e. during unit shutdown. During installation of the plugs in the 24" SW lines to CCHXs and crossconnects between the supply and return lines on each main SW header four TS 168 hour Section 3/4.7.4.1.d Action Statements (AS) will be required. Repeated isolation of the SW headers was previously analyzed and found to have a small effect (1.2-E8) on the core damage frequency (CDF) for both Units. After installation of the plugs (during repair work) two SW headers will be available to supply RSHXs CR Chillers and Charging Pumps. As was mentioned above temporary lines will be installed for the second supply of the charging pumps instrument air compressors Unit 2 CR chillers and SFP. Repair of the encased in concrete 24" diameter SW headers to CCHXs and exposed 24" SW lines between Auxiliary Building wall and isolation valves to/from CCHXs will require isolation of the 24" SW to CCHX for 35 days. A one time amendment of the TS will be required to allow 35 day isolation of one of the two 24" SW headers to CCHXs (two periods for 35 days one for each header). Also the NRC will be requested to exempt the provisions of TS Section 3.D.4 during the two 35 days isolation of one SW header to the CCHXs. During isolation of one SW header to CCHXs no SW pump maintenance or testing will be planned. This activity does not change SW or CCW system configuration (except for the temporary crossconnects) does not create the possibility of the accident of a different type than was previously evaluated in the Safety Analysis Report and insignificantly increases the probability of occurrence of previously analyzed accidents. This TS change request is similar to granted earlier by the NRC TS Change Request No.317 (the NRC Letter Serial 95-540 dated October 11 1995). The only difference is duration which is 35 days versus 49 days. Therefore previously performed analysis is applied and this analysis is conservative due to shorter term of the SW isolation. A review of the equipment affected by this phase of the SW restoration project was performed to evaluate the impact on initiating event frequency. Since the SW system and CC system are support systems used to remove heat a failure in either of these systems does not affect the initiating event frequency of any design basis event. Additionally an estimate of the impact on core damage frequency is provided below. The impact on the North Anna Probabilistic Safety Assessment (PSA) during implementation of this DCP is similar to impact of work performed under DCP-94-010 since scope of work of both DCPs is repair/replacement of different portions of the same 24" SW headers to CCHXs. The only difference from PSA standpoint is that CDF for DCP-94-010 was calculated based on 140 days supply of CCHXs from one SW header while per this DCP it is only 70 days. Therefore results of PSA evaluation for DCP-94-010 are conservatively applied to this DCP.

The assumption that neither unit is utilizing RHR while only one SW header is available is due to the inability to quantify CDF associated with shutdown conditions. PSA experience indicates that this could be an increase in risk. If a steam generator tube rupture occurs the unit should be placed on RHR as necessary. If RHR is needed for any other reason then the best course of action for restoring the second SW header and utilizing RHR cooling should be evaluated based on the DCP status. It may be determined based on the DCP status and the need for RHR cooling that the best course of action is to initiate RHR while only one SW header is operable. The evaluation should include consideration of stopping all SW work to minimize potential damage to the only operable header. It should also consider expediting restoration of the second header and temporarily delaying further DCP work until the unit is off of RHR. This is not intended to allow beginning DCP work when it is known that the unit is scheduled for an outage requiring RHR operation.

Installation of crossconnects between supply and return SW headers with a manual valve on the crossconnect will ensure normal operation of the SW pumps and satisfy GDC-5 "Sharing of Structures Systems and Components" between both units. When one out of two 24 diameter SW header to CCHXs is being repaired, the crossconnect valve will be throttled to ensure SW pump flow of approximately 7400 gpm. It is assumed that all four SW pumps are available in the pre-accident condition and three during the accident. In case of CDA the throttled bypass valve will satisfy the condition of delivering through the SW headers the necessary flow to RSHXs on the accident unit (above 4500 gpm to each of four RSHXs). Availability of four SW pumps in the pre-accident condition and the throttling of the bypass valve will substitute for the automatic closing of motor operated valves to CCHXs of affected unit on CDA signal which will be temporary disabled for the time of CCHX operation from one SW header. NRC approval is required to implement this temporary crossconnect installation and for defeating the automatic closing of SW motor operated valves (MOVs) to CCHXs on a CDA signal during the time of CCHX operation from one SW header. Note that a conservative compensatory requirement will be imposed for the periods of operation while in the 168 hour AS within the 35 day AS. During this time the automatic signal to close the CCHX MOVs on a CDA signal will not be temporarily defeated. Also, when the 168 hour AS is entered simultaneously with the 35 day AS requirement of TS AS 3.7.4.1.d concerning ASW pump availability will be adhered to.

The following actions will be incorporated into the DCP to prevent potential deterioration of the existing SW pumps during extended periods at flow rates above 14000 gpm when very little margin exists between the available and required NPSH. This limitation is based on the recent (November 16 1995) NPSH test performed by Johnston Pump during manufacturing the SW pump replacement (see JCO-95-03):

1. To prevent SW pump operation at high flow rate when CCHXs are aligned on one SW header, the heat exchangers will be throttled to limit SW pump discharge pressure to 40 psig minimum.

2. If during implementation of this DCP, a DBA occurs (SI/CDA initiated), SW flow to two out of four RSHXs should be isolated to reduce flow on the running pumps. This action should occur after the containment pressure is stabilized at subatmospheric conditions and before SW reservoir level reaches 313 ft. This will ensure that SW flows are reduced within the first two hours of the accident and pump runout will not be a concern. Since the reservoir level should remain above 313 ft during first 24 hours after the DBA initiation (Calculation ME-0305) a caution will be added to the procedures regarding maintaining a reservoir level above 313 ft which will ensure that operators are aware of reducing SW flow if more than two hours have passed and two RSHXs have not been secured. Per recommendations of JCO-95-03 station operating and emergency procedures have been revised direct operations to implement isolation of two RSHXs after containment pressure stabilized between 10.5 psia and 13 psia after SI/CDA initiation. The RSHXs which are secured shall be one RSHX associated with one inside RS pump and one RSHX associated with one outside RS pump if possible to maintain a full coverage spray pattern. A caution is added that if less than four SW pumps are running and SW reservoir level drops to 313 than SW pumps must be closely monitored for cavitation and SW flow reduced accordingly. For list of revised procedures see JCO-95-03. This caution is not required if all SW pumps are replaced prior to this DCP implementation.

3. Although it is not necessary bypass valve 1-SW-1337 or 1-SW-1338 may be closed after the DBA initiation. Although no time limitation is imposed this action should take place as soon as practical after the DBA initiation. This action will increase flow to RSHXs and decrease SW pump flow rate. Also as contingency measure CCHXs of the affected Unit may be manually isolated after the DBA initiation (note that no time limit is imposed). This action will increase flow to RSHXs. As stated the safety analysis does not take credit for these actions.

A special procedure will be developed for this mode of operation and referenced in DCP-91-012.

97-SE-OT-38 Rev 2

Description
0-OP-4.25, Rev 8 – "Movement of Fuel Pit Gates"
Change the spent fuel gate slings at the crane hook from wire rope to any manufactured sling material in order to allow use of alternate materials such as synthetic slings. This change is not applicable to the safety cables.

Summary
SFP gate rigging and movement is subject to NUREG-0612, Control of Heavy Loads. In late 1997, Safety Evaluation 97-SE-OT-38, Rev 1 prescribed the required changes to 0-OP-4.25 in compliance with commitments to NUREG-0612 but inadvertently limited the slings to those manufactured of wire rope. At the time 97-SE-OT-38, Rev 1 as prepared, wire rope slings were routinely being used for general rigging. Station use and acceptance of other sling materials (such as synthetics) had increased over the years and should be allowed were appropriate. There is no technical basis or NUREG-0612 commitment which would mandate use of wire rope slings at the crane hook.

Changing spent fuel pit gate slings at the crane hook from wire rope to any manufactured sling material ill not (1) increase the probability of a fuel handling accident, (2) increase the consequences of a fuel handling accident, or (3) create the possibility for a new type of accident.

98-SE-OT-48 Rev 1

Description

Technical Specification Change Request No. 350
UFSAR Change Request, No. FN 97-053A
Delete all references to the specific piezometer numbers in Section 4.7.13.1, referring only to the three zones that are required to be monitored. Modify Tech Spec 3/4.7.13, Table 3.7-6 to raise the allowable groundwater levels along the South East zone of the SWR dike represented by piezometers P-10, P-21, and P-22, from el. 277 ft. to 280 ft. at the toe of slope and from el. 280 ft. to el. 295 ft. at the crest of slope.

Summary

As detailed in a report (Service Water Reservoir Groundwater Level Evaluation, NP-3141, dated December 1996), which was submitted to the NRC on December 10,1996 (Serial No. 96-561), water levels in piezometer P-22 exceeded the Technical Specification (Tech. Spec.) allowable limit on September 13, 1996 (see DRN 96-1835). P-22 is located along the southeast section of the dike. Monitoring of P-22 has indicated the water level in this section has slowly risen. Calculation CE-1386 was performed to determine the stability along the SE section of the SWR dike under an increased water level. Results of this calculation show that the allowable water levels as contained in the Technical Specification may be raised without decreasing the stability of the slope below the Factor of Safety defined by the original design basis calculation.

In addition, Section 4.7.13.1 Surveillance Requirements will be revised to delete all references to individual piezometers numbers and instead list the limiting water levels for the three zones to be monitored. The format for Table 3.7-6 will be revised to eliminate specific piezometer numbers and instead refer to monitoring zones around the dike and piezometer locations such as crest or toe of slope. The remaining pneumatic piezometers, P-10, P-11, and P-15 will no longer be monitored. P-15 has failed and is abandoned and the water level has dropped below P-11's tip so the actual water level cannot be monitored. Pneumatic piezometer P-10 has been replaced by an open standpipe piezometer. Pneumatic piezometers were installed prior to filling the SWR since they can detect rapid changes in water levels. However, pneumatic piezometers tend to fail over time (10 to 20 years). Since current groundwater fluctuations are minimal, the more dependable open tube piezometers have been installed to replace the pneumatic piezometers.

The UFSAR will be revised to reflect these changes and to provide a figure that depicts the current monitoring requirements as defined by this Tech. Spec. Change Request.

Changing the Tech Specs to raise the allowable water level in the Southeast section of the SWR will not 1) increase the probability of an accident by reducing the stability of the SWR slope below that which was calculated in the original design basis calculation, 2) increase the consequences of a slope failure of the SWR, or 3) create the possibility of a new accident.

The proposed changes will not 1) increase the probability of occurrence of malfunctions greater than that defined in the original design basis calculation or 2) increase the consequences of malfunctions causing failure of the SWR slope. The proposed changes will not create the possibility for a malfunction of equipment or failure of a different type than was previously evaluated in the SAR.

The piezometers which are affected by this Tech. Spec. Change Request are at least 300 ft. from the closest electrical or piping system located at the SW Pump House.

On December 29, 1999, the NRC issued Amendments 220 for Unit 1 and 201 for Unit 2 to the NAPS Technical Specifications by letter (serial number 99-630), "North Anna power Station, Units 1 and 2 – Issuance of Amendments regarding a Technical Specification Change to Allowable Groundwater Elevations Associated with the Service Water System." This letter also denied our proposal to eliminate the specific piezometer within the zone as well as the piezometer itself. Based upon this denial, UFSAR Change Request FN 97-053A will maintain the piezometer device within the UFSAR.

99-SE-OT-48 Rev 1

Description

Technical Specification Change Request No. 373
UFSAR Change FN 99-055
Technical Requirements Manual (TRM) Change Request No. 41
The Technical Specification change with supporting Technical Requirements Manual Change Request No. 41 will relocate Section 3/4.6.4.3, "Waste Gas Charcoal Filter System," from the Technical Specifications to the new Technical Requirements Manual Section 5.4.

Correct the values stated in UFSAR Section 15.3.5 for total curie content assumed released in WGDT rupture.

Summary

The proposed changes will remove the operability and surveillance requirements for the Waste Gas Charcoal Filter System from the Technical Specifications and relocate these requirements in the Technical Requirements Manual (TRM). In addition, the values assumed in the WGDT rupture are being revised to reflect the current calculation values.

A waste gas decay tank rupture is highly unlikely, as the waste gas decay tanks are designed and constructed with pressure relief valves to prevent overpressurization, are missile-shielded by installation below grade, and have their gaseous contents controlled to prevent potentially explosive mixtures. The entire gaseous content of the waste gas decay tank is assumed to be released to the atmosphere as a ground-level release. Although the Technical Specifications limit the content of each tank to less that 25,000 curies of noble gases (TS 3.11.2.6), the total activity assumed to be released during a waste gas decay tank rupture is 73,000 Ci of Xe-133 equivalent and 0.084 Ci of I-131 equivalent (SWEC Calculation No. RP-11715-A109-0,"Waste Gas Decay Tank Burst for FSAR & (Rev. 1) & (Rev. 2)", 10/25/72). The waste gas charcoal filter system is not credited for any mitigation of the release in the accident analysis. In addition, the releases associated with a waste gas decay tank rupture are bounded by the existing LOCA releases. Specifically, operation of the North Anna Power Station in accordance with the proposed Technical Specification changes will not:

1. Involve an increase in the probability or consequences of an accident previously evaluated.

Relocating the operability and surveillance requirements for the Waste Gas Charcoal Filter System to the TRM and correcting the Curie content of the WGDT assumed in the accident analysis do not change the operation of the plant. The plant and the radioactive gas waste system will not be operated differently. No new accident initiators are established as a result of the proposed changes. Therefore, the probability of occurrence is not increased for any accident previously evaluated.

Relocating the operability and surveillance requirements for the Waste Gas Charcoal Filter System to the TRM and correcting the Curie content of the WGDT assumed in the accident analysis do not effect the gaseous releases to the environment, which are controlled by the ODCM. Additionally, no credit for these filters is taken in the accident analysis for Waste Gas Decay Tank rupture. Therefore, there is no increase in the consequences of any accident previously analyzed,

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not affect the operation of the plant. The gaseous waste systems will not be operated differently as a result of the proposed changes. No new accident or event initiators are created by moving the operability and surveillance requirements for the Waste Gas Charcoal Filter to the TRM and correcting the Curie content of the WGDT assumed in the accident analysis. Therefore, the proposed changes do not create the possibility of any accident or malfunction of a different type.

3. Involve a reduction in the margin of safety as defined in the bases on any Technical Specifications.

The proposed changes have no effect on any safety analysis assumptions. Credit for the waste gas charcoal filters is not taken in the accident analysis for a Waste Gas Decay Tank rupture. Therefore, the proposed changes do not result in a reduction in a margin of safety. Releases are controlled by the ODCM.

99-SE-OT-55 Rev 1

Description

- Engineering Transmittal CEE 99-0014, Rev. 0, *Station Battery Charger Sizing Evaluation, North Anna Power Station, Units 1 and 2,* dated 10/28/99.
- Technical Specification Change Request # 374, Rev 1
- Deviation Report N 99-1526 and PPR 99-027
- Deviation Report N 99-1723 and PPR 99-033

Technical Specifications require battery charger testing @ ≥ 200 amps @ 125Vdc @ ≥ 4 hours. ET CEE-99-0014 shows that the majority of battery chargers require a capability greater than the 200 ampere minimum test value. Previously performed testing demonstrates the chargers are capable of carrying required loading except chargers 1-III, 1C-II, 2-III, and 2C-II. Admin control and procedures changes have been implemented for the outlying chargers to ensure all chargers are capable of meeting design requirements in the future. The battery charging current specified in TS Table 4.8-3 requires correction from 12 amps to 2 amps.

Summary

Based on the analysis performed in ET CEE-99-0014, the Tech Spec testing values for the batteries do not bound the required battery charger capacity. While the latest testing of the chargers demonstrates the chargers are capable of providing the required capacity with either the current admin controls in place or the proposed procedure changes, Tech Spec surveillance requirements shall be changed to ensure future testing is adequate for its intended function.

The ET also shows that the battery chargers associated with batteries 1-III and 2-III do not meet the UFSAR and design basis requirement of "charging the batteries from the maximum discharge condition to full charge in 24 hours while supplying the normal or emergency steady state loads." Changes to procedures are required to ensure these specific chargers meet these requirements. The ET determined the chargers are able to supply loads and recharge the battery within 24 hours with either the current admin controls in place or the proposed procedure changes. Therefore, no safety concern currently exits and procedure changes will ensure the future capability to meet the UFSAR/design basis requirements.

Deviation Report N99-1723 identifies an error in T.S. Table 4.8-3, note (b) which states "Or battery charging current is less than (12) amps when on charge (station batteries only)." The value of 12 amps was calculated incorrectly and will be changed to the appropriate value of 2 amps. The 2 amp value is based on a battery cell float current demand specified in Section 58.00 of the Exide battery manual. Given the float voltage of 2.25 volts per cell an the 8 hour rating for the station battery of 1800 amp hours, the float current at 100% full charge is 198 milliamps. However, meter accuracy is ~± 5% and a 0-600 amp meter is used to measure the current. Additionally, the float current would be seen for a battery at ~ 95% recharge versus the full 100% recharge documented in the Exide manual. Therefore, it is reasonable to utilize a value of "less than 2 amps" in the Technical Specifications which allows for metering equipment accuracy/readability, temperature variations, variations in float voltage, and % charge less than 100% (as would be expected immediately following a recharge). The Deviation Report identifies no indication that charging current has been less used in lieu of specific gravity measurements during the last ten years. Procedure ½-PT-85 has been revised based on DR N99-1723 to disallow using the 12 amp charging current as an acceptance criteria in lieu of specific gravity measurements.

1 The probability or consequences of an accident or malfunction previously evaluated in the safety analysis report are not increased.

No new accident precursors are introduced. Changes to Technical Specification Surveillance Requirements and procedure changes improve the operation of the battery chargers and ensure they perform as designed. Changes to the Table 4.8-3 requirements correct a previous error.

2 The possibility of an accident or malfunction of a different type than previously evaluated in the safety analysis report is not created.

Based on a review of the SAR, there are no malfunctions of equipment previously considered that could be attributed to battery chargers. The batteries operation and capabilities are not impacted by operation of the battery chargers. The battery chargers will be ensured of correct operation by the proposed changes to the Tech Spec SRs and station procedures. The probability of occurrence of malfunction of the DC power train will not be increased due to these changes. Changes to the Table 4.8-3 requirements correct a previous error.

3 The margin of safety as defined in the basis for any Technical Specification is not reduced.

The actual margin of safety is not specifically addressed in the bases section. Tech Spec bases assume the operability of at least one of each of the onsite AC and DC power sources and associated distribution systems during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite AC source. When subjected to a loss of offsite power and the EDGs energizing the bus as designed, the battery chargers are completely capable of performing their safety function. Should the EDG not energize the bus, the admin controls presently in place and the procedure changes recommended ensure the chargers continue to be capable of performing their safety function once AC power is restored. Changes to the Table 4.8-3 requirements correct a previous error. Therefore, no margin of safety is impacted.

00-SE-OT-01

Description

UFSAR Change Request No. FN 99-052

This safety evaluation supports the revision of UFSAR Table 10.4-1, "AUXILIARY FEEDWATER SYSTEM DESIGN BASIS FOR UNIT 1", to reflect the results of the recently completed new minimum delivered (design basis) AFW flow calculation, ME-0579, Rev. 2. The revised table will show the design basis flows and flow margins for both Units 1 and 2, and explain in the note that the minimum delivered flow calculation provides for margin over and above the design basis flows, (minimum required flow for accident analysis), with mini-flow recirculation and oil cooler flows included. The proposed revision to UFSAR Table 10.4-1 is attached to FN 99-052. The existing table presents the margins for each Unit 1 (only) AFW pump in terms of head (ft.) at a flow of 340 gpm. ME-0579 determines margins relative to the required design basis flow for each AFW pump (300 gpm for the MD AFW pumps and 400 gpm for the TD AFW pumps), which is much more useful than margin expressed in terms of head at a flow that is not a design basis value.

Summary

Updating UFSAR Table 10.4-1 to be consistent with the current plant design basis has no physical impact on any plant system or component, nor does it affect the operation or performance of any equipment. The Technical Specifications are not impacted by this change and the margin of safety associated with the Technical Specifications is not impacted. This change will enhance the accuracy and clarity of the UFSAR description of the Auxiliary Feedwater system design basis. This change does not create the potential for a new accident, or increase the probability or consequences of previously analyzed accidents, and does not constitute an unreviewed safety question.

00-SE-OT-02

Description

North Anna UFSAR Change Request No. FN 99-045

UFSAR Change Request No. FN 99-045 contains a list of changes, some of which are editorial in nature, that need to be corrected or clarified in the UFSAR sections that discuss North Anna's residual heat removal systems. This package is a result of the Integrated Configuration Management Project review of North Anna Power Station's residual heat removal system.

Summary

The above editorial changes are within the current design and licensing basis of the facility. These changes do not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed editorial changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

None of the proposed editorial changes involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed editorial changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

With regard to the criteria set forth in 10 CFR 50.59 and based on the evaluation summarized above, the proposed editorial UFSAR changes have been determined not to represent an unreviewed safety question.

00-SE-OT-03

Description
Design Review for Release of Safety Monitor Models N7E and N7F, North Anna Safety Monitor Shutdown Model
The Safety Monitor shutdown model will be implemented at North Anna for shutdown risk evaluations. An at-power model is already in use.

Summary
MAJOR ISSUES

The Safety Monitor™ shutdown model will be installed at North Anna Power Station. This program will perform shutdown risk calculations to support compliance with 10 CFR 50.65 (the Maintenance Rule). The results will provide a quantitative estimate of plant risk due to any shutdown maintenance configuration. The calculated risk will be used to perform a risk assessment of maintenance activities. The shutdown model will supplement both the previously installed at-power model and the current VPAP-2805 program for shutdown risk control. The current VPAP-2805 program is still the primary and limiting means of controlling shutdown risk.

JUSTIFICATION FOR CHANGE

Implementation of the Safety Monitor shutdown model will support compliance with 10 CFR 50.65 for shutdown applications. The shutdown model has been developed under the same technical and QA processes that were used for the previously installed at-power model.

UNREVIEWED SAFETY QUESTION EVALUATION

Accident Probability and Consequences. Implementation of the Safety Monitor shutdown model will have no impact upon any accident precursor. In fact, continued compliance with 10 CFR 50.65 limits plant risk and allows the plant staff to focus upon a smaller number of evolutions. As a result, accident probability is not increased.

Implementation and use of the Safety Monitor shutdown model will have a positive impact upon the ability of plant equipment to perform its function in support of accident mitigation. The use of the Safety Monitor in support of the requirements of 10 CFR 50.65 will ensure that the number of concurrent maintenance evolutions is limited. This control ensures adequate equipment availability for accident mitigation. As a result, accident consequences are not increased.

Unique Accident Probability. The Safety Monitor will neither make nor support any plant hardware or operating strategy change which produces any unique accident precursor. It solely evaluates the unit configuration risk and thereby identifies high-risk configurations to be avoided. Any future plant changes (i.e., DCP's) which may include a PSA review will include their own safety evaluation and be evaluated on their own merits for accident risk. As a result, there is no potential for an accident of a different type than those that have been previously evaluated.

Margin of Safety. The existing accident consequence acceptance criteria and safety-related equipment performance (including setpoints) will remain unchanged. As a result, the margin of safety will remain unchanged.

Conclusion. These reviews demonstrate that the implementation of the Safety Monitor shutdown model will not result in an unreviewed safety question, consistent with the guidelines of 10 CFR 50.59.

00-SE-OT-04

Description
Design Review for Release of ORAM
The ORAM Code will be implemented at North Anna for Outage Safety Evaluations.

Summary
MAJOR ISSUES

The ORAM code will be installed at North Anna Power Station. This program will perform deterministic outage safety evaluations in compliance with the requirements of VPAP-2805, *Shutdown Risk Program*. The results will provide a qualitative evaluation of plant risk due to any maintenance configuration. The result will be used to document the impact of proposed maintenance activities.

Future revisions of the North Anna model and the ORAM code will be reviewed under the Virginia Power QA system but will not be reviewed by SNSOC.

JUSTIFICATION FOR CHANGE

Implementation of ORAM will maintain compliance with VPAP-2805. The ORAM code and North Anna model have been verified to yield acceptable results when used to evaluate a recent outage.

UNREVIEWED SAFETY QUESTION EVALUATION

Accident Probability and Consequences. Implementation of ORAM will have no impact upon any accident precursor. In fact, continued compliance with VPAP-2805 limits plant risk and allows the plant staff to focus upon a smaller number of evolutions. As a result, accident probability is not increased.

Implementation and use of ORAM will have a positive impact upon the ability of plant equipment to perform its function in support of accident mitigation. The use of ORAM in support of the requirements of 10 CFR 50.65 will ensure that the number of concurrent maintenance evolutions is limited. This control ensures adequate equipment availability for accident mitigation. As a result, accident consequences are not increased.

Unique Accident Probability. ORAM will neither make nor support any plant hardware or operating strategy change that produces any unique accident precursor. It solely evaluates the unit configuration risk and thereby identifies high risk configurations to be avoided. Any future plant changes (i.e., DCP's) which may include a PSA review will include their own safety evaluation and be evaluated on their own merits for accident risk. As a result, there is no potential for an accident of a different type than those that have been previously evaluated.

Margin of Safety. The existing accident consequence acceptance criteria and safety-related equipment performance (including setpoints) will remain unchanged. As a result, the margin of safety will remain unchanged.

Conclusion. These reviews demonstrate that the implementation of ORAM will not result in an unreviewed safety question, consistent with the guidelines of 10 CFR 50.59.

00-SE-OT-05

Description
Technical Report NE-1167 Rev 2, "Reload Safety Evaluation (RSE) North Anna 1 Cycle 14 Pattern XY with EOC Tavg Coastdown," January 2000.
To implement a temperature coastdown at end of cycle for N1C14, followed by a power coastdown, as an alternative to the usual EOC power coastdown operation. The Tavg coast will be no more than 5 ^{0}F with a ±2 ^{0}F operating band as evaluated in RSE Tech Report NE-1167 Rev 2.

Summary

This safety evaluation has been performed to determine whether an unreviewed safety question will result from implementing EOC Tavg coastdown operation for North Anna Unit 1 Cycle 14. A previous safety evaluation [1] covered the refueling and normal operation of N1C14 including the usual power coastdown. This safety evaluation supplements the previous evaluation, and deals only with the changes required for Tavg coastdown operation at EOC.

The temperature coastdown option was approved by a separate 10CFR50.59 evaluation [2] for both North Anna units just before it was implemented for N2C13 at its end of cycle [3]. The current safety evaluation for implementing the Tavg coastdown for N1C14 is based on the earlier safety evaluation [2]. Since all the technical issues have already been resolved, it is sufficient for this evaluation to summarize the results of the North Anna generic evaluation [2] and then concentrate on the cycle-specific issues for EOC implementation in N1C14.

In the generic evaluation [2] the impact of an EOC Tavg coastdown of up to 10 ^{0}F below the nominal full power Tavg of 580.8 ^{0}F was assessed - the results of that evaluation are summarized below. However, N1C14 operation will be limited to a Tavg reduction of 5 F at full power.
(a) All NSSS design basis accident analyses (LOCA & non-LOCA events, & containment integrity) continue to meet the applicable acceptance criteria for a 5 ^{0}F EOC Tavg coastdown, including the subcompartment analysis of the pressurizer cubicle [5]. When the Tavg reaches the target value of 575.8 ^{0}F at full power, a power coastdown shall be initiated. (b) A Westinghouse report [6] concluded that the NSSS systems and components will continue to meet their acceptance criteria for full power operation down to a Tavg of 570.8 ^{0}F (a 10 ^{0}F coastdown), under postulated normal, upset, emergency and faulted conditions. (c) A Mechanical Engineering evaluation [7] concluded that the balance of plant systems and components will have acceptable performance for a 10 ^{0}F coastdown.

In the safety evaluation [1] for N1C14 normal operation, the following reload parameters were found to be outside the range of the existing safety analysis assumptions: (i) the most negative Doppler power coefficient (DPC) for power below 11% rated power; (ii) the least negative MTC at EOC used in the rod ejection analysis; (iii) a peripheral assembly RPD used in PTS evaluation; and (iv) some values in the cycle-specific fuel rod F-H census. These parameters were shown to be accommodated by existing safety analysis margin and/or conservatism.

Evaluations of N1C14 for EOC Tavg coastdown [4] did not identify any additional reload parameters to be out of range, provided two of the analysis limits are changed for the Tavg coastdown phase: the FQ limit for LOCA-ECCS evaluation must be reduced from 2.19 to 2.15; and the hot assembly average RPD limit for LOCA from 1.45 to 1.37. There is no other impact of the temperature coastdown on the reload cycle parameters. All other conclusions of the previous evaluation for normal operation [1] remain valid.

The results of the current evaluation can be summarized as follows:

1. No increase in the probability of occurrence or consequences of an accident will result from the proposed Tavg coastdown operation. The proposed operation creates only incremental changes in the values of parameters previously shown to be significant in determining NSSS response to postulated design accidents. Since the currently applicable safety analyses remain bounding for North Anna Unit 1 Cycle 14, it is concluded that operation with the proposed reload core will neither increase the probability of occurrence nor the consequences of initiating events for any known accident.

2. The proposed temperature coastdown operation is being conducted by a procedurally controlled, gradual process. This process is similar to the power coastdown typically performed at the end of each cycle. The Tref rescaling is controlled in a manner that minimizes the risk of plant trip while ensuring that the effect on the rod control system will not create the possibility of an accident of a different type than any previously evaluated in the Safety Analysis Report.

3. The effects of the proposed operation upon NSSS components, systems and design basis accident response were accommodated within the conservatism of the assumptions used in the applicable safety analyses. These analyses have demonstrated that calculated results meet all design acceptance criteria as stated in the UFSAR. Therefore, the margin of safety is not reduced for the proposed Tavg coastdown operation of North Anna Unit 1 Cycle 14.

The conclusions stated above are based on the following assumptions:

1. Operation of North Anna 1 Cycle 14 at a measured shift average core power not exceeding 2893 MWt.
2. Cycle 14 burnup will not exceed 20,400 MWD/MTU, based on the actual EOC13 burnup of 17,951 MWD/MTU. This limit accommodates a temperature coastdown of 5 ^{0}F at full power below the normal Tavg of 580.8 ^{0}F, followed by a power coastdown.
3. Use of four once-burned FCF lead test assemblies introduced as fresh fuel in Cycle 13.
4. Use of short poison stack (127.2") BP rods.
5. Insertion of VSDAs into two-thirds (24 assemblies) of core peripheral fuel assemblies loaded next to the baffle.
6. Adherence to plant operating limitations stated in the Technical Specifications and the Core Operating Limits Report (Appendix A of RSE Technical Report NE-1167 Rev 2).
7. A maximum FQ of 2.19 during normal operation, but reduced to 2.15 for Tavg coastdown, as modified by K(z) is not exceeded.
8. A fully withdrawn RCCA position of 227 steps, a change of +2 steps from Cycle 13.
9. Maximum steam generator tube plugging fraction does not exceed 7% in any steam generator.

00-SE-OT-06

Description
Technical Requirements (TRM) Change Request Package #35 - Fire Protection Program Revisions
TRM Change #35 was developed to more accurately describe: a) Fire Suppression Water Systems, b) Fire Barriers, Penetration Seals, Fire Doors and/or Fire Dampers, c) Appendix-R Alternate Shutdown Equipment, d) Appendix-R Emergency Lighting, e) Passive Fire Protection Items, f) Fire Detection Instrumentation and their Bases.

Summary
The current North Anna (NAPS) License condition [2.D.3.(u)] allows Virginia Power to make changes to the Fire Protection Program without NRC approval, if those changes do not adversely affect the Station's ability to achieve and maintain safe shutdown in the event of a fire. The proposed changes to the Fire Protection Program have been determined to fall into this category since they pertain to corrections being made due to administrative errors, omissions, contradictions and clarifications to Action Conditions to more accurately describe activities affecting applicable Fire Suppression Water Systems, Fire Doors, Appendix-R Alternate Shutdown Equipment, Appendix-R Emergency Lighting, Passive Fire Protection Items, Fire Detection Instrumentation and Penetration Fire Barriers.

The proposed revisions to the NAPS Fire Protection Program contained in Technical Requirements (TRM) Change Request #35 will enhance the operability of fire protection systems since they will ensure that fire protection surveillance requirements, system descriptions and required action statements are consistent with Industry guidelines. They will incorporate equipment, clarify applicability and action conditions to ensure that the proper required actions / compensatory measures are implemented whenever equipment is found to be inoperable.

A review of the TRM Change Request #35 has determined that the change will not increase the probability, occurrence, or consequences of an accident since it does not involve the elimination of existing requirements or change the function or operability of the Fire Protection Systems. Similarly, these proposed changes will not affect the probability, occurrence or consequences of an accident or malfunction of equipment different from that considered in the Safety Analysis Report (SAR) or the station's compliance with 10 CFR 50, Appendix-A or General Design Criteria 3, since the change will have no effect on existing safety analyses assumptions or involve physical alteration of the plant or changes in methods governing normal plant operation. Additionally, the margin of safety discussed in the SAR will not be changed or adversely affected since the referenced change will not eliminate or alter the performance of any fire protection systems. Therefore, the proposed change will not result in a reduction in the margin of safety nor will it impact the ability of the Fire Protection System to function as designed and it can be concluded that an unreviewed safety question does not exist.

00-SE-OT-07

Description
UFSAR Change Request # FN 99-068
UFSAR Section 9.3.2.1.5 to be changed to state that secondary conductivity monitors may be recorded in the Control Room, at the On-Line Chemistry Monitoring System (OLCMS) panels, or in the Chemistry Lab versus are recorded in the Control Room.

Summary
Background:

Currently, the UFSAR states that secondary conductivity is monitored continuously. This is accomplished via the On-Line Chemistry Monitoring System (OLCMS) by Chemistry on the computer in the Chemistry Lab normally. Conductivity may also be monitored by recorders at the OLCMS panels or by Operations on the Control room recorders (1/2-SS-CR-100/200). The Control Room recorders are on but the information provided isn't used by Chemistry or Operations and they are a maintenance issue to maintain. Operations would like to be able to turn the Control Room recorders (1/2-SS-CR-100/200) off so that maintenance would be minimized, recorder paper would not be wasted and yet they would still be available for use if needed.

Major Issues considered:

Steam Generator Tube Rupture and Main Condenser Tube Leak

- This UFSAR change only clarifies where secondary plant conductivity may be monitored and where OLCMS alarms will occur to warn of abnormal conditions. This change has no effect on the probability or consequence of the accidents/malfunctions considered. Continuous secondary plant conductivity monitoring will continue to occur by Chemistry from the Chemistry Lab via the computer or from the OLCMS panels in the Turbine Building normally. No modifications of the plant will occur as a result of this change. For small changes in conductivity or if a small condenser tube leak were to develop, Chemistry would observe the adverse trends more readily because changes are more observable on the computer. For large condenser tube leaks, Operations has other indication such as loss of vacuum to alert them of the adverse condition.

Unreviewed Safety Question:

This UFSAR change does not result in any increase in probability or consequence to any previously evaluated SAR accident. This change neither creates nor results in a malfunction not previously evaluated. No modification is made to the plant by this change. NAPS tech specs will be complied with as written and will not be altered by this change. This change only clarifies UFSAR statements as to where secondary plant conductivity may be monitored and where OLCMS alarms will occur to warn of abnormal conditions. Based upon the above discussions and review, an unreviewed safety question does not exist and this UFSAR change should be permitted.

00-SE-OT-08

Description
North Anna UFSAR Change Request No. FN 99-042
UFSAR Change Request No. FN 99-042 contains a list of changes, some of which are editorial in nature, that need to be corrected or clarified in the UFSAR sections that discuss North Anna's Vital Bus Power (EV) System and Station Service (ESS) System. This package is a result of the Integrated Configuration Management Project review of North Anna Power Station's EV and ESS systems.

Summary
The changes are within the current design and licensing basis of the facility. The changes do not affect the initiators of analyzed events or the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, these proposed changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

The proposed editorial changes do not involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above proposed UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, these proposed changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

With regard to the criteria set forth in 10 CFR 50.59 and based on the evaluation summarized above, the proposed UFSAR changes have been determined not to represent an unreviewed safety question.

00-SE-OT-09

Description
UFSAR, Section 4.2.1.4.3
UFSAR Change Request Number FN 2000-001
UFSAR Section 4.2.1.4.3 describes new fuel and insert inspections and lists 8 requirements found in new fuel and insert receipt inspection procedures. The section title is being changed to "New Fuel and Insert Inspection" from "Onsite Inspection". The words "preshipment and" are being added to the first paragraph of the section. The word "receipt" is being deleted from the second paragraph. A note is being added to bullet 7. It reads "Note: Fuel assembly drag testing is either performed by the vendor, prior to shipment, or during postshipment inspections".

Summary
Historically, new fuel has been receipt inspected at the station in the new fuel receiving and storage area. On occasion, manufacturing defects have resulted in the return of an assembly. As a result of such an event, Nuclear Analysis and Fuel modified the fuel surveillance program by performing foreign material and manufacturing defect inspections at the manufacturer, just prior to shipment of the fuel to the station (preshipment). The full inspections were then conducted again at the station during new fuel receipt. Control rods and other inserts have been drag checked/tested during receipt inspection and fuel has been drag tested at the vendor.

UFSAR Section 4.2.1.4.3 describes new fuel and insert inspections and lists 8 requirements found in new fuel and insert receipt inspection procedures. The section title is being changed to "New Fuel and Insert Inspection" from "Onsite Inspection". The words "preshipment and" are being added to the first paragraph of the section. The word "receipt" is being deleted from the second paragraph. A note is being added to bullet 7. It reads "Note: Fuel assembly drag testing is either performed by the vendor, prior to shipment, or during postshipment inspections".

These proposed changes do not reduce the inspection scope or level for receipt inspection of new fuel and inserts.

The proposed changes to the UFSAR do not result in an unreviewed safety question as defined in 10 CFR 50.59 for the following reasons:

1) Changing the UFSAR so that the inspections may be done at either the manufacturing facility or at the station, neither increases the probability or consequences of accidents nor increases the probability or consequences of equipment malfunctions described in the SAR. Fuel handling is not impacted by this change. Fuel accountability integrity is not impacted by this change. Component identification verification requirements are not being altered. Fuel and insert drag testing continues to be specified. Performing the inspections at the manufacturing facility may improve the quality of the inspections since environmental (lights, etc.) conditions are better for performing inspections.

2) Changing the UFSAR to allow the fuel and insert inspections to be performed at the manufacturing facility or at the station neither increases the possibility for a different type of accident nor increases the possibility for a different type of equipment malfunction than have been previously described in the SAR. There are no changes to the required inspections, to the fuel design, or to the handling system.

3) The margin of safety as described in the Bases to the Technical Specifications is not reduced by this activity. There is no impact on operating parameters or any safety parameters.

00-SE-OT-10

Description
North Anna UFSAR Change Request No. FN 99-043
UFSAR Change Request No. FN 99-043 contains a list of changes, some of which are editorial in nature, which need to be corrected or clarified in the UFSAR sections that discuss the North Anna Power Station (NAPS) Boron Recovery/ Waste Disposal Systems. This package is a result of the Integrated Configuration Management Project review of NAPS's Boron Recovery/ Waste Disposal Systems.

Summary
The editorial/administrative changes are within the current design and licensing basis of the facility. These changes do not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed editorial/administrative changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

None of the proposed editorial changes involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial/administrative changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed editorial/administrative changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

With regard to the criteria set forth in 10 CFR 50.59 and based on the evaluation summarized above, the proposed editorial/administrative UFSAR changes have been determined not to represent an unreviewed safety question.

00-SE-OT-11

Description
UFSAR Change Request FN-99-074
0-GOP-7.1 Primary and Refueling Purification Filter Configuration Control (NEW PROCEDURE)
The UFSAR will be revised to delete detailed design information associated with the filters in the Chemical Volume and Control (CVCS) and Refueling Purification (RP) systems. A procedure is being generated to provide guidance and strategy for filter change-out.

Summary
As a result of the Enhanced Cobalt Filtration Program, different size (particle retention) filters are used for the filtration of primary and refueling purification filters than that described in the UFSAR. A filter change-out strategy was instituted that employed different filter sizes dependent on plant conditions and the amount of cleanup required. This safety evaluation supports a UFSAR change to remove the detail of filter design out of the UFSAR and to initiate a new General Operating Procedure to establish filter controls.

The initial filter upgrades were accomplished by the following:

1 / 2 CH-FL- 1, 2, & 5 – EWR 88-229 and associated SE 89-SE-MOD-015

1 / 2 CH-FL-3 & 4A,B – The filter retention sizes are the same as described in the UFSAR. REA 1996-001 and 002 were recently approved to upgrade these filters. The upgrade will be accomplished and controlled under current administrative means ensuring proper design basis is maintained. This upgrade is currently in the DCP process. Removal of the design detail from the UFSAR will not affect the design function of the filters.

1-RP-FL-1A, B – Initial filter change-outs were accomplished by EWR 88-033. A safety evaluation was not performed because the screening was checked no for affecting the FSAR design basis. However, the EWR provided justification for the replacement filters meeting the original design basis. The EWR notes that the filter elements were the manufacturers upgrade to outdated products. Historical performance has proven their ability to perform their cleanup function. The change-out from a 6 μm to a 20 μm filter in the RPIX inlet was done to increase the run time on the filter.

The proposed UFSAR change is to delete unnecessary detail provided on primary and RP filters. The purpose of the GOP is to provide guidance that is currently provided by verbal direction and memos. The changes are administrative in nature. The RP and CVCS filters currently used are of the same form, fit, and functions as the original. Historical performance of the filters has proven their effectiveness in general stream cleanup while maintaining design flows. Filter conditions are still monitored by radiation levels and delta pressures. Event precursors are not affected.

System flow requirements for the CVCS and RP systems remain the same. The change to the UFSAR deletes design detail for the primary and RP filters. This will update the UFSAR to document current approved operating practices regarding filters. Current operating practices provide for several filters to be installed in the reactor coolant and RPIX outlet filters and a reduction in filter sizes for other filters. The purpose of the enhanced cobalt filtration program was to reduce primary coolant activity, hot particle population and provide for less plate out/crud. Operation with the higher efficient Sub-Micron filters has been in effect for several years with no adverse effect on plant operations. The filters currently used at NAPS have the same form, fit, and function as the original filters, but with a higher efficiency than those described in the UFSAR. The CVCS and RP system operations were not affected by the change in filter size. Having higher efficiency filters in place has no affect on the consequences of chapter 15 accidents.

The proposed procedure does not change the current way in which we operate filters. The proposed procedure is being written to provide guidance that is currently provided by verbal direction and memos. The changes are administrative in nature and could not create the possibility for an accident of a different type than was previously evaluated.

The filters are monitored and changed out as necessary on high DP or radiation levels. The higher efficient filters have resulted in greater capability to remove suspended solids and ultimately reduced background radiation levels. The initial filter micron size reduction was accomplished by appropriate administrative means. Updating the UFSAR and creating a general procedure for filter change-out strategy does not increase the probability of occurrence or consequences of malfunctions of the CVCS system ability to deliver boric acid or the ability to purify the spent fuel pit.

The changes are administrative in nature and will not create the possibility for a malfunction of equipment of a different type than that previously evaluated in the UFSAR.

The TS basis and margin of safety for the boric acid system are not affected by the UFSAR change or proposed procedure. The changes are administrative in nature. The original change-out of the filters for both the CVCS and RP systems were performed under controlled administrative means.

The UFSAR update and creation of a new general operating procedure will not alter the overall operation of the CVCS and RP systems. An unreviewed safety question is not created by allowing the changes and therefore should be approved.

00-SE-OT-12

Description
UFSAR SAR Change Request Number FN-99-033
Revision of UFSAR Chapter 9, Sections 9.5.1.2.2.2, page 9.5-10 and 9.5 References, page 9.5-61

Summary
The North Anna UFSAR, Section 9.5.1.2.2.2, currently describes the fire suppression system in the Unit 1 and 2 Emergency Switchgear Rooms (ESGRs) as a total flooding Halon 1301 system that is manually actuated. The UFSAR also provides a description of the system design including controls and detection. This activity does not change that information but adds the historical and design basis for the system to be manually actuated.

This change to the UFSAR provides a discussion on why the manual Halon system installed in the ESGRs meets the requirements of a fixed fire protection system as required by Section III.G.3 of Appendix R to 10CFR50. Section III.G.3 requires fire detection and a fixed fire suppression system be installed in the area under consideration. Virginia Power installed manually actuated Halon fire suppression systems in the ESGRs by implementation of Design Change 83-36 as part of the modification made to comply with Appendix R. The system design was based on NFPA 12A and, in accordance with paragraph 1-8.1.1 of NFPA 12A, the system can be manually actuated if acceptable to the authority having jurisdiction. Based on the discussion in Generic Letter (GL) 83-33 the Halon system installed in the ESGRs meets the requirements of a fixed fire protection system as required by III.G.3. The installation of a manual Halon system is in accordance with paragraph 1-8.1.1 of NFPA 12A, 1980 edition.

The UFSAR changes have been evaluated against the existing Appendix R requirements and the station's Fire Protection Program. It has been established that a) the level of fire protection for the station is not being diminished, i.e. defense in depth is maintained, and b) this change will not affect the capacity to achieve and maintain safe shutdown in the event of a fire.

This change does not increase the probability, the consequences or create new accidents not previously analyzed because it does not alter the design, function, ability to function, or method of performing the function as currently described in the UFSAR for the Halon system. Neither does this change increase the probability of the occurrence of, the consequences of, or possibility of a new malfunction, because this change does not diminish the level of fire protection for the ESGRs as currently described in the UFSAR.

The margin of safety has not been reduced since the change demonstrates that the plant maintains an adequate Appendix R design basis and the fire protection program remains intact. No changes are required to the Technical specifications and the changes will not adversely affect the capability to achieve and maintain safe shutdown in the event of a fire.

00-SE-OT-13

Description
QA Topical/UFSAR Chapter 17 Change FN 2000-04
UFSAR Chapter 17.2 Operational Quality Assurance Program - Tables 17.2-0 17.2-2 and 17.2-3
Redefine records retention requirements for operating phase records
Clarify the definition of "QA Record"
Define "Lifetime" as a record retention period

Summary
PROPOSED CHANGES:

- Redefine records retention requirements for operating phase records
- Clarify the definition of "QA Record"
- Establishes and defines "Lifetime" as a record retention period

This Operational QA Program change does not affect the operation or design of the plant or any system, structure or component. No accident analysis assumptions are modified or challenged by this change. Plant equipment will not be operated in a different manner. This change is administrative in nature and redefines the records retention requirements, clarifies the definition of a "QA Record," and establishes "Lifetime" as a record retention period.

Therefore, this proposed Operational QA Program change will not:

- Increase the probability of occurrence or consequences on any accident or malfunction of equipment important to safety previously analyzed in the SAR
- Create an accident or malfunction of equipment of a different type than was previously evaluated in the SAR
- Reduce the margin of safety as defined in any Technical Specification Basis

However, NRC approval is required due to the reduction in commitment per 10 CFR 50.54(a).

00-SE-OT-13, Rev. 1

Description
QA Topical/UFSAR Chapter 17 Change FN 2000-04
UFSAR Chapter 17.2 Operational Quality Assurance Program - Tables 17.2-0, 17.2-2 and 17.2-3
Redefine records retention requirements for operating phase records
Clarify the definition of "QA Record"
Define "Lifetime" as a record retention period
Incorporate NRC comments

Summary
PROPOSED CHANGES:

- Redefine records retention requirements for operating phase records
- Clarify the definition of "QA Record"
- Establishes and defines "Lifetime" as a record retention period

This Operational QA Program change does not affect the operation or design of the plant or any system, structure or component. No accident analysis assumptions are modified or challenged by this change. Plant equipment will not be operated in a different manner. This change is administrative in nature and redefines the records retention requirements, clarifies the definition of a "QA Record," and establishes "Lifetime" as a record retention period.

Therefore, this proposed Operational QA Program change will not:

- Increase the probability of occurrence or consequences on any accident or malfunction of equipment important to safety previously analyzed in the SAR
- Create an accident or malfunction of equipment of a different type than was previously evaluated in the SAR
- Reduce the margin of safety as defined in any Technical Specification Basis

However, NRC approval is required due to the reduction in commitment per 10 CFR 50.54(a).

REV. 1
Incorporate NRC review comments. The original 50.59 remains bounding.
- Include appropriate definition of lifetime on Table
- Revise to clarify training material retention requirements
- Revise 3 years to 3 cycle for requalification record

00-SE-OT-14

Description

North Anna UFSAR Change Request No. FN 2000-003

UFSAR Change Request No. FN 2000-003 contains a list of changes which need to be corrected or clarified in the UFSAR sections that discuss North Anna's Instrument Air System. This package is a result of the Integrated Configuration Management Project review of North Anna Power Station's Instrument Air System.

Summary

The editorial change affects neither the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems or components. The change does not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. There is no impact on the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed change does not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

The proposed editorial change does not involve a physical alteration of the plant, or a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the editorial change to the UFSAR does not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. This UFSAR change does not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the editorial changes do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

Based upon this evaluation regarding the criteria set forth in 10 CFR 50.59, the above listed proposed editorial UFSAR change has been determined not to represent any unreviewed safety question.

00-SE-OT-15

Description

Technical Requirements Change Request No. 38

Changes will convert the existing Technical Requirements Manual (TRM) Microsoft Word documents into a FrameMaker document that will be controlled and maintained by the Configuration Management project.

Summary

The TRM was developed and implemented in 1994 to consolidate the technical requirements used by station and operations personnel in determining compliance with the various requirements that govern the safe operation of the plant. The NRC has approved the TRM concept and has classified the TRM as an approved station-controlled document in their Safety Evaluation by the Office of Nuclear Reactor Regulation (Related to NRC Issued Technical Specification Amendments 187 and 168 – Relocation of the RTS and the ESFAS Response Times to Controlled Documents). Changes to the TRM are controlled similar to those made to the Technical Specifications.

Section 4.8 of the Station Licensing 1st Quarter Self Assessment Report recommended that the Technical Requirements Manual be converted from the existing Microsoft Word documents to a FrameMaker document. FrameMaker is currently being utilized by Virginia Power for design and licensing basis documents such as the UFSAR, Current and Improved Technical Specifications, VPAPs, Design Bases Documents, ISFSI Technical Specifications and Safety Analysis Reports, and the Surry TRM. The proposed changes will convert the existing Microsoft Word documents into a FrameMaker document controlled by the Configuration Management Project located at the Innsbrook Technical Center.

TRM files are currently maintained by the Station TRM Coordinator with a separate Microsoft Word document for each TRM page. Upon the preparation and approval of a TRM Change Request Package, changes are then controlled, typed, and processed by the TRM Coordinator. This includes the preparation of each new/revised page as a Microsoft Word document for formal distribution by Station Records as well as preparing and posting these changes again in the Virginia Power Technical Specification System (VPTSS). This duplication, control, and maintenance of the TRM is considered burdensome, outdated, very time consuming, and not cost effective. FrameMaker has been selected by the Virginia Power Configuration Management for the preparation and maintenance of our design and licensing documents. This is based upon its excellent ability to handle large structured documents, the ability to accommodate massive changes without destroying the inherent structure of the document, and its ability to support production of multiple media output from a common set of source files. Additionally, FrameMaker readily supports the capability to publish the TRM directly to the MIND system.

SAFETY SIGNIFICANCE

The TRM along with the Technical Specifications provide an easy reference for plant support and operations personnel to use in order to make technically accurate decisions concerning the various requirements that govern the safe operation of the station. The administrative changes to convert the TRM to a FrameMaker document from the many Microsoft Word documents do not alter the existing Technical Requirements, Action Statements, or Surveillance Requirements.

These administrative changes have been reviewed against the criteria of 10 CFR 50.59 and concluded that the changes do not pose an unreviewed safety question. Specifically, operation of the North Anna Power Station with these changes to the TRM will not:

1. Increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report
   Plant systems and components will not be operated in a different manner as a result of these administrative changes. Changes will not require any modifications to plant hardware or operating practices. Therefore, the possibility for an accident of a different type than was previously evaluated in the Safety Analysis Report is not created by the changes.

2. Create the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report
Plant systems and components will not create any new accident precursors or methods of operation as a result of these administrative changes. There are no modifications to plant equipment or systems and there is no direct effect on plant operation. Therefore, the possibility for an accident of a different type than was previously evaluated in the safety Analysis Report is not created by the change.
3. Reduce the margin of safety as defined in the bases for any technical specification. Plant systems and components will not be operated in a different manner. Therefore, the accident analysis assumptions for design basis accidents are unaffected and the margin of safety is not decreased by the changes.

00-SE-OT-16

Description
North Anna UFSAR Change Request No. FN 99-067
UFSAR Change Request No. FN 99- 067 contains a list of changes, some of which are editorial in nature, which need to be corrected or clarified in the UFSAR sections that discuss North Anna's fuel handling and storage systems. This package is a result of the Integrated Configuration Management Project review of North Anna Power Station's fuel handling and storage
systems.

Summary
The editorial/administrative changes are within the current design and licensing basis of the facility. These changes do not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed editorial/administrative changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

None of the proposed editorial changes involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial/administrative changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed editorial/administrative changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

With regard to the criteria set forth in 10 CFR 50.59 and based on the evaluation summarized above, the proposed editorial/administrative UFSAR changes have been determined not to represent an unreviewed safety question.

00-SE-OT-17

Description

Chemistry Special Order No. 00-002

Chemistry Special Order No. 00-002 is being proposed to use Calgon Biocide - EVAC in the Bearing Cooling System.

EVAC will be applied directly to the cold-water basin of the Bearing Cooling tower by Chemistry or the vendor, Calgon. The Chemistry Special order will control the location, quantity, duration and frequency of the application per the recommendations of the Chemistry Supervisor. It is projected that EVAC will be used on a semiannual basis. Per VPAP-3001, this evaluation is required as it represents a modification to the chemical treatment program of the BC system.

Summary

UFSAR Chapter 15.2.8 – Loss of Normal Feedwater: The loss of Bearing Cooling could result in a Main Feedwater pump trip or failure because the BC system provides pump seal-oil cooling.

It is unlikely that this interim use of EVAC in the Bearing Cooling tower would result in the loss of Bearing Cooling; however, in the event that Bearing Cooling is not available no safety systems would be impacted as the Auxiliary Feedwater system would be used. The Bearing Cooling system does not communicate with the Auxiliary Feedwater system.

UFSAR Chapter 6.4.1 – Habitability Systems: for the Control Room to ensure that continuous occupancy of the areas is possible for the events described in Chapter 3 as well as all of the postulated accidents discussed in Chapter 15.

The use of EVAC will not impact the Control Room Habitability analysis as it will be stored in warehouse #7, which is outside of the control room habitability inclusion area. When required, EVAC will be transported to the Bearing Cooling Basin in 5-gallon quantities (less than 100 lbs). The chemical will be handled and applied by trained chemical technicians.

Since the Bearing Cooling system does not communicate with the Auxiliary Feedwater system and EVAC will not be stored in the inclusion area, the consequences of the accidents identified above will not be increased. Also, The Bearing Cooling system does not communicate directly with any safety systems or components. Thus, it is unlikely that the use of EVAC in the Bearing Cooling system would create an accident that has not been previously evaluated in the SAR. The operability of the Bearing Cooling system will be enhanced by this activity.

Thus, it can be concluded that no unreviewed safety question exists.

00-SE-OT-18

Description

Reload Safety Evaluation Technical Report NE-1229 Rev 0

Refueling and operation of North Anna Unit 1 Cycle 15. Incorporation of the following features described in Technical Report NE-1229 Rev 0:

1. Four twice-burned lead test assemblies (LTAs) fabricated by Framatome Cogema Fuels (FCF) will be re-inserted into the core for a third cycle of irradiation.
2. Use of the short poison stack (127.2") BP rods as in Cycle 14.
3. 24 of the peripheral assemblies will have replacement top nozzles and vibration suppression damping assemblies.

Summary

A safety evaluation has been performed to determine whether an unreviewed safety question will result from the refueling and operation of North Anna Unit 1 Cycle 15. In this evaluation, reload cycle parameters have been calculated and compared to the existing safety analysis assumptions. These parameters have been shown to be either explicitly bounded, or accommodated by existing safety analysis margin and/or conservatism.

The impact of the following features and assumptions have been accounted for in the appropriate evaluations performed for N1C15:

1. Cycle 15 burnup will not exceed 20,500 MWD/MTU (EOC14 = 19,400 MWD/MTU), or 20,000 MWD/MTU (EOC14 = 20,400 MWD/MTU). These limits include up to a 5 ^{0}F Tavg coastdown at full power, followed by a customary power coastdown for a total coastdown of approximately 2500 MWD/MTU, past the normal Tavg full power end of reactivity. Tavg coastdown operation was approved for both North Anna units by NAPS Safety Evaluation No. 99-SE-OT-26 Rev 1, 08/05/99; and has already been implemented in N1C14 [Safety Evaluation No. 00-SE-OT-05, 01/06/2000]. The maximum Tavg reduction is limited to the value specified in the cycle-specific reload safety evaluation. N1C15 is limited to a 5 ^{0}F coastdown [NE-1229]
2. Use of short (127.2") poison stack BP rods.
3. Twenty-four of the peripheral assemblies to have replacement removable top nozzles (RTNs) and vibration suppression damping assemblies (VSDAs).
4. A maximum FQ of 2.19 during normal operation, but reduced to 2.15 for the EOC Tavg and power coastdown, modified by K(z), as presented in Appendix A of Technical Report NE-1229.
5. A fully withdrawn RCCA position of 229 steps.
6. Maximum steam generator tube plugging fraction does not exceed 7% in any steam generator.

Short (127.2") poison stack BP rods, VSDAs, and Tavg coastdown operation have been implemented in the previous cycle, N1C14. The other assumptions are also consistent with those in previous cycles.

Two of the reload parameters were found to be outside the range of the generic safety analysis input assumptions, and therefore required specific evaluation. In accordance with the Topical Report VEP-FRD-42 Rev 1-A, "Reload Nuclear Design Methodology," an evaluation was performed to determine the impact of these parameters on the currently applicable safety analyses, as described below.

I. The reload cycle fuel rod F–H census is not bounded by the reference limit for all values. Based on the known DNBR sensitivity to F–H in a thermal hydraulic evaluation, a penalty has been assessed against retained DNBR margin to accommodate the unbounded values in the census.

II. The hot assembly average relative power density (RPD) slightly exceeds the limit for the Tavg coastdown. The large break LOCA analysis was reviewed to assess the impact of the unbounded RPD during the coastdown. It is concluded that the hot assembly RPD can be accommodated without a penalty, based on the use of a reduced FQ of 2.15 and up to a 5 ^{0}F coastdown.

The results of this evaluation can be summarized as follows:

1. No increase in the probability of occurrence or consequences of an accident will result from this core reload. The reload creates only incremental changes in the values of parameters previously shown to be significant in determining core response to known accidents. Since the currently applicable safety analyses remain bounding for North Anna Unit 1 Cycle 15, it is concluded that operation with the proposed reload core will neither increase the probability of occurrence nor the consequences of initiating events for any known accident.

2. The N1C15 reload includes four FCF lead test assemblies reinserted for a third cycle of operation, 127.2" short stack BP, and VSDAs used for third and fourth cycles of operation. It has been determined that use of these fuel assemblies and insert components does not result in the safety limits being exceeded for any accident. Further, the effect on system operation and accident response is fully described by the parameters evaluated. Therefore, operation of this core does not create the possibility of an accident of a different type than any previously evaluated in the Safety Analysis Report.

3. The margin of safety is not reduced. The effects of core parameter variations were accommodated within the conservatism of the assumptions used in the applicable safety analyses. These analyses have demonstrated that calculated results meet all design acceptance criteria as stated in the UFSAR.

00-SE-OT-19

Description

North Anna UFSAR Change Request No. FN 99-072

UFSAR Change Request No. FN 99-072 contains a list of changes, some of which are editorial in nature, which need to be corrected or clarified in the UFSAR sections that discuss North Anna's high energy lines. This package is a result of the Integrated Configuration Management Project review of North Anna Power Station's high energy line break report.

Summary

The editorial changes are within the current design and licensing basis of the facility. These changes do not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed editorial changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

None of the proposed editorial changes involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed editorial changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

With regard to the criteria set forth in 10 CFR 50.59 and based on the evaluation summarized above, the proposed editorial UFSAR changes have been determined not to represent an unreviewed safety question.

00-SE-OT-20

Description
North Anna UFSAR Change Request No. FN 2000-002
UFSAR Change Request No. FN 2000-002 contains a list of changes, some of which are editorial in nature, which need to be corrected or clarified in the UFSAR sections that discuss North Anna's Electrical and Communication Systems. This package is a result of the Integrated Configuration Management Project review of North Anna Power Station's Electrical and Communication systems.

Summary
The above editorial/administrative changes are within the current design and licensing basis of the facility. These changes do not affect the initiators of analyzed events or the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed editorial/administrative changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

None of the proposed editorial changes involves a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial/administrative changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed editorial/administrative changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

With regard to the criteria set forth in 10 CFR 50.59 and based on the evaluation summarized above, the proposed editorial/administrative UFSAR changes have been determined not to represent an unreviewed safety question.

00-SE-OT-21

Description

UFSAR Change Request FN 2000-007

Change UFSAR Section 5.5 to state that the RCP thermal barrier may not provide sufficient cooling to maintain pump bearing or seal temperatures less than operating limits during a loss of seal injection if initial pump #1 seal leakoff flow is less than 2.5 gpm.

Summary

Until recently, it was assumed that an RCP could operate without seal injection for a relatively long period of time, such as 24 hours, because the thermal barrier heat exchanger would cool the system water to an acceptable level. The temperature in the bearing/seal annulus was expected to increase but then stabilize at a value less than the operating limit. It has now been postulated by Westinghouse (NSAL 99-005) that during a loss of seal injection with initial #1 seal leakage of less than 2.5 gpm, the bearing and seal operating temperature may be exceeded within 1 to 2 hours. Even though the RCS water is cooled in the TBHX, re-heating of the water could occur as it slowly flows up along the pump shaft.

The UFSAR will be revised to remove the implication that on a loss of seal injection the reactor coolant pump can operate indefinitely. Significant equipment damage of the RCP seals and leak-off line is not expected and the No. 1 seal is expected to control leakage as designed. Loss of seal injection Annunciator Response procedures (1&2-AR-C-G6) alert operators that the seal and bearing temperatures will likely exceed operating limits within two hours if initial seal injection flow is at or below 2.5 gpm. Appropriate actions for responding to this event are provided. Because plenty of time is available in the transient to allow for operator response, and because the higher temperatures are not expected to result in seal failure, there is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report. Because the unlikely event of seal failure is bounded by the loss of coolant analysis, there is no possibility of creating an accident or malfunction of a different type than previously evaluated in the safety analysis report. Since no analyses or setpoints are affected, the margin of safety as defined in the basis for any technical specification is not reduced.

00-SE-OT-22

Description
This Safety Evaluation was performed in response to Plant Issue N-2000-0490-E2. The Plant Issue detailed assignment requested:

1) An evaluation of the administrative controls in place to ensure that the Control Room Bottled Air Pressurization System is currently capable of meeting its design basis function, and
2) Engineering input to Licensing for any Technical Specification changes that are required following completion of the Engineering Evaluation to ensure that the Technical Specifications are consistent with the system's current design basis.

Technical Specification Change Request Package No. 378
To evaluate the current ability of the Control Room Bottled Air Pressurization System to meet its design basis as required by NRC Generic Letter 91-18 for non-conservative Technical Specifications, and to revise Technical Specifications Surveillance Requirement 4.7.7.2.a to increase the minimum number of compressed air bottles from 84 to 102 bottles based upon the results of Engineering Study 87-08 to maintain its design function.

Summary
This activity documents the ability of the current Control Room Bottled Air Pressurization System to meet its intended design function and supports Tech Spec Change Request No. 378. The proposed Tech Spec changes will increase the minimum number of compressed air bottles from 84 bottles to 102 bottles in Surveillance Requirement (SR) 4.7.7.2.a to incorporate the results of Engineering Study 87-08, "Evaluation of Control Room Bottled Air Test." The minimum capacity for the control room bottled air system was evaluated by this study and concluded that the Control Room Bottled Air System was not adequate to consistently satisfy the acceptance criteria in 1-PT-76.4, "Control Room Bottled Air Pressurization System Test." The acceptance criteria includes: 1) maintaining a flow of 340 scfm for 60 minutes, and 2) maintaining a differential pressure of 0.05 inches of water between the Control Room and the surrounding areas for at least 60 minutes. 1-PT-76.4 was completed satisfactorily on 8/2/87 (with a 1.7% margin), but was completed unsatisfactorily on 7/31/87 (flow was unsatisfactory at 58 minutes). EWR 87-528 was generated to add 9 additional bottles to each of the original designed banks to ensure that the system was consistently capable of performing its design basis function. This function is to minimize the radiation exposure of the Operators in the Control Room, and therefore maintain the exposures in accordance with General Design Criteria 19 and 10CFR100. This activity could not conceivably affect the environment, and has no affect on the ability to shut down the plant following a fire. The value of 84 in the current revision of the Technical Specifications will be changed to 102 to reflect the modifications made by EWR 87-528, the current design capacity of the system and to ensure consistency with the implementing procedures.

No physical changes are being made to the plant, and no changes are made to the method of plant operations. This engineering evaluation and Tech Spec changes will not affect the probability or consequences of any accidents considered in Section 68 of this Safety Evaluation (Main Steam Line Break, Steam Generator Tube Rupture, Locked Rotor Accident, LOCA and Fuel Handling Accident). Also, this engineering evaluation and Tech Spec changes will not affect the consequences or probability of any malfunction considered in Section 69 of this Safety Evaluation (Electrical Separation, Single Failure Criteria, Environmental Qualification and Seismic Qualification). In addition, no new type of accident or malfunction is created by this activity, as no physical equipment is affected by this activity. This activity does not affect the margin of safety as described in the Bases section of the Technical Specifications.

The safety analysis assumes that the Control Room Bottled Air Pressurization System is able to meet it's design basis requirement of maintaining 0.05" water column differential pressure between the Control Room and the surrounding areas, at a flow rate of 340 standard cubic feet per minute. This is required for one hour after the start of the accident. The EWR (#87-528, written in response to Engineering Study 87-08) added 9 more bottles to each bank of bottled air (18 per unit). The bottle addition was done to ensure that sufficient margin for the design basis requirement to provide at least 340 standard cubic feet per minute at the end of one hour could be provided. The implementing document (0-PT-76.3) currently ensures that 102 total bottles (84 original bottles + 18 additional bottles) are verified to be in place per unit. This

Periodic Test (PT) is performed monthly while the unit is in Modes 1-4. However, the Technical Specifications were not updated to reflect the design capacity. The system is currently functionally tested in accordance with 0-PT-76.4, "Control Room Bottled Air Pressurization System Test." The existence of these two Periodic Tests provides sufficient measures to ensure that the design basis is maintained, as required by the safety analysis.

00-SE-OT-23

Description

Topical Report Change Request package to the Topical Report section 17.2.17 (for Surry and North Anna) Revise our Topical Report Section 17.2.17 to allow for the retention of quality records in various electronic formats recommended by the Nuclear Information and Records Management Association (NIRMA) technical guidelines and endorsed by the NRC.

Summary

This proposed Topical Report Change only allow for alternative collection and storage of quality assurance records necessary to meet 10 CFR 50 Appendix B criterion 17. Instead of paper copies that are eventually filmed for archival storage this Topical Report change acknowledges the significant advancement in technology made in recent years such that many of these records are now created in an electronic format. Long term archival storage of these quality records in their native electronic format on an approved storage media with a media lifetime meeting or exceeding the records retention requirements is a logical extension of our Records Management process. This Topical Report change is necessary to take advantage of this quality record storage option.

The major issues considered are:

Does the nature of the quality record support the need for electronic storage?

Does the record need to be frequently accessed such that on-line storage would be an advantage?

Is the record created in an electronic format that is not application specific? (i.e. Not Proprietary)

Does the current automation process support transfer of the record to an approved electronic storage retention system separate from the production system?

Does the designated storage media lifetime meet or exceed the record retention requirements for the quality record?

Do our systems ensure that electronic identities meet the same authority level as manual handwritten signatures?

Almost all our documents are now produced electronically, but this does not automatically necessitate that they all be stored electronically.

Our quality assurance records are evidence of activities or events that have content, structure, and context to accurately convey to the reader the events at the time they happened. Our electronic storage systems must support the collection storage, retrieval and maintenance of these records for the duration specified in the applicable record retention schedule.

The NIRMA guidelines, pertaining to the control of Electronic Records are listed here:

TG 15 - 1998 'Management of Electronic Documents'

TG 13 - 1986 'Records Turnover'

TG 11 - 1998 'Authentication of Records & Media'

TG 16 - 1998 'Software Configuration Management and Quality Assurance"

TG 21 - 1998 'Electronic Records Protection and Restoration'

The NRC is expected to endorse these documents in a soon to be issued generic letter as the basis for an acceptable method to create and store Quality Records that meet the requirements of 10 CFR 50 Appendix B Criterion 17. This new generic letter is expected to clarify the original guidance provided in GL 88-18.

Therefore, since this proposed Topical Report Change only changes the method by which quality assurance records are stored and the fact that these quality records represent only historical evidence of events, it does not introduce any unreviewed safety question.

00-SE-OT-24

Description
TN-32 Dry Storage Cask Topical Safety Analysis Report (TSAR), Revision 9A
North Anna ISFSI SAR
The radiograph for the flange to inner shell weld of TN-32 Cask 21 to be used at North Anna does not meet the inspection requirements of Section V Article 2 Paragraph T-282.2 of the ASME Code. The code requires that the density variations of the film be between –15% and +30%. Due to misalignment between the source and the weld, the worst case density variation is –17.2%. However, the film meets the density limitation requirements of 1.8 to 4.0 in all areas. Two NDE Level III inspectors have examined the films and deemed the weld structurally acceptable.

Summary
This is a 10 CFR 72.48 safety evaluation.

Section 7.1.1 of the TN-32 TSAR Rev. 9A states, "Paragraph NB-4300 is applied for all confinement vessel welds and examination and acceptance meets the requirements of NB-5210, NB-5220, NB-5320, NB5330, NB-5340, or NB-5350 as appropriate." Paragraph NB-5210 applies to the flange to inner shell weld and requires the weld to be examined by radiograph. The radiographic method is detailed in Section V Article 2 of the ASME Code. As described in SNCR 755, the radiograph for the flange to inner shell weld of TN-32 Cask 21 to be used at North Anna does not meet the inspection requirements of paragraph T-282.2 contained in Article 2. The paragraph requires that the density variations of the film be between – 15% and +30% relative to the density adjacent to the wire. Due to misalignment between the source and the weld, two areas less than an inch wide fall just outside the lower bound of –15%. The worst variance is –17.2%. However, the film meets the density limitation requirements of 1.8 to 4.0 in all areas. The variance deviation was overlooked by the RT Level II examiner that initially accepted the film. Consequently, the inner and shield shells were assembled and the weld is now inaccessible for additional NDE. Precision Components Corporation's (PCC) NDE Level III Inspector and Virginia Power's corporate NDE Level III Inspector have examined the films and deemed the weld structurally acceptable. They base their conclusions on the following:

1. The pre-heat treat radiographs meet the variation requirements and show no indications in the areas of concern.
2. The post-heat treat radiographs are clear and readability is good. Using the radiographs from adjacent zones, the required Image Quality Indicator (IQI) is readily seen at the density levels of the discrepant areas demonstrating the acceptability of the set-up to provide adequate sensitivity. In fact, the sensitivity of the exposures exceeds the specification requirements. Per the ASME Code the 0.040" wire must be visible. On all radiographs for this weld the 0.032" wire was visible.
3. The post-heat treat radiographs show no indications in the areas of concern.
4. The film meets the density limitation requirements of 1.8 to 4.0 in all areas.

This deviation is therefore only to the inspection process for the weld, not to the quality of the weld.

An unreviewed safety question does not exist for the following reasons:

1. The change will not increase the consequences or probability of accidents evaluated in the UFSAR. Accidents that were reviewed are included in North Anna ISFSI SAR Sections A.1.5, "Cask Sliding and Tip Over Accidents"; 8.2.9, "SSSC Drops"; 8.2.10, "Loss of Confinement Barrier"; and TN-32 TSAR sections 11.2.8, "Hypothetical Cask Drop and Tipping Accidents"; and 11.2.9, "Loss of Confinement Barrier." All code requirements relating to the quality of the weld continue to be met. The intent of Section V Article 2 is to ensure that the radiograph is of sufficient quality to reveal any defects in the weld. Though film density variance relative to the area adjacent to the IQI wire was slightly outside the code requirement in two areas, the NDE Level III inspectors were able to determine by other means that 1) the quality of the radiographs met or exceeded the standard intended by the code and 2) that the weld is structurally acceptable. Therefore, the cask confinement boundary is not affected and the probability of occurrence of any analyzed accident will not be increased. Since

the change has no detrimental effect on the cask's structural integrity, there will be no increase in the consequences of an accident.

2. The change will not create the possibility for an accident of a different type than was analyzed in the UFSAR. The inspection deviation has no detrimental effect on the cask structural, thermal, criticality, or shielding analyses. Therefore, no parameters are affected that could form a precursor to another accident scenario.
3. The change does not increase the consequences or probability of malfunctions of equipment related to safety evaluated in the UFSAR. The malfunctions that were considered are those that compromise the cask's ability to maintain its confinement boundary and structural integrity in addition to its ability to be safely handled, transfer heat, maintain subcritical margin of stored fuel, and provide shielding. The inspection deviation will not affect the cask confinement boundary, structural integrity, thermal performance, criticality control, or shielding evaluation. The consequences of a malfunction of equipment identified above are the release of radioactive material to the environment. These consequences are evaluated in the North Anna ISFSI SAR. Since the North Anna ISFSI SAR assumes, for the loss of confinement accident, that all fuel stored in the cask fails, including cladding and fuel pellets, there is no malfunction which would produce an increase in consequences.
4. The change will not create the possibility for malfunction of equipment of a different type than was analyzed in the UFSAR. Two NDE Level III inspectors have examined the films and deemed the flange to inner shell weld acceptable. Therefore, the inspection deviation does not affect the cask confinement boundary or structural integrity and no parameters are affected that could form a precursor to another accident scenario.

00-SE-OT-25

Description
North Anna UFSAR Change Request #FN 99-062
UFSAR Change Request No. FN 99-062 contains a list of 43 proposed changes, some of which are editorial in nature, that need to be corrected or clarified in the UFSAR sections that discuss North Anna's FP System. The UFSAR change package is the result of the Integrated Configuration Management Project review of the North Anna Power Station Fire Protection System.

Summary
The editorial/administrative changes are within the current design and licensing basis of the facility. These changes do not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed editorial/administrative changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

None of the proposed editorial changes involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial/administrative changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed editorial/administrative changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

With regard to the criteria set forth in 10 CFR 50.59 and based on the evaluation summarized above, the proposed editorial/administrative UFSAR changes have been determined not to represent an unreviewed safety question.

00-SE-OT-26

Description

North Anna UFSAR Change Request No. FN 99-069

UFSAR Change Request No. FN 99-069 contains a list of changes, some of which are editorial in nature, which need to be corrected or clarified in the UFSAR sections that discuss North Anna's fuel pit cooling, chilled water, turbine generator, and bearing cooling water systems, general plant descriptions and classifications, and compliance with safety guides. This package is a result of the Integrated Configuration Management Project team review of North Anna Power Station.

Summary

The editorial/administrative changes are within the current design and licensing basis of the facility. These changes do not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed editorial/administrative changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

None of the proposed editorial changes involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial/administrative changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed editorial/administrative changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

With regard to the criteria set forth in 10 CFR 50.59 and based on the evaluation summarized above, the proposed editorial/administrative UFSAR changes have been determined not to represent an unreviewed safety question.

00-SE-OT-27

Description
North Anna UFSAR Change Request No. FN 2000-019
UFSAR Change Request No. FN 2000-019 contains a list of changes which need to be corrected or clarified in the UFSAR sections that discuss North Anna's Electrical-Instrumentation and Plant Computer System (EI System). This package is a result of the Integrated Configuration Management Project review of North Anna Power Station's EI system.

Summary
The editorial/administrative changes are within the current design and licensing basis of the facility. These changes do not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed editorial/administrative changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

None of the proposed editorial changes involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial/administrative changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed editorial/administrative changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

00-SE-OT-28

Description

North Anna UFSAR Change Request No. FN 99-044

UFSAR Change Request No. FN 99-044 contains a list of changes, some of which are editorial in nature, which need to be corrected or clarified in the UFSAR sections that discuss the North Anna Power Station (NAPS) ventilation system. This package is a result of the Integrated Configuration Management Project review of NAPS's ventilation system.

Summary

The editorial/administrative changes are within the current design and licensing basis of the facility. These changes do not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed editorial/administrative changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

None of the proposed editorial changes involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial/administrative changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed editorial/administrative changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

With regard to the criteria set forth in 10 CFR 50.59 and based on the evaluation summarized above, the proposed editorial/administrative UFSAR changes have been determined not to represent an unreviewed safety question.

00-SE-OT-29

Description
Technical Specification Change Request No. 376
UFSAR Change Request FN 2000-016
A revised analysis basis is proposed for the existing North Anna Units 1 and 2 Technical Specification Reactor Coolant System (RCS) Pressure/Temperature (P/T) Operating Limits, Low Temperature Overpressure Protection System (LTOPS) Setpoints, and LTOPS Enable Temperature. The proposed revised analysis basis extends the cumulative core burnup applicability limits (EFPY) for these Technical Specification LCOs, and utilizes ASME Section XI Code Cases that have not been adopted into 10 CFR 50 Appendix G in accordance with 10 CFR 50.55a. Therefore, the proposed revised analysis basis requires NRC review and approval.

Summary
Following the analysis of North Anna Unit 1 capsule W [1], Virginia Power performed a detailed evaluation of available reactor vessel materials surveillance data [11]. This information was transmitted to the NRC by Reference [2]. As documented in Reference [2], the PTS screening calculation results for North Anna Units 1 and 2 were determined to meet the applicable screening criteria for cumulative core burnups up to 32.3 Effective Full Power Years (EFPY) and 34.3 EFPY (corresponding to end-of-license) for Units 1 and 2, respectively. However, the cumulative core burnup limit for the current North Anna Unit 1 RCS P/T limits, LTOPS setpoints, and LTOPS enable temperature (T_{enable}) values documented in the North Anna Technical Specifications was determined to be no longer valid. Specifically, the newly acquired North Anna Unit 1 surveillance data caused the cumulative core burnup limit for the Unit 1 RCS P/T limit curves to be reduced from 30.7 EFPY to 17.2 EFPY. (The cumulative core burnup limit of 17.0 EFPY for the currently applicable North Anna Unit 2 P/T limit curves was determined to remain valid.) North Anna Unit 1 is predicted to achieve 17.2 EFPY in May 2001. Therefore, Virginia Power committed to provide a licensing submittal with revised North Anna Unit 1 Technical Specification P/T limits, LTOPS setpoints, and T_{enable} value by June 30, 2000.

Two sources of analytical margin are available to support extension of the cumulative core burnup applicability limits of the existing P/T limits and LTOPS setpoints. These two sources of margin are (a) ASME Section XI Code Case N-640 [6], which supports use of the K1c fracture toughness curve, instead of the K1a curve employed in the development of the existing P/T limits and LTOPS setpoints [3] [4], and (b) utilization of an alternate formulation for the LTOPS Enable Temperature (T_{enable}) based on a plant-specific analysis, instead of the generic ASME Section XI Code Case N-514 formulation employed in References [3] and [4]. The evaluation documented in [12] demonstrates that adequate safety margins are preserved and that required conservatisms are maintained.

Substitution of these alternate methodologies into the P/T limits, LTOPS and T_{enable} design analyses provides sufficient margin to extend the cumulative core burnup applicability limits for the existing P/T limits and LTOPS setpoints to values corresponding to the end of the current license period. This approach greatly simplifies the implementation process for the revised P/T limits, LTOPS setpoint, and T_{enable} design analysis. This simplification is accomplished by demonstrating that the current curves are bounding, i.e. only the applicability is changed.

The evaluation documented in [12], demonstrates that the existing P/T limit curves [3] [4] [9] [10], which are based on the ASME Section XI Appendix G K1a formulation and a limiting value of RTNDT of 162.9°F, conservatively bound the proposed revised design basis P/T limits curves, which are based on the ASME Section XI Appendix G K1c formulation and a limiting ¼-T RTNDT value of 218.5°F [7]. Thus, the cumulative core burnup applicability limit for the existing North Anna Units 1 and 2 Technical Specification P/T limit curves may be extended by simply revising the design and licensing basis P/T limit curves.

In a similar manner, the existing North Anna Units 1 and 2 Technical Specification LTOPS setpoints are demonstrated to provide conservative bounding protection for 100% of the proposed revised design basis isothermal limit curve at an extended cumulative core burnup applicability limit. Finally, a plant-specific

implementation of the analysis methodology, which supports ASME Section XI Code Case N-514, provides sufficient margin to extend the cumulative core burnup applicability limits for the existing North Anna Units

00-SE-OT-30

Description
UFSAR Change Request # FN 2000-021
UFSAR Section 6.2.1.3.2.5 will be changed to state that Steam Generator & Pressurizer cubicle blowout panels are steel sheet metal versus stainless steel. Also, 6.2.1.3.2.5 will state that a raised steel dome with holes is over the 6" refueling cavity drain opening versus a raised wire basket. UFSAR Section 6.2.2 will be changed to state that the containment recirculation sump first stage coarse mesh screen opening in Unit 2 is 0.558" versus 0.615".

Summary
Background:

UFSAR change request FN 2000-021 is being submitted to resolve some Containment Integrated Review Team (IRT) concerns discovered while validating certain Containment related UFSAR statements within Section 6.2 that were documented in PI N-1999-1988-R3.

Currently UFSAR Section 6.2.1.3.2.5 states that the Steam Generator and Pressurizer cubicle door blowout panels are made of thin-gauge stainless steel sheet. The blowout panel sheet metal material was incorrectly stated to be stainless steel. Based upon field verification during the past Unit 2 (9/99) and Unit 1 (3/00) Refueling Outages, it was observed that the related blowout panels are indeed galvanized carbon steel sheet metal. The galvanized carbon steel has no adverse impact on the Containment LOCA accident peak pressure analysis. This same UFSAR section also describes that a raised wire basket was installed over the 6" refueling cavity drain to prevent blowout panels from blocking the drain opening. The raised basket over the reactor cavity drain was incorrectly stated to be a wire basket. It is a raised steel dome with holes that provides for proper cavity drainage and the same function (i.e. to prevent blowout panels from blocking the drain opening) as would a wire mesh basket.

Currently UFSAR Section 6.2.2.2 states that the Unit 2 Containment recirculation sump first stage coarse mesh screen openings are 0.615 inches. The containment recirculation sump first stage coarse mesh screen opening in Unit 2 was incorrectly stated to be 0.615 inches. The openings on Unit 2 were measured by Mechanical Maintenance on 9/22/99 to be 0.558 inches which is in accordance with As Built drawing 11715-FV-1S, Recirculation Sump Screens, Sheet 1. The containment recirculation sump first stage coarse mesh screen opening is an administrative change that does not affect the accident analysis because any pump head loss due to the screen itself is negligible compared to the head loss due to debris accumulated on the screen (ref. SWEC calculation USB-275, Rev. 0, 7/94).

Major Issues considered:

DBA Loss of Coolant Accident (LOCA), Recirculation Spray Pump and LHSI Pump performance

- The proposed change to the UFSAR provides corrections to reflect current As Built configuration as discussed above. This change has no effect on the probability or consequence of the accidents/malfunctions considered. No modification of the plant will occur as a result of this change. The S/G & PZR cubicle doors blowout panel capability to function during an accident remains unchanged regardless of whether they are stainless steel sheet metal or galvanized steel sheet metal of the required thickness. The function of the raised basket over the refueling cavity 6" drain to prevent blowout panels from blocking the drain is met regardless of whether it is a raised wire basket or a raised steel dome with holes. The containment recirculation sump first stage coarse mesh screen openings size is an administrative change that does not affect the accident analysis because the effect on pump performance due to screen size itself is small compared to losses due to debris accumulation on the screens.

Unreviewed Safety Question:

This UFSAR change does not result in any increase in probability or consequence to any previously evaluated SAR accident. This change neither creates nor results in a malfunction not previously evaluated. No modification is made to the plant by this change. NAPS tech specs will be complied with as written and will not be altered by this change. This UFSAR change is administrative in nature to reflect current Containment As Built configuration. Based upon the above discussions and review, an unreviewed safety question does not exist and this UFSAR change should be permitted.

00-SE-OT-31

Description
Technical Specification Change Request #375
UFSAR Change Request FN-2000-027
The current Technical Specifications requirements specify that the refueling water storage tank (RWST) and the casing cooling tank (CCT) be at a concentration between 2300 and 2400 ppm and the safety injection accumulators (SIAs) be at a concentration between 2200 and 2400 ppm. The boron concentration of the spent fuel pool (SFP) is not explicitly stated in the Technical Specifications. This change will increase the boron concentration limits in the RWST, CCT, and SFP to 2600 – 2800 ppm and to 2500 – 2800 for the SIAs. The boron concentration of the SFP is being increased to keep the boron concentration consistent with the refueling canal and all portions of the reactor coolant system during refueling.

Summary
This change involves increasing the boron concentration in the refueling water storage tank (RWST), casing cooling tank (CCT), and the spent fuel pool (SFP) from the current Technical Specification limits of 2300 – 2400 ppm to 2600 – 2800 ppm and from 2200 – 2400 ppm to 2500 – 2800 ppm for the safety injection accumulators (SIAs).

It has been the Company's outage planning philosophy to stagger outages whenever possible in order to avoid load management, logistical and economic disadvantages associated with concurrent outages. In order to accommodate this outage planning philosophy, the fuel management plan for each unit provides for flexibility in the final end-of-cycle burnup including the use of power and RCS average temperature (Tavg) coastdowns.

While end-of-cycle coastdowns are fully evaluated from a safety analysis perspective, they represent an off-nominal operational mode that is undesirable from the standpoint of maximizing electrical generation. Designed reload cores with increased initial core reactivity is one means to reduce the need for extended end-of-cycle coastdowns. Increased core reactivity will require higher boron concentrations than previous cycles to meet increased shutdown requirements. One of the limiting parameters for core designers is the post-LOCA sump boron concentration limit. Increasing the boron concentration in the RWST, CCT, SIAs, and SFP will remove one obstacle currently preventing longer full power cycles.

Therefore, more reactive cores will reduce the duration of T-avg and power coastdowns, resulting in more energy production. Wider control bands on boron concentration limits will also provide greater operational flexibility.

Safety Significance

The following evaluations were performed to assess the impact of the proposed Technical Specification changes:

- Non-LOCA transients were evaluated, and it was determined that only the boron dilution event was potentially affected by the proposed increased boron concentrations.

- The effects of increased boron concentrations in LOCA evaluations were also considered. The time to switchover from cold to hot leg recirculation for long-term cooling following a loss of coolant accident (LOCA) was analyzed to determine the impact of the increased boron concentrations. The post-LOCA sump boron concentration limit was recalculated to ensure adequate post-LOCA shutdown margin. The post-LOCA containment sump and quench spray (QS) pH were calculated with an increased boron concentrations in the RWST, CCT, and SIAs to ensure that the pH remains within acceptable limits.

- Other evaluations, such as boron solubility, equipment qualification, and RWST and boric acid storage tank requirements were reviewed to ensure that a higher boron concentration does not adversely impact the safe operation of the plant.

These evaluations revealed that increased boron concentration limits in the RWST, CCT, SIAs, and SFP generally provide an analytical benefit from a reactivity management and accident mitigation standpoint. Potential adverse effects in the boron dilution event are accommodated in the reload verification process (Reference 2). The pH limits specified in the Standard Review Plan (Reference 1) continue to be met with increased boron concentration limits. The time interval for switchover from cold-to-hot leg recirculation to avoid boron precipitation in the vessel has been recalculated, and will be implemented upon approval of the increased limits. The increased boron concentration limits cause no adverse effects on the environmental qualification of equipment in the containment. A detailed discussion of these safety considerations is presented below.

Non-LOCA Chapter 15 Transients

Of the non-LOCA transients, only the results of the Boron Dilution accident analysis were found to be potentially adversely affected by the proposed increased boron concentrations. The adverse effect is a result of the increased RCS boron concentrations that would become feasible with the increased RWST boron concentration. The other non-LOCA transients were either not impacted or were made less severe as a result of the increased boron concentrations. For example, an increased boron concentration in the RWST and, hence, in the safety injection system, would provide less limiting Main Steamline Break analysis results. The Startup of an Inactive Loop accident analysis is insensitive to the refueling boron concentration, since this accident is precluded by Technical Specification requirements governing loop stop valve operations.

The Boron Dilution event at Refueling, Cold Shutdown, Intermediate Shutdown, and Hot Shutdown conditions is precluded by administrative lock-out of the primary grade water flow path in accordance with North Anna Units 1 and 2 Technical Specification 3.1.1.3.2. However, the Boron Dilution at Startup and at Power analyses are potentially impacted by the proposed increased RWST boron concentration. The impact on the Startup and At Power scenarios is indirect, and is a result of the increased allowable critical RCS boron concentrations resulting from the increased RWST boron concentration. An increased RCS boron concentration is explicitly considered in reload evaluations of the boron dilution event at startup and at power scenarios. As required by the current analysis of record, the reload evaluations of the Boron Dilution at Startup and at Power ensure that at least 15 minutes are available for corrective operator action between positive indication of a dilution in progress and complete loss of shutdown margin.

As previously indicated, the proposed increased boron concentrations can result in increased critical boron concentrations, which would result in higher reactivity insertion rates during a boron dilution event. The Departure from Nucleate Boiling (DNB) effect of these increased reactivity insertion rates were also considered, and were determined to be easily bounded by the rod withdrawal at power analysis. Therefore, the DNB acceptance criterion for the boron dilution event continues to be met.

Large Break LOCA

The effect of increased boron concentrations on the LOCA transient analysis was considered for both the large and small break scenarios. The large break LOCA is characterized by a rapid depressurization that causes the generation of significant voiding in the RCS. In accordance with Appendix K, the docketed North Anna LBLOCA analysis does not assume control rod insertion. As a result, heat generation in the core is reduced to decay heat levels by negative void reactivity. Therefore during the blowdown phase of the LBLOCA the core is shutdown and remains shutdown due to void reactivity.

The refill/reflood portion of the injection phase begins with the highly voided core and continues from downcomer refill through core reflood. During this time, void reactivity is of primary importance at the start and gradually begins to be replaced by boron as the primary source of negative reactivity. The docketed North Anna LBLOCA analysis shows that the peak clad temperature is reached prior to the time the boron becomes significant in maintaining core shutdown. In fact, boron concentrations are not modeled in peak clad temperature cases. Therefore, the increased boron concentration has no effect on the calculated results for the LBLOCA and would in fact provide a benefit if accounted for in the analysis. The proposed increase in RWST and SIA boron concentrations provides additional unmodeled conservatism.

Small Break LOCA

The small break LOCA (SBLOCA) analysis falls into the category of those transients that cause safety injection actuation. The small break LOCA model assumes the insertion of control rods in the calculation of core shutdown. Consequently, the boron concentration required to achieve the level of negative reactivity necessary to assure shutdown for the small break LOCA is significantly lower than the concentration required to assure shutdown for a large break LOCA. The increase in RWST and SIA boron concentration provides additional conservatism for the small break LOCA.

Cold-to-Hot Leg Recirculation Switchover Time

Following a LOCA, borated water from the RWST and accumulators enters the core region through the cold leg during the injection phase of the transient. Assuming a cold leg break, borated coolant enters the core region from the intact cold leg, down the downcomer, and into the core. Steam exits through the hot leg, and excess safety injection water spills out the break. Although the water vapor exits the core and condenses in the containment, only a small fraction of the dissolved boron is carried off in the steam. Therefore, the concentration of boron increases over time in the reactor vessel. If the boron concentration reaches the solubility limit, boron will begin to precipitate out of solution, forming a sticky paste that can block the coolant flow channels in the core. Such a condition may lead to inadequate cooling of the fuel.

If the break is in the hot leg or in the pressurizer, safety injection water will flow down the downcomer, up through the core, and out the break, thereby continuously replacing the boric acid solution in the core region. In such a situation, switchover to hot leg recirculation is not necessary. However, there is no unambiguous way to locate the pipe break from the control room, so switchover from cold leg to hot leg injection is required at a specific time for all LOCAs.

Because of the proposed boron concentration increase, the recirculation switchover time must occur sooner to avoid boron precipitation in the reactor vessel. The currently accepted boron precipitation limit is 23.5 weight percent boron, which includes a four weight percent safety margin to account for uncertainties. With a RWST and CCT boron concentration between 2600 – 2800 ppm and a SIA boron concentration between 2500 – 2800 ppm, a 5.26 hour switchover time has been calculated (Reference 4). For convenience, a 5 hour switchover time will be implemented, replacing the 7 hour time to prepare for switchover and the 10 hour switchover time currently in the North Anna Emergency Operating Procedures.

A potential issue was raised by Westinghouse concerning the possibility of inadvertent recriticality following switchover from cold leg to hot leg injection (Reference 9). The accumulation of boron in the reactor vessel following a large break LOCA, and prior to cold-to-hot leg switchover, results in a decrease in the sump boron concentration. Westinghouse postulates that switchover from cold leg to hot leg injection may wash out the concentrated boric acid in the core region, and replace it with the sump fluid which is depleted in boric acid. If the reduction in sump boron concentration during cold leg injection is sufficient, the cold-to-hot leg switchover may result in inadvertent re-criticality. This issue has been addressed by developing a Reload Safety Analysis Checklist (RSAC) parameter that ensures that the sump boron concentration and xenon reactivity at the time of cold-to-hot leg switchover is adequate to keep the reactor subcritical.

Post-LOCA Sump Boron Concentration Limit

Following a Small or Large Break Loss of Coolant Accident (SBLOCA or LBLOCA), fluid from various volumes accumulate in the containment sump. At North Anna, these volumes include the RWST, the chemical addition tank (CAT), the SIAs, the safety injection system piping (SI Piping), the reactor coolant system (RCS), the boron injection tank (BIT) and the CCT. All of these volumes contain boric acid solution with the exception of the CAT, which contains a sodium hydroxide solution. Depending on the magnitude of the loss of coolant accident (LOCA), some or all of the liquid contained in these volumes will be introduced to the containment, and will ultimately accumulate in the containment sump. It is assumed in the sump boron analysis for the design basis LBLOCA, that all of the liquid in these volumes is transferred to containment.

It is necessary to have a sufficiently high boric acid concentration in the sump mixture to ensure that the reactor remains subcritical. As more reactivity is loaded into the core, increased amounts of boron are required. The post-LOCA sump boron concentration limit for an increased boron concentration of 2600 to 2800 ppm in the RWST and CCT has been recalculated and will be incorporated into the Reload Safety Analysis Checklist (RSAC) (Reference 2) upon approval of the boron concentration increase (Reference 5).

Post-LOCA Sump and Quench Spray pH Limits

Limits are placed on the containment sump and QS pH because of material considerations and to reduce the evolution of iodine from the liquid. A post-LOCA sump pH range of 7.0 to 9.5 is specified in the Standard Review Plan (SRP) to avoid to onset of stress corrosion cracking (Reference 1). A pH range from 8.5 to 10.5 is specified in the SRP (Reference 1) to minimize the evolution of iodine during post-LOCA operation of the containment spray system.

The pH of the post-LOCA sump is determined by a volume-weighted average of the boric acid and sodium hydroxide concentrations from each analyzed volume. Because the data table used to interpolate the pH assumes that boric acid and sodium hydroxide concentrations are expressed as molarities (moles solute per liter), each volume's concentration (weight percent) is converted to a molarity prior to mixing the contents of the individual volumes in the sump.

The pH of the QS is calculated on the basis of the molarity and volumetric flow rate of liquid drawn from the RWST and CAT into the QS pump suction. The molarity of the RWST and CAT solutions is a simple conversion based on the weight percentage of the solute in the solution, and the specific gravity of the solution.

After consideration of the proposed increased RWST, CCT, and SIA boron concentrations, the post-LOCA containment sump and QS pH continue to meet the acceptance criteria (i.e., post-LOCA sump pH must be greater than 7.0 and less than 9.5 and the QS pH must be greater than 8.5 and less than 10.5) (Reference 6).

Boron Solubility

A boron concentration of 2800 ppm does not approach the solubility limit at the temperatures of the RWST. The temperature of the RWST fluid is limited to between 40 °F and 50 °F in Technical Specification 3.5.5. Figure 6.3-18 of Reference 3 shows that a boron concentration of about 2.5 weight percent boron (~4370 ppm) remains soluble at temperatures above 32 °F (Reference 3).

Equipment Qualification

Chemical spray is one of the environmental factors used to qualify the class 1E electrical equipment to assure operation when required. For the North Anna units, this environmental factor is considered for equipment inside containment experiencing a LOCA environment. There are two sources of chemical spray: quench spray and recirculation spray. The quench spray takes borated water from the RWST and a NaOH solution from the chemical addition tank (CAT). The recirculation spray system takes suction from the containment sump.

Increasing the boron concentration to 2600 – 2800 ppm in the RWST and CCT and to 2500 – 2800 ppm in the SIAs will not adversely affect the environmental qualification of equipment in the Equipment Qualification Master List (EQML). The corrosive agent in chemical spray is primarily NaOH. Increasing the boron concentration lowers the solution pH making it less corrosive (more neutral). Therefore, higher boron concentration limits are acceptable, even for those components qualified at a lower boron concentration (Reference 7).

RWST and Boric Acid Storage Tank (BAST) Volume Requirements

Technical Specification Bases 3/4.1.2 requires that the boration capability of the RWST and the boric acid storage tank (BAST) be sufficient to provide a 1.77%Δk/k shutdown margin from end-of-cycle (EOC) hot full power conditions after xenon decay and cooldown to 200 °F. Furthermore, the same shutdown margin must be maintained after cooldown from 200 °F to 140 °F.

The volume requirements are calculated by determining the reactivity required to achieve cooldown to either 200 °F from HFP or to 140 °F from 200 °F. The volume required to achieve this concentration is determined by converting the required reactivity by a differential boron worth. The required reactivity is determined in a conservative fashion by adding the temperature defects, xenon reactivity, and shutdown margin. A simple mixing model is used to determine the volume of RWST and BAST volume needed to achieve the required boron concentration in the vessel (Reference 8).

As part of this evaluation, Reload Safety Analysis Checklist (RSAC) parameters have been developed in order to ensure the BAST requirements are met on a cycle to cycle basis. The revision and incorporation of RSAC parameters is included in the Technical Specification Change Action Plan.

Based on the above evaluation, the proposed changes to the RWST, CCT, SIA, and SFP boron concentration do not adversely affect the safe operation of the plant.

Transition Consideration for Use of Opposite Unit's RWST

Upon increasing the boron concentration limits for the first unit, and prior to implementing the increased concentrations in the second unit, charging header cross-connect will allow flow from the opposite unit's RWST which will be at a higher or lower boron concentration than the accident unit. Accidents requiring flow from the opposite unit's RWST are outside of the design basis and therefore not formally analyzed. However, use of the cross-connect in beyond design basis events (loss of all injection flow from the accident unit, for example) will continue to be effective (that is, water of slightly lower boron concentration but high with regard to SDM requirements is preferable to no water, for instance). Therefore no changes to the procedural guidance for RWST/charging header cross-connect is required for this change.

Summary

1. Increasing the boron concentration limits for the RWST, CCT, SIAs, and SFP will not increase the probability of occurrence of any known accident and does not adversely affect the safe operation of the plant. Appropriate design constraints were analyzed for changes to T.S. 3.1.2.7, 3.1.2.8, 3.5.1, 3.5.5, 3.6.2.2, 3.9.1, and Bases 3/4.1.2 and 3/4.9.1 and none were found to be more limiting than currently documented in the UFSAR.
2. Increased boron concentration limits for the RWST, CCT, SIAs, and SFP will not increase the consequences of any accident previously evaluated in the Safety Analysis Report. The increased boron concentration limits reduce the time to switchover from cold to hot leg recirculation, which will prevent boron precipitation in the reactor vessel following a LOCA. A reduced switchover time will be implemented in the EOPs as part of the Technical Specification Implementation Plan. The post-LOCA sump boron concentration limit is revised to ensure adequate post-LOCA shutdown margin. The post-LOCA containment sump and quench spray (QS) pH remain within the limits specified in the Standard Review Plan. All other transients either were not impacted or were made less severe as a result of the increased boron concentrations. Therefore, accident analysis results meet all design criteria as stated in the UFSAR.

3. The proposed boron concentration increases do not add new or different equipment to the facility, nor do they significantly change the manner that installed equipment is being operated. There are no changes to the methods utilized to respond to plant transients and no alterations to the way that the plant is normally operated. The proposed UFSAR and Technical Specification changes do not alter instrumentation setpoints that initiate protective or mitigative actions. As a result, no new failure modes are being introduced. Therefore, the possibility for an accident of a different type than was previously evaluated in the SAR is not created.

00-SE-OT-32

Description
Technical Specification Change Request No. 369A
Technical Specification Change Request Package 369A: 1) modifies TS 3.4.1.4 - reactivity requirements for an isolated-filled loop to be consistent with the mode dependent requirements for the active portion of the RCS; 2) incorporates additional controls (LCOs and SRs) in TS 3.4.1.6 to address the reactivity concerns associated with the seal injection source to an RCP in isolated drained loop [during the vacuum-fill evolution]; and 3) revises Bases Section 3/4.4.1 to discuss the reactivity controls associated with vacuum-assisted loop backfill.

Summary
The current Technical Specifications permit returning an isolated Reactor Coolant System (RCS) loop to service by either of two methods. The first method, when the loop is isolated but not drained, requires the isolated loop to be operated on recirculation flow for a specified period of time prior to returning the loop to service. This activity serves to equalize reactor coolant temperature and boron concentration among the isolated and operating loops. The second method, when a loop is isolated and drained, permits returning the loop to service by back-filling the loop from the active portion of the RCS volume through partially opened loop stop valves. Specific controls are being established in the Technical Specifications to ensure reactivity and coolant inventory control during the loop backfill evolution.

To address the NRC's request for additional information, the original Technical Specification change has been modified to include additional limiting conditions for operation and surveillance requirements for the source of borated water for seal injection to the RCPs and modified reactivity controls for an isolated-filled loop. The Bases have been revised to further discuss the additional controls for the loop backfill evolution. The proposed changes will ensure that an inadvertent/undetected positive reactivity addition does not occur. The revised TS package supercedes the August 4, 1999 proposed Technical Specifications change in its entirety.

Compliance with current Technical Specifications administratively precludes the possibility of an inadequate boron concentration in makeup flow derived from the reactor cavity or RWST. Technical Specifications controls [LCOs and SRs] are being established for blended makeup flow from the BAST and PG water storage tank to ensure adequate boron concentration, and to eliminate the potential for inadvertent under-boration due to improper blending during the vacuum-assisted backfill evolution (seal injection operating).

The philosophy of the Startup of an Inactive Loop accident analysis centers on avoidance of the preconditions for the accident (i.e., reduced boron concentration and/or temperature in the isolated loop). The proposed Technical Specifications changes augment the current Technical Specifications to permit establishing RCP seal injection into the isolated-drained loop to establish a partial vacuum in the isolated loop. As described below, the proposed revised Technical Specifications explicitly consider the potential effects of RCP seal injection into the isolated loop, and include LCOs and surveillance requirements that will continue to preclude the preconditions for a Startup of an Inactive Loop accident.

The existing Technical Specifications provide protection against a Startup of an Inactive Loop accident when isolated-drained loops are restored by non-vacuum-assisted back-fill operations, or isolated-filled loop are restored by recirculation. As described in the bases for TS 3/4.4.1, the filled loop recirculation activity ensures equilibration of the isolated loop's boron concentration and temperature with the active portion of the RCS. Non-vacuum-assisted backfill operations preclude the preconditions for a Startup of an Inactive Loop accident by requiring 1) the loop to be verified drained before commencing backfill of the loop from the active volume of RCS, 2) operable source range instrumentation to provide secondary indication of any RCS makeup boron concentration discrepancy, and 3) a minimum RCS volume to ensure adequate decay heat removal capability. The proposed Technical Specifications provide protection against a Startup of an Inactive Loop accident under these conditions, and also when loops are restored to service by vacuum-assisted back-fill operations. The proposed Technical Specifications governing vacuum-assisted back-filling of isolated-drained loops preclude the preconditions for a Startup of an Inactive loop accident by requiring 1) the water used to makeup to the active volume of the Reactor Coolant System for water to the reactor coolant pump seal injection to be adequately borated to meet the shutdown margin requirements for the applicable mode of operation, 2) operable source range instrumentation, and 3) a minimum RCS volume to ensure decay heat removal. These controls (boron concentration, nuclear instrumentation, and RCS volume) have been established for each phase of returning an isolated-drained loop to service. Additionally, conservatively bounding analyses demonstrate that the reactivity effects of temperature differences between the isolated and non-isolated portions of the RCS will not result in a significant reactivity insertion.

During the backfill evolution, if blended makeup is used as a minimum it must aligned to the normal charging path (B cold leg) to ensure that makeup is mixed with the active RCS volume other paths can be used simultaneously (e.g., auxiliary spray). Continuous mixing of the active RCS volume is provided by the Residual Heat Removal System. Therefore, secondary indication of mis-blending makeup flow from the BAST and PG water storage tank is provided by operable source range instrumentation. These controls ensure that makeup flow to the active RCS volume and to the isolated loop (through RCP seal injection) will not result in an inadvertent and undetected boron concentration less than that required by Technical Specifications in a reactor coolant loop being brought back to service.

The proposed TS controls and associated procedural requirements for the drained loop backfill procedure preclude the possibility of inadvertent and unidentified introduction of under-borated water to an isolated and drained loop. Adequate TS controls are being proposed to ensure that the initiation of seal injection in order to establish a partial vacuum in an isolated and drained loop will not create the potential for an inadvertent and undetected introduction of under borated water into the isolated loop prior to returning the isolated loop to service.

The proposed TS controls and existing procedural controls will ensure that an inadvertent introduction of under-borated water into the RCS will not go undetected. In addition, those same controls ensure that the makeup source for RCP seal injection and the active volume of the RCS boron concentration is > shutdown margin boron concentrations for the applicable mode. Repeated sampling during RCP seal injection and backfill/makeup operation, mixing of reactor coolant by RHR flow, and continuous monitoring by source range instrumentation ensure than an inadvertent and undetected introduction of reactor coolant with a reduced boron concentration to an isolated and drained loop will not occur. Therefore, the proposed changes do not create an unreviewed safety question.

00-SE-OT-33

Description
Technical Specification Change Request No. 379.
UFSAR Change Request FN-2000-026.
Periodic Test Procedure 1(2)-PT-17.2, Rod Drop Time Measurement.
Obtain a License condition from the USNRC that sanctions the elimination of the seismic allowance adjustment from the RCCA (rod) drop time surveillance criteria.
Replace existing rod drop time test criteria with test limits that are based on the safety analysis limits and design uncertainties. Monitor and trend rod drop time data to identify and evaluate any adverse trends.
Remove the reference to the seismic adjustment from Technical Specification3/4.1.3.4 – Rod Drop Time BASIS
Remove the reference to the seismic adjustment from UFSAR Section 4.2.3.4.2- Control Rod Drive Mechanisms

Summary
A. Background

On September 6, 1990 (Reference 1), the USNRC issued OL Amendments No. 139 and 122, associated with North Anna Units 1 and 2, respectively. These amendments increased the control rod drop time requirements specified in LCO 3.1.3.4 from a previous value of 2.2 seconds to a new value of 2.7 seconds. This change to the allowable control rod drop time was requested to support a planned fuel design change from the Westinghouse Low Parasitic 17 x 17 (LOPAR) fuel assembly to a new 17 x 17 assembly with Westinghouse VANTAGE 5H fuel assembly design features. The new design, designated North Anna Improved Fuel (NAIF), utilizes Zircaloy grids and smaller diameter thimble tubes. The TS amendment request was made, and granted, on the basis that the increased drop time was required because of a reduction in thimble tube diameter associated with NAIF.

Discussions with Westinghouse during this same time frame revealed that the recommended increase in drop time from Westinghouse included an allowance to incorporate the effects of a seismic event on the calculated drop time, and was not solely based on the change in the fuel design. Since the 2.7 second drop time specified in new LCO 3.1.3.4 was assumed in the accident analyses supporting the LOPAR/NAIF transition, the question arose at the time of NAIF implementation as to whether or not the seismic effect should be taken into account by reducing the drop time surveillance test acceptance criterion from the 2.7 second value to a smaller value. Obviously the seismic effect cannot be measured at the plant.

Ultimately, Virginia Power concluded that the seismic allowance should be subtracted from the 2.7 second specified drop time to yield a smaller surveillance procedure limit.

B. Technical Basis of Proposed Change

Virginia Power is currently considering use of advanced fuel products in North Anna cores which incorporate a number of features to improve fuel reliability, ease of use and increase DNBR margins for anticipated transients. However, the fuel product design also may cause a modest increase in measured RCCA drop times. The expected drop times will be within the 2.7 second limit reflected in TS LCO 3.1.3.4 and the accident analyses. However, it is anticipated that, when the seismic adjustment is subtracted from the 2.7 second limit for the new fuel product, the resulting surveillance limit may not be met for all RCCA's.

One possible option would be to reperform the accident analyses and increase the 2.7 second analysis limit on the drop times, thereby increasing available margins in the corresponding surveillance limit. Based on our review of this option, we believe that such a change would also necessarily involve concurrent protection system changes (such as reductions in the high pressurizer pressure and/or low RCS flow reactor trip setpoints) which have the potential to reduce normal operating margin and increase the potential for reactor trip events and the resulting plant transients and equipment stresses.

C. Proposed Solution

As a result of these considerations, Virginia Power is proposing the elimination of the current practice of subtracting the seismic allowance from the Technical Specification LCO value to yield the limit applied in the surveillance test procedures. Rather than reduce the LCO value by the seismic value, the new surveillance limit will be reduced from the LCO to a value that accounts for uncertainties. Measured rod drop time data will be trended to ensure that adverse trends in performance are readily identified and evaluated.

D. Technical Justification

The following sections provide the technical justification of the proposed request, and in particular our basis for concluding that our request is consistent with the NRC's policy statement on probabilistic risk analysis (PRA). Reference 2. This justification is presented in a format which is consistent with the Principle Elements of Risk-Informed, Plant-Specific Decisionmaking set forth in Regulatory Guide 1.177 (Reference 3). That is to say, the technical justification demonstrates that:

- The proposed change meets the current regulations. No exemption or rule change is being requested.
- The proposed change is consistent with the defense-in-depth philosophy. Traditional engineering considerations have been used to demonstrate this consistency.
- The proposed change maintains sufficient safety margins. Traditional engineering considerations have been used to demonstrate that this is the case.
- The proposed change produces a negligible change in core damage frequency or risk and is consistent with the Commission's Safety Goal Policy Statement
- There is no impact on Virginia Power's Configuration Risk Management Program. RCCA drop time performance will continue to be monitored in a manner that ensures a high degree of reliability for the RCCA's to insert upon demand.

1. The proposed change meets the current regulations. No exemption or rule change is being requested.

The Control Rod Drive Mechanisms (CRDM's) and Rod Cluster Control Assemblies are designed to continue to function after a seismic event. This is consistent with General Design Criterion (GDC) 2.

Westinghouse has traditionally taken a conservative approach in applying GDC –2 to the functionality of the RCCA's as follows- not only were the RCCA's to be confirmed to insert during and following a seismic event, but also the RCCA drop time calculated to result from a concurrent reactor trip and seismic event was reflected in the accident analyses.

Since the seismic effect cannot be measured in surveillance testing, Virginia Power opted to reflect the seismic effect by maintaining the drop time in the Limiting Condition for Operation (LCO) consistent with the safety analysis assumption and reducing the test acceptance criterion for as-found RCCA drop times by a seismic allowance.

This proposed change to the non-LOCA accident analysis basis will not eliminate the confirmation of the capability of the RCCAs to insert during and following a seismic event. The only change proposed is the elimination of the surveillance test adjustment for an assumed increase in the RCCA drop time resulting from a concurrent trip and seismic event. The technical arguments which demonstrate that this change is consistent with the considerations of GDC-2 are set forth in subsequent discussions. The control rod drop time assumed in the accident analysis remains unchanged. {Note: the large break LOCA analysis does not credit RCCA insertion, per the rules of 10 CFR 50.46. The small break LOCA analysis will continue to reflect the effects of a concurrent seismic event in the assumed RCCA drop times i.e. the analysis value is adjusted upward from the LCO limit to account for the seismic effect }.

2. The proposed change is consistent with the defense-in-depth philosophy.

Historical analytical experience has shown that only a few accidents in the UFSAR are impacted, in terms of available margins to acceptance limits related to the integrity of fission product barriers, by increased control rod drop times. Specifically, these events are:

- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From A Subcritical Condition (RCS overpressure case) – UFSAR Chapter 15.2.1
- Loss of external electrical load (RCS and main steam system overpressure case)- UFSAR Chapter 15.2.7
- Locked Reactor Coolant Pump Rotor (RCS overpressure and fuel integrity cases)– UFSAR Chapter 15.4.4
- Complete Loss of Forced Reactor Coolant Flow (fuel integrity case) – UFSAR Chapter 15.3.4
- Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (fuel integrity case)- UFSAR Chapter 15.4.6.

Review of each of these events shows that significant conservatism will continue to apply to each analysis, ensuring that analytical results continue to bound the actual results of any actual transient. Among the major conservative assumptions which apply are the following:

- The surveillance test ensures that the slowest measured rod has a drop time which is less than the acceptance value. Since there is a distribution of drop times and many rods will have a drop time which is significantly less than the acceptance value, this ensures additional conservatism in the analysis.
- The total trip reactivity for the inserted rods is set at a conservatively low value with respect to typical reload core design results. The most reactive RCCA is assumed to stick in its fully withdrawn position after trip. Trip reactivity vs. rod position is calculated based on a conservative bottom-peaked axial power distribution, which effectively delays the reactivity insertion.
- For the overpressure cases, no credit is taken for the operation of pressurizer sprays or pressurizer PORV's, and the pressurizer safety valves are all assumed to be at the positive end of the setpoint tolerance band specified in Technical Specifications LCO 3.4.3.
- Conservative coefficients of reactivity (i.e. Doppler and moderator) are chosen which bound the most limiting times of life. Typically, the analysis assumes a moderator temperature coefficient which is representative of conditions which are only achievable for a few EFPH at the very beginning of cycle.
- The Reactor Protection System (RPS) setpoints assumed in the safety analysis (i.e., the low coolant loop flow rate reactor trip) are demonstrated to be conservative by the inclusion of appropriate uncertainties for process measurement and signal delay.
- No credit is taken in these analyses for
    - direct reactor trip on turbine trip
    - the steam relieving capacity of the condenser steam dumps for cases where condenser vacuum is available
    - pressurizer spray
    - automatic rod control

Consideration of all of these conservatisms in the aggregate leads one to conclude that superimposing them upon a concurrent seismic event represents an event of extremely low probability. This is equivalent to stating that the probability of exceeding 110% of the design RCS pressure limit or violating the design DNB in the event of one of the initiating events discussed above is extremely small. Elimination of the seismic allowance from the rod drop time surveillance test criterion does not alter this conclusion.

3. The proposed change maintains sufficient safety margins.

The guidance contained in Regulatory Position 2.2, "Traditional Engineering Considerations," applies various aspects of maintaining sufficient safety margin to the subject of changes to TS.

Because the proposed change will not alter the limiting results of the safety analyses presented in Chapter 15 of the UFSAR, safety margins are maintained. The proposed change will not alter the ability of the reactor protection and control system to perform their design functions or to meet the applicable criteria set forth in the IEEE and ANSI standards and in 10 CFR 50 Appendix A. The reactor coolant system and main steam system will continue to meet applicable ASME code requirements.

Analyses continue to demonstrate that the RCCAs will perform their design function by inserting into the core following a reactor trip, even with a postulated concurrent seismic event. The only change proposed is elimination of the seismic adjustment from the rod drop time surveillance test limits. The value of rod drop time assumed in the current safety analyses is not changing.

4. The proposed change produces a negligible change in core damage frequency or risk and is consistent with the Commission's Safety Goal Policy Statement.

The rod drive system, including the control rods themselves, is not explicitly modeled in the North Anna Probabilistic Safety Assessment (PSA) model. The RCCA drop time seismic component will thus have no effect upon the basic events (i.e. the component failure probabilities) in the PSA model. In general the accident sequence analysis in the PSA model is dependent upon the function of each system, including the reactor trip system. As discussed previously, while a seismic event could potentially delay a reactor trip by a marginal amount, it would not plausibly prevent a reactor trip.

The Virginia Power assessment therefore focused on estimating the impact of delayed reactor trip from a seismic event upon the PSA initiating events. For purposes of the assessment, initiating events were classified in two categories: dependent events, or those which could result from seismic activity (e.g. a loss of external electrical load resulting from seismically induced switchyard damage) and independent events, such as a locked reactor coolant pump rotor. The independent events considered here are not expected to occur as a result of seismic activity, but their consequences could be potentially increased by the seismically induced RCCA drop time.

Based on a review of our current safety analyses, Virginia Power concluded that the limiting event from the standpoint of a challenge to the reactor coolant pressure boundary integrity is the loss of external electrical load. This event was discussed earlier in the Engineering Evaluation. A probability estimate of a seismically induced loss of load leading to RCS failure, loss of coolant and subsequent core damage was made. We conclude that the probability of this event is several orders of magnitude below the North Anna PSA baseline Core Damage Frequency (CDF), and also less than the 5.0e-7 incremental conditional core damage probability (ICCDP) threshold that is considered small in NUREG-0800 Section 16.1.

For an examination of a limiting independent event, the locked reactor coolant pump (RCP) rotor event was examined. This event was discussed earlier in the Engineering Evaluation. A probability estimate of a locked RCP rotor event coincident with a seismic event was made. This event was assumed conservatively to lead to a LOCA in every case. Even with this conservative assumption, the core damage probability was well below the 5.0e-7 threshold for classification as "small" and several order of magnitude below the North Anna PSA baseline Core Damage Frequency (CDF).

5. There is no impact on Virginia Power's Configuration Risk Management Program. RCCA drop time performance will continue to be monitored in a manner that ensures a high degree of reliability for the RCCA's to insert upon demand.

Regulatory Guide 1.177 requires a three-tiered approach for evaluating the risk associated with proposed Technical Specifications Allowed Outage Time Changes. Virginia Power has applied Regulatory Guide 1.177 as a framework for developing the basis of the proposed change (Reference 4).

- Tier 1: PRA Capability and Insights. As noted above, an incrementally increased RCCA drop time will have a small impact upon the North Anna CDF. The Large Early Release Frequency (LERF) impact was not quantified, but is assessed by similar methods as also being "small".
- Tier 2: Avoidance of Risk-Significant Plant Configurations. Plant risk is monitored and controlled by the Configuration Risk Management Program set forth in North Anna Technical Specification 6.8.4.g. This program already works to minimize risk and avoid risk-significant configurations. Its ability to do so will not be affected by the proposed change in treatment of control rod drop times in the accident analyses. The proposed change does not affect or change the risk significance of any component or group of components.
- Tier 3: Risk Informed Configuration Risk Management. The risk management program presently in place continuously reviews planned maintenance configurations to ensure that risk is maintained at acceptable low levels. Emergent configurations are evaluated via the same tools. Typically, the corrective actions imposed by Technical Specifications provide appropriate compensatory measures. Additional compensatory measures will be added as a part of the 10 CFR 50.65 a (4) implementation program, if ongoing review finds that such measures are warranted.

In conclusion, the proposed change to the safety analysis treatment of allowable RCCA drop time will have a negligible impact upon the Core Damage Frequency at North Anna and on the Configuration Risk Management Program.
While a USQ is posed, we believe that the proposed change is acceptable and should be approved by the Commission on the basis that

The proposed change meets the current regulations. No exemption or rule change is being requested.

The proposed change is consistent with the defense-in-depth philosophy. Traditional engineering considerations have been used to demonstrate this consistency. The proposed change maintains sufficient safety margins. Traditional engineering considerations have been used to demonstrate that this is the case.

The proposed change produces a negligible change in core damage frequency or risk and is consistent with the Commission's Safety Goal Policy Statement. There is no impact on Virginia Power's Configuration Risk Management Program. RCCA drop time performance will continue to be monitored in a manner that ensures a high degree of reliability for the RCCA's to insert upon demand.

00-SE-OT-34

Description

UFRAR Change Request No. FN 2000-029

PI (Plant Issue) N-1999-3012 pointed out that the methodology for calculating SW of reservoir water losses per 0-PT-75.8 and Section 3.8.4.8 of the UFSAR are different. The UFSAR calculation focuses on calculating seepage losses. The method used in the UFSAR employs calculational methods that require parameters not currently measured and recorded (i.e. continuous wind speed at the SW reservoir). It also specifies using a methodology which has never been developed (i.e. Drift loss will be determined by developing a correlation between wind and drift loss).

NAPS determines net loss rate from the SW reservoir using simple mass balancing calculations. Since assurance of design basis inventory for 30 days without makeup is the reason for measuring leakage, the current reservoir net loss test and calculation are more appropriate than the methodology described in the UFSAR. Thus, the UFSAR will be changed to describe current measurement techniques.

Summary

PI (Plant Issue) N-1999-3012 pointed out that the methodology for calculating SW reservoir water losses per 0-PT-75.8 and Section 3.8.4.8 of the UFSAR are different. The UFSAR calculation focuses on calculating seepage losses. The method used in the UFSAR employs calculational methods that require parameters not currently measured and recorded (i.e. continuous wind speed at the SW reservoir). It also specifies using the methodology which has never been developed (i.e. Drift loss will be determined by developing a correlation between wind and drift loss).

NAPS determines net loss rate from the SW reservoir using simple mass balancing calculations. Since assurance of design basis inventory for 30 days without makeup is the reason for measuring leakage, the current reservoir net loss test and calculation are more appropriate than the methodology described in the UFSAR. Thus, the UFSAR will be changed to describe current measurement techniques.

This UFSAR Change Request is issued based on Plant Issue Resolution N-1999-3012-R4 that requires revising UFSAR Section 3.8.4.8 to bring it in agreement with methodology of 0-PT-75.8.

This UFSAR update does not involve an unreviewed safety question: Updating the method of calculating SW reservoir loss based on actual reservoir inventory instead of theoretical calculations provides the station with a more reliable and practical way to evaluate SW reservoir loss which includes possible SW system leakage (if any). This method is based on actual measurements of reservoir loss and was implemented in 1978.

Probability of LOCA or any other accident is not affected by this UFSAR update. This UFSAR update does not create the possibility for an accident of a different type than was previously evaluated in the SAR since no equipment, mode of operation or surveillance requirements are affected.

This UFSAR update does not affect any station system or equipment malfunction. It documents a reliable method of evaluation of SW reservoir loss.

This update of the UFSAR describes the existing method for calculating the SW reservoir loss. The margin of safety of TS as described in the basis section is not affected.

00-SE-OT-35

Description
UFSAR Change Request No. FN 99-063 affecting UFSAR Sections 15.4.1.7.3 and 15.4 References and Tables 6.2-42 and 6.3-6.UFSAR Section 15.4.1.7.3 and Tables 6.2-42 and 6.3-6 are being updated to reflect an increase in total allowable ECCS leakage from 900 cc/hr to 4800 cc/hr as assumed for the Technical Specification Change Request No. 366 (TSCR-366, Reference 1) analysis submitted for NRC approval. The reference section of UFSAR Section 15.4 is being changed to update the reference for UFSAR Section 15.4.1.7.3.
Currently, Section 15.4.1.7.3 indicates that allowable ECCS leakage in areas of the Auxiliary Building that are indirectly filtered (i.e., outside the charging pump cubicle) is limited to 600 cc/hr. This limit is based on a total ECCS leakage of 900 cc/hr. The LOCA offsite and control room dose consequences resulting from the ECCS leakage into both the directly filtered areas (the charging pump cubicles and the Safeguards Building) and the indirectly filtered areas of Auxiliary Building have been evaluated based upon the assumptions of TSCR 366. As a result of the re-analysis, the control room, EAB, and LPZ doses remain below the values submitted to the NRC for TSCR-366. The revised allowable ECCS leakage in indirectly filtered areas on the Auxiliary Building remains 600 cc/hr based on a total allowable ECCS leakage of 4800 cc/hr.

Summary
In Technical Specification Change Request No. 366 (TSCR-366, Reference 1), offsite and control room dose consequences resulting from a Loss of Coolant Accident (LOCA) were analyzed to evaluate a proposed increase in the amount of allowable leakage from ECCS components from 900 cc/hr to 4800 cc/hr. This analysis was performed using the LOCADOSE code and methodology consistent with the Standard Review Plan (NUREG – 0800, Revision 2). UFSAR Change Request No. FN 99-063 changes the UFSAR for two reasons.

The first reason that the UFSAR is being updated is to reflect the increase in total allowable ECCS leakage from 900 cc/hr to 4800 cc/hr as assumed in the TSCR-366 analysis submitted for NRC approval (Reference 1). The changes to UFSAR Section 15.4.1.7.3 and Tables 6.2-42 and 6.3-6 were not previously identified by the review process for TSCR-366 because the references to the ECCS leakage limit of 900 cc/hr were added to the UFSAR after TSCR-366 was submitted to the NRC for approval.

The second reason that the UFSAR is being updated is that Section 15.4.1.7.3 indicates that ECCS leakage in the indirectly filtered areas of Auxiliary Building (i.e., outside the charging pump cubicles) is limited to 600 cc/hr based on a total ECCS leakage limit of 900 cc/hr (Reference 3). As a result of the proposed increased ECCS leakage for TSCR-366 the offsite and control room dose consequences resulting from ECCS leakage both into directly filtered areas (charging pump cubicles and safeguards) and indirectly filtered areas of auxiliary building have been evaluated. This evaluation, which is documented in Reference 2, is based on a total allowable ECCS leakage of 4800 cc/hr. As a result of this re-analysis, the control room, EAB, and LPZ doses remain below the values submitted to the NRC for the TSCR-366. The allowable ECCS leakage in indirectly filtered areas of the Auxiliary Building remains 600 cc/hr. The UFSAR is being changed to replace the reference to the analysis based on 900 cc/hr total leakage to the one for 4800 cc/hr.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased as a result of this change. The ECCS leakage does not affect the probability of occurrence of an accident or equipment malfunctions. The dose consequences are within the previously analyzed results reported in TSCR-366.

The possibility for an accident or malfunction of equipment of a different type than any evaluated previously in the safety analysis report is not increased. The total allowable ECCS leakage has not changed from the previously evaluated limits proposed in TSCR-366.

The margin of safety as defined in the bases section for any Technical Specification is not reduced. The proposed changes are consistent with the total ECCS leakage assumption of the analysis for TSCR-366.

Also, the doses that result from the directly and indirectly filtered ECCS leakage discussed above are less than the values submitted with TSCR-366 for NRC approval. Additionally, the dose consequences of the accident were determined to meet the 10 CFR 100 limits and the limits specified in GDC-19. Therefore, an unreviewed safety question does not exist as defined in 10 CFR 50.59.

00-SE-OT-36

Description
North Anna UFSAR Change Request No. FN 2000-025
UFSAR Change Request No. FN 2000-025 proposes changes to the North Anna UFSAR to provide the description of the licensing basis for feedwater isolation for the Rupture of a Main Steam Pipe accident in Chapter 15 and editorial clarifications to equipment descriptions, locations, and functions for various components included in Table 3C-3. This package is a result of the Integrated Configuration Management Project review for North Anna Power Station.

Summary
Change (1)

Provide a description in the UFSAR of the licensing basis for feedwater isolation redundancy assumed as part of the large Rupture of a Main Steam Line (MSLB) accident analysis in UFSAR Section 15.4.2. The automatic trip of the main feedwater pumps and closure of the main feedwater pump discharge motor operated valves on receipt of a SI actuation signal is credited for redundant feedwater line isolation as discussed in 15.4.2. This proposed change to the UFSAR description of the response to a large MSLB provides a description of the licensing basis for the credit of this redundancy feature as a back-up to automatic closure of the safety grade feedwater regulating valve bypass valves.

The feedwater isolation design provides safety grade isolation capability with the feedwater regulating valves, feedwater isolation valves and the feedwater regulating bypass valves. However, a single failure of the bypass valves to close requires dependence on the SI actuation initiated trip of feedwater pumps and closure of the pump discharge isolation valves. This actuation and certain components that perform the isolation function are non safety grade. The basis for reliance upon this commercial grade equipment to perform the back-up isolation function is provided on a generic industry wide basis in NUREG-0138. This NRC document provides allowance for the utilization of commercial grade equipment as back-up to safety grade equipment in the isolation of feedwater or steam systems following a main steam line break accident. The basis for this acceptability is documented in NUREG-0138 as primarily due to the low probability of steam line break accident and the expected low significance of resultant dose consequences, as well as the expected reliability of commercial grade equipment in this application. Calculation SM-0938 evaluated the failure of a feedwater bypass valve to close and the reliance on feedwater pump trip and closure of pump discharge valves to isolate the failed bypass valve. The calculation concluded that the accident consequences met applicable acceptance criteria and this scenario was bounded by the consequences resulting from the MSLB accident analysis documented in UFSAR 15.4.2. Main feedwater pump discharge valve closure time assumptions in SM-0938 are supported by the valve design specification for stroke time and by periodic MOV maintenance and testing.

The limiting safety analysis requirement for isolation of main feedwater is provided by the analysis of the Rupture of a Main Steam Line accident (MSLB) in UFSAR 15.4.2. The reliance upon the SI actuation initiated trip of feedwater pumps and automatic closure of the feedwater pump discharge isolation valves in order to back-up the safety grade feedwater isolation following a MSLB cannot increase the probability of an accident occurrence, or cause a different accident than those currently analyzed, since feedwater isolation is not an accident initiator for this limiting accident. Feedwater isolation reduces the severity of postulated MSLB accidents. The use of commercial grade equipment for redundant feedwater isolation cannot increase the consequences of analyzed accidents. Feedwater isolation is expected to be accomplished by the safety grade feedwater regulating valves, feedwater isolation valves and the feedwater regulating valves bypass valves that receive automatic closure signals in response to accidents that result in a SI actuation (e.g., MSLB). In the event of a failure of feedwater bypass valve to close (postulated single active failure), the SI actuation initiated trip of the main feedwater pumps and automatic closure of the feedwater pump discharge valves provides back-up feedwater isolation. NUREG-0138 documents NRC acceptance of this commercial grade plant design feature on an industry basis as discussed above. Calculation SM-0938 evaluated the failure of a feedwater bypass valve to close and the reliance on feedwater pump trip and closure of pump discharge valves to isolate the failed bypass valve. The calculation concluded that the accident consequences met applicable acceptance criteria and this scenario was bounded by the consequences resulting from the MSLB accident analysis documented in UFSAR 15.4.2. The probability of equipment malfunction is not increased, nor is the possibility of new malfunction created, by the reliance of non-safety grade equipment for back-up feedwater isolation since this redundancy credit has been evaluated and accepted by NUREG-0138. Additionally, the credited non-safety grade equipment is considered to exhibit high reliability in service as documented in NUREG-0138. The margin of safety associated with the MSLB is not reduced as determined by the results of calculation SM-0938.

Change (2)

Make editorial changes to the Mark No., Equipment Description, Location or Function description for various entries in Table 3C-3 to correspond the Equipment Database System information and the installed location and configuration in the plant.

The proposed changes to Table 3C-3 information are editorial in nature, and are consistent with descriptive information located elsewhere in the UFSAR. These changes provide consistency with location information and equipment descriptions provided in the Equipment Database System. The proposed enhancements to the equipment function descriptions is consistent with other sections of the UFSAR. No new or different equipment or component functions are created by this proposed editorial clarification to the information presented in Table 3C-3.

00-SE-OT-37

Description
UFSAR Change Number FN 2000-031.
NAPS UFSAR Section 6.2.1.2.5 will be revised to indicate that the NS surge tank and associated supports will maintain their structural integrity during a Design Basis Earthquake and thus, will not impact the integrity of SR components.

Summary
MAJOR ISSUE
The Civil / Structural / Seismic Design & Licensing Basis Integration Review team identified open issues related to the North Anna UFSAR. Section 6.2.1.2.5 states that a seismic failure of any component within the neutron shield tank water cooling subsystem would not damage Seismic Class 1 structures, systems or components. However, it appears that the seismic failure of the NS surge tanks could impact surrounding SR components.

JUSTIFICATION
Calculation CE-1497 was performed by Corporate Engineering Mechanics to demonstrate the seismic adequacy of the NS surge tanks and associated supports. Therefore, there will be no gross failure of the NS surge tanks and associated supports during a seismic event and Safety-Related components will not be impacted.

The UFSAR will be revised to clarify that there will be no seismic failure of the NS surge tank. This revision does not change the conclusion of the UFSAR statement, but it clarifies how that conclusion is reached.

UNREVIEWED SAFETY QUESTION ASSESSMENT
1. Accident and malfunction probability has not changed by the proposed UFSAR change. An engineering evaluation demonstrates the structural adequacy of the NS surge tank and associated supports under seismic conditions, thus ensuring that Safety-Related components will not be impacted. No modifications to equipment or operating procedures are involved with the proposed UFSAR change. The evaluation performed by engineering provides assurance that the previous UFSAR conclusions are correct and the UFSAR change clarifies the basis for those conclusions.

2. Accident and malfunction consequences are not changed by the proposed UFSAR change. An engineering evaluation demonstrates the structural adequacy of the NS surge tank and associated supports under seismic conditions, thus ensuring that Safety-Related components will not be impacted. No modifications to equipment or operating procedures are involved with the proposed UFSAR change. The evaluation performed by engineering provides assurance that the previous UFSAR conclusions are correct and the UFSAR change clarifies the basis for those conclusions.

3. No new accident or malfunctions are created by the proposed UFSAR change since no modifications to equipment or operating procedures are involved.

The margin of safety is not reduced by the proposed UFSAR change because the integrity of Safety-Related components is assured as demonstrated by engineering evaluation.

00-SE-OT-38

Description
North Anna UFSAR Change Request No. FN 2000-013
UFSAR Change Request No. FN 2000-013 contains a list of changes, some of which are editorial in nature, which need to be corrected or clarified in the UFSAR sections that discuss North Anna's Reactor Mechanical Design System. This package is a result of the Integrated Configuration Management Project review of North Anna Power Station's Reactor Design system.

Summary
The above editorial/administrative changes are within the current design and licensing basis of the facility. These changes do not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed editorial/administrative changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

None of the proposed editorial changes involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial/administrative changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed editorial/administrative changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

With regard to the criteria set forth in 10 CFR 50.59 and based on the evaluation summarized above, the proposed editorial/administrative UFSAR changes have been determined not to represent an unreviewed safety question.

00-SE-OT-39

Description

UFSAR Change request FN-99-076

TRM Change request #39

Delete the fuel transfer tube manual isolation valves (1-FH-1 and 2-FH-1) from the list of containment isolation valves. The containment isolation function is satisfied by the design of the blank flange inside containment. The design of the blank flange employs a testable, dual O-Ring seal configuration. This is consistent with industry standard practices for fuel transfer tube penetrations, and meets the design basis criteria stated in UFSAR section 6.2.4.1

Summary

This change will remove references in the North Anna Technical Requirements Manual and in the North Anna UFSAR that imply the fuel transfer tube manual isolation valves function as containment boundaries. This change involves a change in status only for station valves 1-FH-1 and 2-FH-1. These valves are not containment isolation boundaries. The containment isolation function is performed by the fuel transfer tube blank flange, located inside containment. The sealing mechanism for this blank flange consists of a testable, dual O-Ring seal configuration.

A type "B" test is performed to verify the integrity of the two O-Ring seals on the fuel transfer tube blank flange. This is consistent with other containment penetrations whose sealing mechanism consist of two O-Ring seals, such as the containment electrical penetrations and access hatches. This design is similar to the equipment hatch plate seal configuration, in that the containment isolation boundary is provided by two O-Ring seals. This application is consistent with industry standards, and meets the design basis criteria for containment isolation as stated in UFSAR section 6.2.4.1.

The manual isolation valves 1-FH-1 and 2-FH-1 are not type "C" tested, as is the case for all containment isolation valves. This is also consistent with standard industry practice.

The testing requirements for the fuel transfer penetration components will not be altered by this change. The description of these valves in the UFSAR and Technical Requirements Manual should be modified to describe their intended functions. These functions are to provide a means of mitigating a loss of spent fuel pool level or reactor cavity level during refueling operations, and to allow for isolation of the fuel building transfer canal such that the blank flange may be removed for refueling operations.

The station configuration and normal position of these valves will not be altered by this change. Therefore, the effects of postulated accidents on the penetration will not be altered, and the probability of such accidents occurring will not be increased. The testing frequency and testing requirements for the fuel transfer tube penetration will not be altered. The fuel transfer isolation valves will continue to be maintained in the closed position during normal operation, and opened during mode 5 and 6 to support refueling operations. As a result, there will be no impact on station environmental practices or policies.

Failure of a single containment penetration boundary will not affect the overall integrity of the fuel transfer penetrations. Two barriers to the outside environment are provided by the O-Ring seals of the fuel tube blank flange. If one of the seals should fail, the other seal will maintain the integrity of the penetration.

No changes are required to the North Anna Technical Specifications or the North Anna station operating license due to this change. The North Anna Technical Specifications will continue to be complied with as written.

This change of status will not create the possibility for a malfunction of equipment differing from those previously evaluated, nor will it affect station effluents or unit power level. The ability of the station to achieve and maintain safe shutdown conditions will not be affected by this change, and no changes in occupational exposure will occur.

00-SE-OT-40

Description

UFSAR Change Request FN 2000-010

The results of Va. Power Calculation ME-0581 showed the environmental impact on adjacent EQ rooms from a high energy line break (HELB) in the turbine building. This calculation found the maximum breach size into control room envelope (CRE) and the emergency diesel generator rooms (EDGRs) so that these rooms would not become a harsh environment. This UFSAR change will explain these environmental impacts on above rooms during a turbine building HELB.

Summary

This UFSAR change discusses the environment impact from a turbine building HELB on the safety related equipment located in the CRE and EDGR. Calculation ME-0581 found the worst case turbine building temperature, pressure and humidity profiles in the turbine building after a HELB. From this the maximum breach into the CRE and EDGR was calculated as a function of room temperature so that these rooms would not become harsh environments. No physical or procedural changes to the power plant will result from this UFSAR change. During a Turbine Building HELB, the temperatures in the control room envelope (CRE) could potentially reach 120°F. This value has not changed. This temperature limit concurs with UFSAR Section 9.4.1. If the worst case of the Turbine Building HELB occurred, the duration of these CRE temperatures would be less than 30 minutes. The ability of the operators to control the plant or potential for personnel injury will not change. The potential for a Turbine Building HELB will be unchanged.

From ME-0581, the non-running EDG room temperature profile was determined as a function of the EQ room breach size. For the operating EDG, UFSAR Section 9.5.5 states that the maximum outside ambient temperature that the EDG's can operate under design basis conditions is 104°F. The air intake to the EDGs is located appreciably far away from the turbine building louvers and exhaust fans so that steam from a turbine building will not influence the operation of the running EDG. Steam infiltration from the turbine building will be minimal and will not effect the EDGs. As a result, the temperature change in the EDGR with the EDG running will be insignificant.

00-SE-OT-41

Description
Supplier Nonconformance Report 781 (Precision Components Corporation (PCC) NR 77232-44c)
North Anna ISFSI SAR
The inspection requirements for a weld repair performed on the TN-32 cask 19 inner shell were not met. ASME Code Section III Subsection NB paragraph 5120 (d) requires that the repair be MT inspected after stress relief unless the area is inaccessible. PCC performed MT inspection on the repaired area before stress relief but failed to perform a post-heat treatment inspection. Though the repair is not structurally inaccessible, operations required to perform the inspection pose substantial risk to the cask and its internals. Furthermore, evaluation shows that the repair is structurally acceptable.

Summary
This is a 10 CFR 72.48 safety evaluation.

The inner shell of TN-32 cask 19 contained misplaced layout punch marks. The proper process to weld repair the punch marks is to weld, stress relieve, MT inspect the repairs, then aluminum spray the inside diameter of the cask. PCC performed MT inspection prior to stress relief but failed to re-inspect after heat treatment. The aluminum spray was then applied, and the basket and rails were installed into the cask. The inspection deviation was discovered during the document review process.

Section 7.1.1 of the TN-32 TSAR Rev. 9A states, "Even though the Code is not strictly applicable to storage casks, it is the intent to follow Section III, Subsection NB of the Code as closely as possible for design and construction of the confinement vessel." ASME code Section III, Subsection NB Paragraph 5120 (d) requires that the repair be MT inspected after stress relief unless the area is inaccessible. Though the repaired area is not structurally inaccessible, operations required to perform the inspection pose substantial risk to the cask and its internals. The basket, rails, and aluminum spray must be removed and the cask must be laid on its side to perform the inspection. These operations have the potential to cause damage to the basket, basket rails, the cask containment shell, seal seating surface, and the cask outer wrapper and neutron shield boxes. Additionally, evaluation shows that the weld repair is acceptable as-is and the possibility of anomalies occurring after the pre-heat treatment MT inspection is remote. This conclusion is based on the following:

1. Evaluation given in SNCR 781 shows that the punch marks did not result in a violation of the minimum wall thickness.
2. The pre-heat treatment MT inspection was acceptable. For minor repairs of this nature, experience at PCC and Virginia Power indicates that areas that have acceptable MT inspections prior to heat treatment also have acceptable post-heat treatment MT inspections.
3. The repaired SA-203 base material is a Group B, high strength low-alloy steel with excellent toughness properties and good weldability.
4. The repair was made using the GTAW process in accordance with procedure SW-58-002. This welding method is a low hydrogen process and provides preheat and heat input rates that adequately heat soak the material for slow cooling. Thus, martensite formation and hydrogen diffusion, the mechanisms by which failure or cracking can occur in the heat affected zone, are prevented.
5. The repairs made are shallow unrestrained "joints" for which there is no concern for stress cracking or cold cracking upon weld completion.

This deviation is therefore only to the inspection process for the repair, not to the quality of the repair.

An unreviewed safety question does not exist for the following reasons:

1. The change will not increase the consequences or probability of accidents evaluated in the UFSAR. Accidents that were reviewed are included in North Anna ISFSI SAR Sections A.1.5, "Cask Sliding and Tip Over Accidents"; 8.2.9, "SSSC Drops"; 8.2.10, "Loss of Confinement Barrier"; and TN-32 TSAR sections 11.2.8, "Hypothetical Cask Drop and Tipping Accidents"; and 11.2.9, "Loss of Confinement Barrier." All code requirements relating to the quality of the

repair continue to be met and evaluation shows that the repair is structurally acceptable. Therefore, the cask confinement boundary is not affected and the probability of occurrence of any analyzed accident will not be increased. Since the change has no detrimental effect on the cask's structural integrity, there will be no increase in the consequences of an accident.

2. The change will not create the possibility for an accident of a different type than was analyzed in the UFSAR. The inspection deviation has no detrimental effect on the cask structural, thermal, criticality, or shielding analyses. Therefore, no parameters are affected that could form a precursor to another accident scenario.

3. The change does not increase the consequences or probability of malfunctions of equipment related to safety evaluated in the UFSAR. The malfunctions that were considered are those that compromise the cask's ability to maintain its confinement boundary and structural integrity in addition to its ability to be safely handled, transfer heat, maintain subcritical margin of stored fuel, and provide shielding. The inspection deviation will not affect the cask confinement boundary, structural integrity, thermal performance, criticality control, or shielding evaluation. The consequences of a malfunction of equipment identified above are the release of radioactive material to the environment. These consequences are evaluated in the North Anna ISFSI SAR. Since the North Anna ISFSI SAR assumes, for the loss of confinement accident, that all fuel stored in the cask fails, including cladding and fuel pellets, there is no malfunction which would produce an increase in consequences.

00-SE-OT-42

Description
UFSAR/ISFSI SAR Change Request #FN 2000-034
Topical Report Change Request Dated July 27, 2000
Changes will delete the title of "Radwaste Team Leader" from Chapter 17, Section 17.2.1.2.e and Figure 17.2,1-2 of the UFSAR. The Radwaste Team Leader position is being reassigned to the Operations Department at Surry and will therefore no longer report to the Superintendent Radiological Protection as stated. The Radwaste Team Leader is a "Surry only" position. The North Anna UFSAR requires change to delete the title however this change has no impact on North Anna.

Summary
The Radwaste Team Leader position will be eliminated, and a new position created. The Supervisor Water Treatment is the new position. The Supervisor Water Treatment position is not listed in the UFSAR; however, this position reports to the Supervisor Operations Support which is mentioned in the UFSAR.

An unreviewed safety question does not exist. The safety analysis is concerned with an administrative change regarding the reorganization of the Surry Radwaste Facility. The administrative change applies to the Radwaste Team Leader, which reports to the Superintendent of Radiological Protection. The UFSAR will be changed to delete mention of the Radwaste Team Leader in Chapter 17 Section 17.2.1.2 and to delete the position from the Onsite Nuclear Organization in Figure 17.2.1-2.

The administrative change in no way affects the plant configuration, therefore the probability of accidents occurring has not been changed. This change does not affect the facility or equipment at the Radwaste Facility. The margin of safety as described by Technical Specifications remains unaffected. The administrative change of title and reporting structure does not require a change to the Operating License or Technical Specifications.

00-SE-OT-43

Description

North Anna Appendix R Report Change Request #1999-N-015

Appendix R Report Change Request #1999-N-015 contains 76 proposed changes, some of which are editorial in nature, that need to be corrected or clarified in the Appendix R Report sections that discuss North Anna's FP System. The change package is the result of the Integrated Configuration Management Project review of the North Anna Power Station Fire Protection System.

Summary

The above editorial/administrative changes are within the current design and licensing basis of the facility. These changes do not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed editorial/administrative changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

None of the proposed editorial changes involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial/administrative changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed editorial/administrative changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

With regard to the criteria set forth in 10 CFR 50.59 and based on the evaluation summarized above, the proposed editorial/administrative UFSAR changes have been determined not to represent an unreviewed safety question.

00-SE-OT-44

Description

UFSAR Change Request FN 2000-033 ISFSI SAR Change Request IN 2000-002
Plant Issue N-2000-1212
UFSAR Figure 3.8-32; IFSFI SAR Figures 2-9 and 4-4; ISFSI Environmental Report Figures 2.4-3, 2.1-1 and 9.1-1 indicate that there is a fence and gate around the Service Water (SW) Pumphouse. In addition, UFSAR Figure 3.8-32 indicates that the area around the SW Pumphouse is a "PROTECTED AREA". There may have been a fence and gate around the SW Pumphouse many years ago but they have long since been removed. In addition, the SW Pumphouse is not considered to be part of the "Protected Area" at NAPS.

Summary

The major issue considered in this safety evaluation was whether security fencing is required at the SW Pumphouse and the consequences of not having fencing installed at that location. A review of the Physical Security Plan and the Security Implementing Procedures indicate that this fencing is not required. Most likely this fencing was installed at one time and the SW Pumphouse may have been considered a Protected Area since the SW Pumphouse contains safety related components. When the fencing was determined to be no longer required for security reasons and removed, the referenced UFSAR and ISFSI SAR figures were not revised to reflect the change. Removal of fencing from the area around the SW Pumphouse allows easier access for operators to monitor equipment and precludes possible damage that could be caused by the fence if it were to come loose and be blown onto equipment/components. The SW Pumphouse is missile protected and the entrance requires a key to gain access to the structure. No physical modification to the site is required by this change.

Therefore based on the above discussion an Unreviewed Safety question does not exist as a result of removing the security fence from around the SW Pumphouse.

00-SE-OT-45

Description
Technical Specification Change Request No. 339 ; UFSAR Change Request No. FN 2000-037.
This evaluation considers the establishment of Technical Specifications requirements (T.S. 3/4.7.14 and 3/4.7.15) for spent fuel pool (SFP) soluble boron concentration and fuel assembly loading restriction based on burnup and enrichment. These new Technical Specification requirements will permit elimination of the Boraflex credit from the spent fuel pool criticality calculations. Updates to the UFSAR are also required to reflect the elimination of Boraflex credit and addition of soluble boron credit.
The maximum fuel enrichment for North Anna Units 1 and 2 is to be increased from the current Technical Specifications limit of 4.3 weight percent U^{235} to 4.6 weight percent U^{235}. This will require modifications to North Anna Technical Specifications 5.3.1 and 5.6.1. Updates to the North Anna UFSAR are also required to reflect the higher enrichment limit.

Summary
Pursuant to 10 CFR 50.90, Virginia Electric and Power Company (Virginia Power) proposes to establish Technical Specifications requirements for spent fuel pool (SFP) soluble boron concentration and fuel assembly loading restriction based on burnup and enrichment. These new Technical Specification requirements will permit elimination of the Boraflex credit from the SFP criticality calculations. This requires upgrading the SFP boron concentration from an administrative limit of 2300 ppm to a Technical Specification limit of 2500 ppm. In addition, Virginia Power proposes increasing the maximum fuel enrichment for North Anna Units 1 and 2 from the current Technical Specifications limit of 4.3 weight percent U^{235} to 4.6 weight percent U^{235}.

The increase in enrichment, in conjunction with previously proposed Technical Specifications change to increase the soluble boron concentration in plant safety systems (Reference 1), will reduce the need for extended periods of reduced power operation (coastdowns) at the end of each operating cycle and permit fuel discharge burnups more compatible with the current lead rod limit of 60,000 MWD/MTU. This will help optimize fuel cycle costs while continuing to satisfy the current core power distribution and safety limits.

Due to gamma irradiation and exposure to a wet pool environment, degradation of the Boraflex panels was identified at several utilities. In June, 1996, the NRC issued Generic Letter 96-04 (Reference 2) which requested that licensees with spent fuel storage racks containing Boraflex provide an assessment of the physical condition of the Boraflex and verify that a subcriticality margin of 5% can be maintained in the spent fuel pool. Virginia Power conducted blackness tests at the North Anna Power Station in January of 1997. These tests and subsequent analysis confirmed that the 5% subcriticality margin can be maintained at North Anna (Reference 3). However, Virginia Power made an internal commitment to reinspect the condition of the Boraflex at 5-year intervals, with the next testing required by January, 2002 (Reference 4). The elimination of the Boraflex credit in the SFP criticality calculations will eliminate this periodic blackness testing requirement.

The use of higher enrichment fuel and the increase in the SFP boron concentration has the potential to impact the criticality analysis for fuel handling and storage. Calculations were performed to address the impact on the new and spent fuel storage areas, which are common to North Anna Units 1 and 2. The criticality analysis for the fresh fuel storage racks (Reference 5) demonstrated that the North Anna new fuel storage area meets the criticality limit of K_{eff} less than 0.98, and is safe under the criticality specifications set forth in Section 9. 1.1 of the Standard Review Plan (NUREG-0800).

The North Anna spent fuel racks were reanalyzed for a maximum enrichment of 4.6 weight percent U^{235} using the approved Westinghouse Owners Group methodology (References 6 and 7. It was demonstrated that a boron concentration of 230 ppm is sufficient to maintain the SFP $K_{eff} < 0.95$ during normal operating condition including uncertainties. An additional 550 ppm accounts for the postulated "worst-case" misload accident, and 120 ppm accounts for the burnup credit uncertainty. Therefore, the K_{eff} in the North Anna spent fuel pool will be maintained < 0.95, including uncertainties, under normal and accident conditions (excluding the boron dilution accident) with a soluble boron concentration of 900 ppm. The proposed

Technical Specification SFP boron concentration limit of 2500 ppm accounts for the boron dilution accident and allows a sufficient amount of time for operators to discover and mitigate any postulated boron dilution event in the SFP before the SFP boron concentration reaches 900 ppm (Reference 8).

Reference 1 involved a Technical Specification change to increase the SFP boron concentration to between 2600-2800 ppm during Mode 6. The SFP boron concentration proposed here is 2500 ppm, bounded by reference 1. The safety evaluation completed for Reference 1 (Reference 9) indicated that increasing the SFP boron concentration does not adversely affect the safe operation of the plant or result in an unreviewed safety question.

The environmental effects of the higher enrichment fuel were reviewed, as were impacts to the North Anna safety analyses. It was determined that the generic fuel data (temperature and pressures) provided by Westinghouse for input to safety analyses are valid for fuel enrichments which exceed the requested change (Reference 10), so inputs to existing safety analyses are unaffected by the proposed enrichment increase. The consequences of accident scenarios are also unchanged, because the source terms used to determine the releases from fuel during accidents are a function of burnup, rather than initial enrichment. As the operating power and fuel burnup limits are not being changed when the fuel enrichment is increased, it is concluded that there are no adverse effects on the types or amounts of any radiological releases.

Other areas identified as potentially being affected by the enrichment increase included the impact on reactor vessel fluence, control rod insertion, decay heat load in the SFP during a full core off load, dry fuel storage, core physics parameters and rod integrity (Reference 11). The higher enrichment and change to the SFP design bases has no appreciable effect on these areas except for the core parameters. Of the core parameters, existing limits for peaking factors, MTC, FTC and control rod worths will continue to be met. However, an increase in the critical boron concentration is expected. The critical boron concentration is evaluated as part of the reload design process, so the impact of any required increase will be calculated prior to each reload to ensure compliance with all applicable Technical Specifications.

The use of fuel with a higher initial U^{235} enrichment and the changes in the SFP design basis will not result in an unreviewed safety question as defined by 10 CFR 50.59. The basis for this determination is summarized below.

1. The proposed increase in maximum fuel enrichment or the changes to the SFP design basis will not increase the probability of an accident previously evaluated in the North Anna Units 1 and 2 UFSAR. The only accidents for which the probability of occurrence is potentially affected by the fuel enrichment and SFP changes involve criticality events during fuel handling and storage (e.g., fuel mispositioning). Positioning of fuel assemblies in the SFP has always been administratively controlled. The new Technical Specifications place additional restrictions on the determination of the appropriate position in the SFP for each fuel assembly, but these considerations are incorporated into our administrative tools and guidelines. The controls on fuel movement, including any checks to ensure the assembly is placed in the specified location, remain unchanged. Criticality safety analyses have been performed that demonstrate that the K_{eff} during the handling and storage of both new and spent fuel is low enough to ensure subcriticality during postulated accident conditions. In addition safety analyses of the dilution of the spent fuel pool have been performed to ensure that there is adequate time for a dilution accident to be found and mitigated before criticality is reached in the spent fuel pool. The probability of occurrence of criticality during fuel handling or storage is therefore not increased. Since subcriticality is maintained, no releases would result from the above handling and storage accident scenarios. In addition, since the burnup limit will not be increased beyond that approved in Reference 12, radiological consequences of other accidents previously evaluated in the North Anna Units 1 and 2 UFSAR will not be increased.

2. The possibility of an accident which is different from any already discussed in the North Anna Units 1 and 2 UFSAR is not created. Although there are new restrictions on placement of fuel in the SFP, the controls on fuel movement to the administratively specified locations in the pool are unchanged. The higher enrichment fuel and the new Technical Specifications for the spent fuel pool do not require any new or different plant equipment, and do not change the manner in which currently installed equipment is operated. There are no changes to normal core operation, and the units will meet all applicable design

criteria and will operate within existing Technical Specifications limits. Adherence to these standards and criteria precludes new challenges to components and systems that could introduce a new type of accident. Existing safety analyses of record will remain applicable for use of fuel with the higher initial enrichment.

3. The probability of a malfunction of equipment important to safety previously evaluated in the North Anna Units 1 and 2 UFSAR is not increased. The design of cores which incorporate fuel at the higher initial enrichment and the spent fuel pool will meet all applicable design criteria. Adherence to applicable standards and acceptance criteria, including existing limits on fuel burnup precludes new challenges to components and systems that could increase the probability of any previously evaluated malfunction of equipment important to safety. The use of a higher maximum fuel enrichment will not impose new performance requirements on any system or component such that any design criteria for fuel operation or storage will be exceeded. No new modes or limiting single failures are created by the use of a higher fuel enrichment. Safety analyses for the fuel storage area have demonstrated that subcriticality will be maintained during fuel handling and storage, including fuel mispositioning and pool dilution scenarios. The new Technical Specifications on the spent fuel pool do not introduce any new effluents or release paths, and do not affect the magnitude of any currently analyzed releases. Also, since the burnup limit of the fuel is not being increased, the radiological consequences of any malfunction of equipment important to safety previously evaluated in the North Anna UFSAR is not increased by the use of a higher fuel enrichment limit.

4. The possibility of a malfunction of equipment important to safety different from any already evaluated in the North Anna Units 1 and 2 UFSAR is not created. The design for North Anna cycles, which incorporate the higher enriched fuel, will meet all applicable design criteria. Fuel handling and storage in the spent fuel pool with additional restrictions on the selection of fuel storage locations and increased boron concentration will also meet all applicable design criteria. Adherence to existing Technical Specifications and design limits will preclude new challenges to components and systems that could introduce a new type of malfunction of equipment important to safety. No new failure modes have been created for any system, component, or piece of equipment, and no new single failure mechanisms have been introduced. No new or different plant equipment is introduced, and the operation of currently installed equipment is not changed. The use of higher enriched fuel and the changes to the SFP design basis have the potential to affect only criticality events during fuel handling and storage. Safety analyses demonstrated that K_{eff} will remain sufficiently low to ensure subcriticality, so no new releases will result and there is no impact on radiological consequences of accidents.

5. The margin of safety as defined in the Bases to any North Anna Technical Specification is not reduced. Safety analyses of record will remain applicable for the operation of fuel with a higher initial U^{235} enrichment and changes to the spent fuel pool. Criticality analyses demonstrate that the limits on K_{eff} for the new and spent fuel storage areas will be satisfied. Therefore, there is adequate margin to ensure subcriticality during the storage and handling of fuel, and the requirements of 10 CFR 50 Appendix A General Design Criterion 62 are satisfied.

The North Anna Units 1 and 2 Technical Specifications ensure that the plants operate in a manner that provides acceptable levels of protection for the health and safety of the public. The Technical Specifications are based upon assumptions made in the safety and accident analyses, including those relating to the fuel enrichment and the design of the fuel storage areas. The North Anna safety analyses for core operation will remain applicable for cores which use fuel with the higher U^{235} enrichment, and analyses have demonstrated that subcriticality will be ensured during fuel storage and handling accident scenarios. Therefore the regulated margin of safety as defined in the Technical Specifications is not affected by the proposed increase in initial fuel enrichment or changes to the spent fuel pool design basis.

Based on the evaluations and analyses results presented in the foregoing safety significance evaluation, it has been demonstrated that increasing the North Anna Units 1 and 2 maximum initial fuel enrichment to 4.6 weight percent U^{235} and changing the design basis of the spent fuel pool to eliminate any credit for Boraflex but take credit for soluble boron in the pool will not result in the acceptable safety limits for any incident being exceeded, or in any unreviewed safety questions as defined in 10 CFR 50.59.

00-SE-OT-46

Description
00-TSR-041
About 500 pounds of borated poly panels will be temporarily installed on the spent fuel cask lid and near the upper trunnions while the loaded cask is parked in the Decontamination Building north bay.

Summary
The activity proposed is to install, use and then remove a temporary radiation shield of borated poly-panels on the top of a loaded spent fuel cask in the Decontamination Building north bay. A safety evaluation must be performed per VPAP-2105 because the shielding will be directly loaded onto an operable safety related cask.

Installation of temporary radiation shielding performed at a loaded spent fuel cask is needed to reduce personnel exposure per ALARA policies. A detailed engineering evaluation of the shielding design (considering all issues directed by the Temporary Shielding Program) concluded that the shielding could be installed without affecting the operability of the safety related spent fuel cask. The basis for this conclusion is that the shielding loading on the cask lid is bounded by the original cask design which includes a 1000 + pound gamma shield that will not be in place until after the 500 lb. temporary shielding is removed.

The accident previously evaluated in the Safety Analysis Report was a load drop of a spent fuel cask. The activity will not be in place during any cask lifting operations and has no potential to affect lifting operations even after the shielding has been removed; therefore, the probability of occurrence for this accident will not be increased and the consequences of the accident is not affected. An accident of a different type attributable to a fault in the activity is not considered significant or credible. Therefore, the proposed activity does not constitute an unreviewed safety question.

The malfunction of equipment important to safety, previously evaluated in the Safety Analysis Report was the failure of the spent fuel cask to maintain an air-tight seal. The activity will be placed on top of the cask lid in a position that will not interfere with the O-rings, the sealing surfaces or the lid bolts. As such, the probability of a cask seal leak will not be increased by the activity and the consequences of a cask seal leak will be unchanged. A malfunction of a different type attributable to a fault in the activity is not considered significant or credible. Therefore, the proposed activity does not constitute an unreviewed safety question.

There are no applicable Technical Specifications bases sections to be affected; therefore, the proposed activity does not constitute an unreviewed safety question.

The proposed activity does not constitute an unreviewed safety question because it will not:
- require a change to the Operating License or Tech Specs,
- affect the ability of the station to achieve and maintain safe shutdown in the event of a fire,
- affect the Final Environmental Statement,
- affect effluents or power level in any way,
- have a significant adverse environmental impact,
- involve a change to the Environmental Protection Plan.

00-SE-OT-47

Description
TRM Change Request Number 42
Changes revise TRM Table 7.2-2 to correctly identify fire dampers 1-FP-FDMP-1015, 1-FP-FDMP-1017, 2-FP-FDMP-1016, and 2-FP-FDMP-1018 as required for Appendix R; add EQ doors 01-BLD-STR-S54-2 and 02-BLD-S54-13 and their allowed breach duration to TRM Table 12.2-2; and clarify TRM Section 7.3 to identify that fire brigade personnel must have sufficient training in or knowledge of safety-related systems to understand the effects of fire and fire suppressants on safe shutdown capability.

Summary
The NAPS Fire Protection Program provides assurance through a defense-in-depth design that a fire will not prevent the performance of necessary safe-shutdown functions and will not significantly increase the risk of radioactive releases to the environment. To accomplish this, the NAPS Fire Protection Program consists of: a) fire detection systems, b) fire extinguishing systems/equipment, c) administrative controls/procedures and d) trained personnel. The current North Anna License condition allows the licensee to make changes to the Fire Protection Program without NRC approval if those changes do not adversely affect the Station's ability to achieve and maintain safe shutdown in the event of a fire. The proposed changes to the TRM Fire Protection Program requirements have been determined to fall into this category. These changes will add two EQ Doors in Table 12.2-2, correct the classification of four fire protection program dampers from non-Appendix R to Appendix R in table 7.2-2, and add clarification for the minimum manning requirements in Section 7.3, "Fire Brigade" to be consistent with the latest engineering evaluations performed, the TRM and the Fire protection Program, VPAP-2401.

Correction of Fire Dampers from "Non-Appendix R" to "Appendix R" in TRM Table 7.2-2
Table 7.2-2, "Fire Dampers" lists fire dampers 1-FP-FDMP-1015, 1-FP-FDMP-1017, 2-FP-FDMP-1016, and 2-FP-FDMP-1018 located within the Unit 1 and 2 Cable Tray rooms as Non-Appendix R. The dampers are installed within the walls that separate the Cable Tray Rooms from the Turbine Building and Mechanical Equipment Rooms. Chapter 2, "Identification of Fire Areas" Table 2-1 of the Appendix R Report lists the Unit 1 and Unit 2 Cable Tray Rooms as Appendix R fire areas. Each room contains communication components that are required for safe shutdown.

Based upon the above and in response to Plant Issue N-1999-2748 and ET SE-99-073, TRM Table 7.2-2 is being revised to add fire dampers 1-FP-FDMP-1015, 1-FP-FDMP-1017, 2-FP-FDMP-1016, and 2-FP-FDMP-1018 as required for Appendix R.

Addition of EQ Doors and Allowed Duration for EQ Door Breaches in TRM Table 12.2-2
TRM Table 12.2-2, "EQ Doors Breach Duration" provides a listing of EQ Doors, a description of the door's location and the allowed breach duration. Changes will add the following EQ doors 01-BLD-STR-S54-2 and 02-BLD-S54-13 to TRM Table 12.2-2. The same duration as that assigned to the EQ door at the entrance to the Unit 2 ESGR from the turbine building is assigned to the added doors on the basis that the postulated consequences for all the doors are the same. ET NAF-96-0173 Rev 1 and Calculation SM-967 provide the supporting details for these EQ Doors and their allowed door breach duration.

Fire Brigade
TRM Section 7.3 requires a Fire Brigade of at least 5 members to be maintained onsite at all times. In order to be consistent with the minimum manning requirements of Section 6.6, "Fire Brigade Program" of the Fire Protection Program (VPAP-2401), TRM Section 7.3 will be clarified to identify those personnel who must have sufficient training in or knowledge of safety-related systems to understand the effects of fire and fire suppressants on safe shutdown capability. The proposed changes will add to Section 7.3: "To comply with the minimum manning requirements, three members of the Fire Brigade will be from the Operations Department, one of which will be the Scene Leader, and two from the Security Department. The Fire Brigade Scene Leader and at least two brigade members shall have sufficient training in or knowledge of safety-related systems to understand the effects of fire and fire suppressants on safe shutdown capability."

An unreviewed safety question does not exist. The safety analysis provides corrections and clarifications regarding the TRM Fire Protection Program requirements. Clarification of the manning requirements for the fire brigade and correcting Tables 7.2-2 and 12.2-2 will not change the operation of the plant nor the fire protection systems. No new accident initiators are established as a result of the proposed changes. Therefore, the probability of concurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report are not increased, the possibility of any accident or malfunction of a different type than previously evaluated in the SAR is not created, and the margin of safety as described in the Bases remains unaffected. Changes and clarification do not require a change to the Operating License or Technical Specifications.

00-SE-OT-48

Description

North Anna UFSAR Change Request No. FN 99-037 UFSAR Change Request No. FN 99-037 contains a list of changes, some of which are editorial in nature, which need to be corrected or clarified in the UFSAR sections that discuss North Anna's Reactor Nuclear Design System. This package is a result of the Integrated Configuration Management Project review of North Anna Power Station's Reactor Design system.

Summary

The editorial/administrative changes are within the current design and licensing basis of the facility. These changes do not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed editorial/administrative changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

None of the proposed editorial changes involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial/administrative changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed editorial/administrative changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

With regard to the criteria set forth in 10 CFR 50.59 and based on the evaluation summarized above, the proposed editorial/administrative UFSAR changes have been determined not to represent an unreviewed safety question.

00-SE-OT-49

Description
ET CEP 00-0024, Rev. 1, "Technical Changes to Exemption Requests 1, 27 and 31"
Rev. 1 to ET CEP 00-0024 has been prepared to revise Exemption Request 27 by deleting reference in the exemption to Excore Neutron Flux Monitoring.

Summary
The regulatory requirement for source range nuclear monitors to be considered Appendix R safe shut-down equipment has been questioned based on section 2.03 b 1 (d) of the NRC Baseline Inspection module 71111.05. The section states "source range neutron indication in not necessarily required and an alternative method of reactivity measurement can be provided." In the Appendix R SER dated Nov. 18, 1982, Virginia Power did not commit to having source range neutron indication for safe shutdown areas, only for alternative shutdown areas. The containment is not an alternative shutdown area based on this SER or the current Appendix R Report. According to the Appendix R Report Tables 4-2A and 4-2B, the containment is either an III.G.1 area (sufficient equipment outside the area) or III.G.2 area (adequate separation), depending on the system or systems relied upon.

IN 84-09, states that source range neutron indication (SRNI) is required, again it is only required for alternative shutdown. A fire in containment may effect the SRNI, but adequate margin would be available to maintain hot shutdown for 72 hours by injecting borated water, since no equipment outside of containment would be affected. Without SRNI there would be no direct indication of reactivity, yet there would be ample means to increase shutdown margin thus eliminating the chances of a criticality. This is believed what is meant by the inspection procedure stating that SRNI is not necessarily required. It was originally believed, during the drafting of the Appendix R Report, that any fire would require the performance requirements of III.L.2 to be met. In fact, the requirement to meet any of III.L only comes about for alternate shutdown areas. Therefore, the absolute requirement for SRNI availability is a misnomer, and it is only required in those areas which North Anna classifies as alternate shutdown.

Exemption Request 27, titled "Separation of Instrumentation Inside Containment – Intervening Combustibles with Fire Stops" was submitted and approved by the NRC in SER dated November 6, 1986. In the SER the NRC states that this was an exemption from 10CFR50 Appendix R III.G.2.d. The SER therefore considers Containment an III.G.2 area, not subject to 10CFR50 Appendix R III.L performance monitoring criteria. 10CFR50 Appendix R, section III.G.1 and III.G.2 require there be separation of redundant trains for safe shutdown. If separation of redundant trains is not available then an alternate shutdown method can be used per Appendix R Section III.G.3. If an alternate shutdown method is used, the criteria of III.L must be followed.

The North Anna Containment's are not considered alternate shutdown areas since there is available redundant equipment. This redundant equipment is not alternate shutdown equipment; therefore there is no regulatory requirement for the performance requirements included in III.L of 10CFR50 Appendix R to be met. The purpose of the SRNI is to indicate to the operators if the reactor is going critical during the shutdown process. Borated water would be injected following a containment fire and loss of SRNI, as currently directed in 1(2)-FCA-5, to provide adequate shutdown margin. Control room monitoring of the charging system would not be affected by a containment fire since this system is located outside of the containment structure and would not be fire affected. Since borated water would be injected, dilution would not occur; therefore hot shutdown could be achieved and maintained. The FCAs direct the performance of 1/2-PT-10 "Shutdown Margin Determination" which provides an alternative method of reactivity determination; therefore the station can reach and maintain safe shutdown conditions for a containment fire condition without SRNI. Cold shutdown would be required within 72 hours. This would allow ample time for the operations staff to verify adequate RCS boration prior to cooldown in accordance with 1/2-FCA-5.

Rev. 1 to ET CEP 00-0024 clarifies exemption 27 to agree with the position currently stated in the Appendix R Report (Note 25 of Table 3-1) and ET CEP 00-0037 that there is no requirement for nuclear instrumentation to be available for Appendix R safe shutdown for a fire in containment. There is no adverse impact on Unreviewed Safety Questions or the ability to safely shutdown in the event of a fire. No new accidents or malfunctions are created and current postulated accidents and malfunctions are not affected.

00-SE-OT-50

Description
Temporary Shielding Request 00-TSR-042
It is proposed to install temporary lead blanket shielding on operable/operating 6" diameter Safety Related CH piping on the inlet to the Charging Pumps to provide protection to workers installing DCP 99-010 for replacement of SW lines in the Charging Pump cubicles. Each blanket weighs 15 lb./sq. ft. and the plan is to double the blankets to provide additional protection.

Summary
Temporary Shielding Request 00-TSR-042 will install seismically qualified lead blanket shielding on operable/operating 6" diameter Safety Related CH piping on the inlet to the Charging Pumps. Lines from both the Low Head SI Pump Discharge and the Volume Control Tank will be shielded. The shielding will be installed in each of the six Charging Pump cubicles in a consistent manner. The shielding will be installed and removed under the VPAP-2105 Temporary Shielding Program. The proposed activity will not:

- Increase the probability of occurrence for a seismic event,
- increase the consequences of a seismic event,
- create the possibility for an accident of a different type

No postulated equipment failures could be identified which are associated with temporary radiation shielding of this piping.

There are no applicable Technical Specification basis descriptions; therefore, the margin of safety will not be affected by the proposed activity. The proposed activity will be conducted in compliance with VPAP-2105 in order to satisfy Technical Specification 6.11. No changes to the Operating License or Technical Specification are required.

Applying 30 pounds of lead blanket shielding per linear foot around the outside of the 6" CH system piping will have no affect on any station operations following a fire.

There are no discernable environmental concerns associated with the proposed activity.

CONCLUSION

Installing temporary lead shielding as proposed does not constitute an unreviewed safety question because:

- The probability of experiencing a design basis seismic event is unrelated to the act of installing temporary lead shielding.
- Since the affected piping has been seismically analyzed for the applied shielding load and the response was found to be within acceptable limits, there is no identified increase in the consequences related to a design basis seismic event.
- Seismic concerns, the line temperature limit (for compatibility with lead blankets) and the need to avoid interference with the nearby valve reach rod are the main issues associated with the proposed activity. No other concerns were identified and an accident of a different type could not be postulated.
- Technical Specification margins of safety are not reduced. Seismic analysis has shown that the level of stress with shielding in place is within the design limits.

00-SE-OT-51

Description
TSCR #381, Administrative change to NPF-4, TS 2.1, Safety Limits, and TS 2.2, Limiting Safety System Settings.
TSCR #381 involves the removal of completed, redundant, expired, or otherwise non-applicable license conditions and associated Technical Specifications from the Facility Operating License for North Anna Unit 1. Includes editorial corrections and the relocation of the License Condition 2.D(3) u and Appendix C.

Summary
The proposed change to the North Anna Unit 1 Facility Operating License (FOL), NPF-4, and Technical Specification Sections 2.1 and 2.2 make minor editorial corrections, relocates two license conditions within the license itself, and removes completed, redundant, expired or otherwise non-applicable license conditions and associated Technical Specification requirements and provides a license document that is directly applicable to the current plant licensing and design bases. There is no safety significance associated with this proposed change since the change does not alter any currently applicable Facility Operating License requirements. Accordingly, the current North Anna Unit 1 licensing and design bases are unchanged and an Unreviewed Safety Question (USQ), as defined in 10 CFR 50.59 does not exists.

The proposed change is administrative (and in part editorial) in nature and neither station operations nor design are affected by the change. The removal of license conditions and associated Technical Specifications of complete (FOL Sections 2.D(1), 2.D(3)j, 2.F, FOL Attachment 1, TS 2.1, TS 2.2, and FOL Attachment 3), redundant (FOL Sections 2.D(3)c, and 2.D(3)r), or no longer necessary or expired (FOL Sections 2.B, 2.D(3)o and 2.D(3)v) requirements has no impact on plant operations since these requirements no longer have a legitimate means of being applied. The relocation within the Facility Operating License of the single Appendix C license condition to Section 2.C(3)d, and the fire protection requirements (2.D(3)u) does not alter the technical basis, requirements or the implementation of these items. The split of Section 2.C(2) to separately account for the amendments to the Technical Specifications and the Environmental Protection Plan for reasons of accuracy does not alter the intent of the original license condition. The impact of the re-design of the SG and RCP supports on the previously evaluated accidents was performed, approved and documented by the issuance of the license amendments for FOL Section 2.F. Removal of the license conditions which refer to completed work and which have a design and licensing bases that is documented in the UFSAR also does not alter station operation or the design of the affected components. The proposed change is within the current design and licensing bases of the facility. This change does not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. This change does not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These proposed changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed change to the North Anna Unit 1 Facility Operating License and Technical Specifications does not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

The license conditions and Technical Specifications that are being removed or relocated by this proposed change do not impact station operations or station equipment in any manner. The proposed change does not involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients that has not been previously analyzed. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed change to the North Anna Unit 1 Facility Operating License and Technical Specifications does not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. Since station operations are not affected by the proposed change, the change does not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed change to the North Anna Unit 1 Facility Operating License and Technical Specifications does not involve a reduction in any margin of safety described in the bases of the Technical Specifications.

00-SE-OT-52

Description

UFSAR Change Request FN 2000-041

The UFSAR is being updated to include revised 10 CFR 50.61 Pressurized Thermal Shock (PTS) screening calculation results for North Anna Unit 2 based on currently available reactor vessel materials surveillance data, including that associated with the recently-analyzed North Anna Unit 2 Capsule W.

Summary

PURPOSE

The purpose of this safety evaluation is to support implementation of revised 10 CFR 50.61 Pressurized Thermal Shock (PTS) screening calculation results for North Anna Unit 2 based on currently available reactor vessel materials surveillance data, including that associated with the recently-analyzed North Anna Unit 2 Capsule W (1).

DISCUSSION

10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", requires that licensees "consider plant specific information that could affect the level of embrittlement." Recently-acquired North Anna Unit 2 reactor vessel materials surveillance program Capsule W analysis results have been incorporated into 10 CFR 50.61 PTS screening calculations, as well as into calculations that demonstrate the conservatism of analyses previously performed for compliance with 10 CFR 50 Appendix G, "Fracture Toughness Requirements". (See Engineering Transmittal ET-NAF-2000-0100, "Evaluation of North Anna Unit 2 Reactor Vessel Materials Surveillance Capsule W Analysis Results, North Anna Unit 2," dated August 2000 (2).)

SAFETY SIGNIFICANCE

Surveillance program results have been obtained recently from the North Anna Unit 2 plant-specific surveillance program (Capsule W) (1). The evaluation documented in Reference (2) integrates this recently obtained data into the analyses that demonstrate compliance with 10 CFR 50 Appendix G and 10 CFR 50.61. Analyses that consider reactor vessel surveillance data include (a) RCS P/T limit curves, (b) the LTOPS setpoint and enabling temperature, and (c) 10 CFR 50.61 PTS screening calculations. The Reference (2) evaluation considers the impact of the newly acquired surveillance data on licensing basis analyses in a manner consistent with applicable regulatory guidance. Specifically, the calculation of the Reference Temperature for the Nil Ductility Transition (RT_{NDT}) is performed in accordance with Regulatory Guide 1.99 Revision 2 (4), and the regulatory guidance provided in the meeting minutes from the November 12, 1997 NRC/Industry meeting on reactor vessel integrity (5). PTS screening calculations were performed in accordance with 10 CFR 50.61 (3). Supporting calculations are documented in Reference (6).

As documented in the Reference (2) evaluation, the PTS screening calculation results for North Anna Unit 2 continue to meet the applicable screening criteria. Further, the RT_{NDT} value used in the development of the current North Anna Unit 2 Technical Specification P/T limits, LTOPS setpoints, and LTOPS enabling temperature remains conservative. In addition, after consideration of the North Anna Unit 2 Capsule W analysis results, it has been determined that the RT_{NDT} values assumed in the Reference (7) Technical Specification change request remain conservative.

CONCLUSIONS

The calculations described in Reference (2) were performed in accordance with applicable regulatory guidance. These calculations demonstrate that applicable regulatory criteria continue to be met, that affected design and licensing basis analyses remain valid, and that the RT_{NDT} values assumed in the Reference (7) submittal remain conservative.

The proposed UFSAR changes do not increase the probability of occurrence or consequences of accidents previously analyzed. The proposed changes update the North Anna Unit 2 PTS screening calculations performed in accordance with 10 CFR 50.61. The 10 CFR 50.61 PTS screening criteria are met for all

North Anna Unit 2 reactor vessel beltline materials. Therefore, the consequences of PTS events are not increased by the revised screening calculations. Reactor vessel material properties are not PTS event initiators. Therefore, the probability of occurrence of PTS events are not increased by the revised PTS screening calculation results.

The proposed UFSAR changes do not increase the possibility for an accident of a different type than previously identified in the Safety Analysis Report. The PTS screening calculations were performed in accordance with the methods prescribed by 10 CFR 50.61. None of the analysis parameters constitute new or unique accident initiators. Therefore, no possibility exists for creating an accident of a different type than previously analyzed in the Safety Analysis Report.

The proposed UFSAR changes do not reduce the margin of safety. ET-NAF-2000-0100 Revision 0 (2) demonstrates that the proposed revised analyses provide an acceptable margin of safety.

Required UFSAR changes are documented in UFSAR Change Request FN 2000-041.

00-SE-OT-53

Description

North Anna UFSAR Change Request No. FN 2000-023

UFSAR Change Request No. FN 2000-023 contains a list of changes, some of which are editorial in nature, which need to be corrected or clarified in the UFSAR sections that discuss North Anna's Reactor Thermal and Hydraulic Design System. This package is a result of the Integrated Configuration Management Project review of North Anna Power Station's Reactor Design system.

Summary

The above editorial/administrative changes are within the current design and licensing basis of the facility. These changes do not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed editorial/administrative changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

None of the proposed editorial changes involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial/administrative changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed editorial/administrative changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

With regard to the criteria set forth in 10 CFR 50.59 and based on the evaluation summarized above, the proposed editorial/administrative UFSAR changes have been determined not to represent an unreviewed safety question.

00-SE-OT-54

Description

North Anna Power Station submittal to the NRC for conversion of CTS (Current Technical Specifications) to ITS (Improved Technical Specifications

North Anna Power Station submittal to the NRC for conversion of CTS (Current Technical Specifications) to ITS (Improved Technical Specifications.

Summary

MAJOR ISSUES

The North Anna Plant Current Technical Specifications (CTS) are being converted to the Improved Technical Specifications (ITS) based on NUREG-1431, Revision 1 and subsequent changes. During the conversion process there will be numerous changes to the Limiting Conditions for Operation (LCOs) and the associated Actions and Surveillances. In addition, those CTS LCOs that do not meet the NRC criteria of 10 CFR 50.36(c)(2)(ii) for inclusion into the ITS are being relocated to the Technical Requirements Manual (TRM). The NRC has reviewed all of the generic changes resulting from a plant conversion to the ITS based on NUREG-1431 and found these changes not to be detrimental to the public health and safety. However, each plant that is converting to the ITS must evaluate each change in requirements for applicability to their plant design and justify these changes from an operational and safety standpoint. Each of these changes falls into one or more of the five categories:

1. Administrative: Those changes that do not change technical requirements,

2. More Restrictive: Those changes that result in added restrictions or elimination of flexibility,

3. Relocated: CTS LCOs or Specifications that do not meet 10 CFR 50.36 criteria and are relocated to licensee controlled documents,

4. Removed Detail: Those changes that remove certain details and information from Specifications and place them in the ITS Bases, the UFSAR, the TRM, or other documents under licensee control, or

5. Less Restrictive: Those changes that result in relaxed requirements, deleted requirements, or additional flexibility.

Each of these changes is fully discussed in the ITS submittal package that is sent to the NRC for their review and approval. The NRC will review the entire ITS submittal package and will then issue a Safety Evaluation Report (SER) prior to implementation.

The implementation of the ITS at NAPS will result in numerous changes that will affect many aspects of operations, procedures, training, maintenance, programs, etc. Due to the complex nature of the implementation, a North Anna Implementation Team is being assembled with members from the various departments or groups at the station and corporate. This team is tasked with identifying each individual change due to the conversion and determining how and who will accomplish these changes, and tracking them to completion.

Accident Probability and Consequences.

Could the activity increase the probability of occurrence for the analyzed accidents: The proposed Technical Specifications conversion to the ITS involves numerous changes and modifications to the

CTS LCOs including the relocation of CTS LCOs to the TRM. The LCOs being relocated to the TRM do not modify any technical requirements currently contained in their respective CTS LCOs. Each change to the converted CTS LCOs that are of a technical nature are classified as either More Restrictive or Less Restrictive in the ITS submittal package. Each of these changes are evaluated and justified to be acceptable for the safe operation of the plant. They do not result in operation that will increase the probability of initiating an analyzed event in that the requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the current safety analyses and within the guidelines established by the NRC in NUREG-1431. Therefore, these changes do not increase the probability of occurrence of any analyzed accidents.

Could the activity increase the consequences for the analyzed accidents: The proposed Technical Specifications conversion to the ITS does allow for extended outage times for certain equipment and also allows Modes changes with certain equipment inoperable. Therefore, there may be a slight increase in the consequences of analyzed accidents that could occur. However, the technical specifications will continue to ensure that the process variables, structures, systems, and components are maintained consistent with the current safety analyses and within the guidelines established by the NRC in NUREG-1431. As this change does result in an increase in the consequences for the analyzed accidents an Unreviewed Safety Question does exist and this change must be approved by the NRC.

Could the activity create the possibility for an accident of a different type than was previously evaluated: The proposed changes do not involve any physical modifications of the plant (no new or different type of equipment will be installed) or change the methods governing normal plant operation. The proposed changes do impose different requirements. However, the requirements are consistent with the assumptions in the safety analyses and within the guidelines established by the NRC in NUREG-1431. Therefore, these changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Margin of Safety as Described in Technical Specification Bases: As a result of the ITS conversion an entirely new Technical Specification Bases was developed. These new Bases have been evaluated to ensure that the LCO requirements are applied in the Modes and specified conditions assumed in the safety analysis. Therefore, these changes do not involve a reduction in the margin of safety.

00-SE-OT-55

Description
TSCR #382, Administrative changes to Facility Operating License, NPF-7.
TSCR #382 involves the removal of completed, redundant, expired, or otherwise non-applicable license conditions from the Facility Operating License for North Anna Unit 2. The change includes editorial corrections and the relocation of the various license conditions.

Summary
The proposed changes to the North Anna Unit 2 Facility Operating License (FOL), NPF-7, make minor editorial corrections, relocates three license conditions within the license itself, and removes completed, redundant, expired or otherwise non-applicable license conditions and provides a license document that is directly applicable to the current plant licensing and design bases. There is no safety significance associated with this proposed change since the change does not alter any currently applicable Facility Operating License requirements. Accordingly, the current North Anna Unit 2 licensing and design bases are unchanged and an Unreviewed Safety Question (USQ), as defined in 10 CFR 50.59, does not exists.

The proposed change to the North Anna Unit 2 Facility Operating License, NPF-7, is administrative (and in part editorial) in nature and neither station operations nor design are affected by the change. The removal of license conditions regarding completed (Facility Operating License (FOL) Sections, 2.C(3) through 2.C(21), and 2.F), or no longer needed (FOL Sections 2.C(9), 2.C(24), 2.D, and 2.H) and expired (FOL Section 2.C(2)(b)) requirements has no impact on plant operations since these requirements no longer have a legitimate means of being applied. The creation of a separate FOL section (2.C(3)) for "Additional Conditions" and the relocation within the Facility Operating License of the single Appendix C license condition to Section 2.C(3)b, the requirements for notification regarding changes to the plant effluent treatment systems to Section 2.C(3)a, and the fire protection requirements to Section 2.D does not alter the technical basis, requirements or the implementation of these items. The split of Section 2.C(2) to separately account for the amendments to the Technical Specifications and the Environmental Protection Plan for reasons of accuracy does not alter the intent of the original license condition. The impact of the re-design of the SG and RCP supports on the previously evaluated accidents was performed, approved and documented by the issuance of the license amendments for FOL Section 2.F. Removal of a license condition which refers to completed work and which has a design and licensing bases that is approved and documented in the UFSAR also does not alter station operation or the design of the affected components. The proposed change is within the current design and licensing bases of the facility. This change does not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. This change does not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These proposed changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed change to the North Anna Unit 2 Facility Operating License does not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

The license conditions that are being removed or relocated by this proposed change do not impact station operations or station equipment in any manner. The proposed change does not involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients that has not been previously analyzed. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed change to the North Anna Unit 2 Facility Operating License does not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. Since station operations are not affected by the proposed change, the change does not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed change to the North Anna Unit 2 Facility Operating License does not involve a reduction in any margin of safety described in the bases of the Technical Specifications.

00-SE-OT-56

Description
North Anna UFSAR Change Request No. FN 99-066
UFSAR Change Request No. FN 99-066 contains a list of changes, some of which are editorial in nature, which need to be corrected or clarified in the UFSAR sections that discuss or reflect aspects of North Anna's Plant Safety Analyses. This package is a result of the Integrated Configuration Management Project review of the North Anna Power Station Plant Safety Analyses.

Summary
The above editorial/administrative changes are within the current design and licensing basis of the facility. These changes neither affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed editorial/administrative changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

None of the proposed editorial changes involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial/administrative changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed editorial/administrative changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

With regard to the criteria set forth in 10 CFR 50.59 and based on the evaluation summarized above, the proposed editorial/administrative UFSAR changes have been determined not to represent an unreviewed safety question.

00-SE-OT-57

Description

This Safety Evaluation was performed in response to Plant Issue N-2000-1203-E1. The Plant Issue identified a non-conservative Technical Specification Surveillance Requirement, 4.7.7.1.d.1. The current Surveillance Requirement requires that the HEPA and charcoal differential pressure (D/P) be maintained less than 6 inches of water. Since the fan can only put out 5.2 inches of water, the existing system D/P is non-conservative, and a new value must be incorporated into the Technical Specifications. This Safety Evaluation is for ET N 00-057, Revision 1, which provides Engineering input to Licensing for the Technical Specification changes that are required.

Technical Specification Change Request Package No. 378

To revise Technical Specifications Surveillance Requirement 4.7.7.1.d.1 to decrease the maximum Control Room Emergency Ventilation System (CREVS) Filter Differential Pressure (Demister Filter, HEPA Filter and Charcoal Adsorber). The new D/P value in the Surveillance Requirement will be conservative, site-specific, and can be supported by the current system/fan configuration.

Summary

This activity supports Tech Spec Change Request No. 378. The proposed Tech Spec changes will modify the CREVS Filter Differential Pressure (D/P) limit in Surveillance Requirement (SR) 4.7.7.1.d.1. The existing limit is non-conservative, and does not represent the current, site-specific, system/fan configuration. The existing limit apparently came from a generic value that was identified in NUREG 0452. This value was identified in parentheses (e.g. (6)) to indicate that it was a generic value that should be replaced with a site-specific value, once a more appropriate value was determined. The Technical Specification value must be revised to ensure that the Control Room Emergency Ventilation System Units (1-HV-F-41 & 1-HV-FL-8, 1-HV-F-42 & 1-HV-FL-9, 2-HV-F-41 & 2-HV-FL-8, 2-HV-F-42 and 2-HV-FL-9) are operated such that their design basis function can be performed. This function is to minimize the radiation exposure of the Operators in the Control Room, and therefore maintain the exposures in accordance with General Design Criteria 19 (Appendix A of 10CFR50). This activity could not conceivably affect the environment, and has no affect on the ability to shut down the plant following a fire.

No changes are being made to the method of plant operations, except for providing a realistic D/P for controlling the Control Room Emergency Ventilation Units. Physical changes will be required upon approval and issuance of NRC issued Amendments to the Technical Specifications and prior to the effective implementation date of the Amendments. The required physical changes will only consist of adding D/P instruments across the demister filters. This will ensure that the revised Technical Specification requirements can be met.

The modifications required to add the differential pressure instrumentation for the control room emergency ventilation system demister unit will be implemented in accordance with the Design Change Program. The modification will also include enhancements to the method of flow control from the emergency ventilation units. These Tech Spec changes will not affect the probability or consequences of any accidents considered in Section 68 of this Safety Evaluation (Main Steam Line Break, Steam Generator Tube Rupture, Locked Rotor Accident, LOCA and Fuel Handling Accident). Also, these Tech Spec changes will not affect the consequences or probability of any malfunction considered in Section 69 of this Safety Evaluation (Electrical Separation, Single Failure Criteria, Environmental Qualification and Seismic Qualification). In addition, no new type of accident or malfunction is created by this activity, as no physical equipment is directly modified by this activity. In addition, this activity does not affect the margin of safety as described in the Bases section of the Technical Specifications.

The safety analysis assumes that the Control Room Emergency Ventilation System is able to meet it's design basis requirement of maintaining 0.04" water column differential pressure between the Control Room and the surrounding areas, at a flow rate of 1000 +/-10 % standard cubic feet per minute.

00-SE-OT-58

Description

North Anna UFSAR Change Request No. FN 2000-022

UFSAR Change Request No. FN 2000-022 contains a list of changes, some of which are editorial in nature, which need to be corrected or clarified in the UFSAR sections that discuss North Anna's Reactor Design - Miscellaneous System. This package is a result of the Integrated Configuration Management Project review of North Anna Power Station's Reactor Design system.

Summary

The above editorial/administrative changes are within the current design and licensing basis of the facility. These changes do not affect the initiators of analyzed events nor the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems, or components. These changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any event, and the successful functioning of at least one train or division of the equipment credited with mitigating the event. These changes do not impact the capability of the credited equipment to perform, nor is there any change in the likelihood that credited equipment will fail to perform. As a result, the proposed editorial/administrative changes to the UFSAR do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

None of the proposed editorial changes involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the proposed editorial/administrative changes to the UFSAR do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

Margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The above UFSAR changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the proposed editorial/administrative changes to the UFSAR do not involve a reduction in any margin of safety described in the bases for the Technical Specifications.

With regard to the criteria set forth in 10 CFR 50.59 and based on the evaluation summarized above, the proposed editorial/administrative UFSAR changes have been determined not to represent an unreviewed safety question.

00-SE-OT-59

Description

Temporary Shielding Request 00-TSR-037

It is proposed to install temporary lead blanket shielding on operable/operating 2" diameter Safety Related DG piping to provide protection to workers installing DCP 99-010 for replacement of SW lines in the Auxiliary Building basement. Each blanket weighs 15 lb./sq. ft. providing an additional loading of 15 lb. per linear foot of piping.

Summary

Temporary Shielding Request 00-TSR-037 will install seismically qualified lead blanket shielding on operable/operating 2" diameter Safety Related DG piping in the Auxiliary Building basement. The shielding will be installed and removed under the VPAP-2105 Temporary Shielding Program. The proposed activity will not:

- Increase the probability of occurrence for a seismic event,
- increase the consequences of a seismic event,
- create the possibility for an accident of a different type

No postulated equipment failures could be identified which are associated with temporary radiation shielding of this piping.

There are no applicable Technical Specification basis descriptions; therefore, the margin of safety will not be affected by the proposed activity. The proposed activity will be conducted in compliance with VPAP-2105 in order to satisfy Technical Specification 6.11. No changes to the Operating License or Technical Specification are required.

Applying 15 pounds of lead blanket shielding per linear foot around the outside of the 2" DG system piping will have no affect on any station operations following a fire.

There are no discernable environmental concerns associated with the proposed activity.

CONCLUSION

Installing temporary lead shielding as proposed does not constitute an unreviewed safety question because:

- The probability of experiencing a design basis seismic event is unrelated to the act of installing temporary lead shielding.
- Since the affected piping has been seismically analyzed for the applied shielding load and the response was found to be within acceptable limits, there is no identified increase in the consequences related to a design basis seismic event.

- Seismic concerns, the line temperature limit (for compatibility with lead blankets), and the need to avoid interference with the 1-DG-107 valve handwheel are the main issues associated with the proposed activity. These issues are addressed in the TSR. No other concerns were identified and an accident of a different type could not be postulated.
- Technical Specification margins of safety are not reduced. Seismic analysis has shown that the level of stress with shielding in place is within the design limits.

00-SE-OT-60

Description
Technical Specification Change Request No. 376A (Supplement to TSCR 376)
UFSAR Change Request FN 2000-048 (Supersedes FN 2000-016)
A supplement to Technical Specification Change Request (TSCR) No. 376 (TSCR 376A) and a revised UFSAR Change Request (FN 2000-048) are needed to address a NRC request for additional information (RAI) on TSCR 376. The NRC has requested consideration of pressure and temperature measurement uncertainties in the proposed revised design basis Reactor Coolant System (RCS) Pressure/Temperature (P/T) Operating Limits, Low Temperature Overpressure Protection System (LTOPS) Setpoints, and LTOPS Enable Temperatures. The NRC has requested inclusion of instrument uncertainties in order for them to grant an exemption to the requirements of 10 CFR 50 Appendix G to permit utilization of ASME Section XI Code Case N-640 (use of the Appendix A K_{Ic} fracture toughness curve, Figure A-4200-1).
This safety evaluation also supports a reduction in the Units 1 and 2 reactor vessel head bolt-up temperatures from 90°F to 60°F.

Summary
PURPOSE

This safety evaluation supports Technical Specification Change Request 376A, which supplements TSCR 376 (2). TSCR 376 and TSCR 376A propose revisions to the Technical Specifications to implement revised design basis analyses for the North Anna Units 1 and 2 Technical Specification Reactor Coolant System (RCS) Pressure/Temperature (P/T) operating limits, Low Temperature Overpressure Protection System (LTOPS) setpoints, and the LTOPS enable temperature (T_{enable}). TSCR 376A addresses a NRC Request for Additional Information (RAI) requiring incorporation of margin to accommodate pressure and temperature measurement uncertainties in the P/T limits and LTOPS setpoints. This safety evaluation also supports implementation of a revised reactor vessel head bolt-up temperature. Although the revised reactor vessel head bolt-up temperature does not require NRC review and approval for implementation, this change will be implemented as part of the TSCR 376A Action Plan.

DISCUSSION

TSCR 376A

A Technical Specification Change Request (TSCR) concerning the North Anna Units 1 and 2 RCS pressure/temperature (P/T) limits and low temperature overpressure protection system (LTOPS) setpoints was submitted to the NRC on June 22, 2000 (2). The basis for this TSCR is described in Reference (3). The objective of the submittal was to justify continued use of the existing Technical Specification P/T limits and LTOPS setpoints on the basis of a margin assessment. The margin assessment required an exemption to the requirements of 10 CFR 50 Appendix G to permit application of ASME Section XI Code Case N-640. N-640 supports use of the ASME Section XI Appendix A K_{Ic} fracture toughness curve (Figure A-4200-1), instead of the ASME Section XI Appendix G K_{Ia} curve (Figure G-2210-1) that was employed in the development of the existing Technical Specification P/T limits and LTOPS setpoints. During a November 7, 2000 teleconference, NRC staff indicated that application of margin to accommodate pressure and temperature measurement uncertainties would be required in order for this exemption request to be granted. Therefore, it became necessary to supplement the Reference (2) submittal with an evaluation of the effects of incorporating pressure and temperature measurement uncertainties into the proposed design basis P/T limits.

As demonstrated in Reference (1), the existing Technical Specification LTOPS setpoints remain conservative and valid to 32.3 EFPY and 34.3 EFPY for North Anna Units 1 and 2, respectively, after application of pressure and temperature measurement uncertainties to the LTOPS design basis P/T limit curve. However, the conservatism of the existing Technical Specification P/T limits could not be confirmed. Therefore, the proposed revised design basis P/T limits, including allowances for pressure and temperature measurement uncertainty, must be incorporated into the Technical Specifications and supporting operating procedures.

Revised Reactor Vessel Minimum Bolt-Up Temperature

The current design and licensing basis composite RCS pressure/temperature operator curves for North Anna Units 1 and 2 [4] [5] include a minimum reactor vessel head bolt-up temperature of 90°F. This bolt-up temperature was designed to conservatively bound the highest reactor vessel flange RT_{NDT} value, including allowance for the effects of temperature measurement uncertainty. ASME Section III Paragraph G-2222(c) provides recommendations for the bolt-up temperature, indicating that the temperature of the stressed region (i.e., the vessel and closure head flanges) must be greater than the limiting RT_{NDT} value of the stressed materials. As documented in UFSAR Tables 5.2-26 and 5.2-27, and in Reference (6), the highest reactor vessel flange or closure head flange RT_{NDT} value for the North Anna Units 1 and 2 is –22°F (vessel flange materials).

As documented in Reference (7), Westinghouse developed a generic minimum bolt-up temperature of 60°F, based on an evaluation of available flange RT_{NDT} values for Westinghouse-designed plants. Because (a) the RT_{NDT} values for the North Anna Units 1 and 2 vessel flanges and closure head flanges are all well below 40°F, and (b) the RCS wide range temperature measurement uncertainty is less than 20°F (8), a revised reactor vessel bolt-up temperature of 60°F is being implemented by the attached safety evaluation. The revised vessel bolt-up temperature will be implemented as part of the Action Plan for TSCR 376A.

COMMITMENT EVALUATION SUMMARY

Original Commitment Description: The Virginia Power response to NRC Generic Letter 91-06, Resolution of Generic Issue A-30, Adequacy of Safety-Related DC Power Supplies, Pursuant to 10 CFR 50.54(f), dated October 28, 1991, provided specific details associated with the design of the station DC system. Specifically, the response stated that each division of the station DC system has the following separate, independently annunciated alarms in the control room:

- Battery charger AC input breaker "open",
- DC output breaker "open",
- DC system ground,
- DC bus undervoltage,
- DC bus overvoltage, and
- Battery charger failure.

The response also stated that there was not a separate, independently annunciated alarm in the control room for battery discharge, but the other alarms described above would alert operators that the batteries are being discharged.

Also, the response stated that the emergency diesel generator (EDG) DC system has the following separate, independently annunciated alarms in the control room:

- DC bus undervoltage,
- DC bus overvoltage, and
- Battery charger failure.

In addition, the response to the Generic Letter stated that testing is performed on the DC systems. It could be implied that testing of alarm functions was also performed.

Clarification of the response to NRC Generic Letter 91-06 is required because the station DC system and the EDG DC system do not include separate, independently annunciated alarms in the control room for the above listed conditions, annunciator response procedures are not provided for each of the above listed conditions, and testing of alarms was not being performed at the time of the Generic Letter response. Additional information on these clarifications is provided below.

Source Document: Response to NRC Generic Letter 91-06

Revised Commitment Description: The purpose of this commitment evaluation is to clarify the response to NRC Generic Letter 91-06. This is because the station DC system and the EDG DC system do not include separate, independently annunciated alarms in the control room as described above. The Generic Letter response should have indicated that alarm functions associated with the station DC system are referenced in UFSAR Section 8.3.2.1. UFSAR Section 8.3.2.1 states that "a condition of high output voltage, low supply voltage, low output current, low output voltage, or ground fault of a particular battery charger will activate an alarm on the main control room board annunciator." Also, the alarm functions associated with the EDG DC system are tied to a single trouble alarm for the "J" and "H" battery systems and the alarm is actuated by either low or high EDG DC system voltage. No battery charger failure alarm exists. Transducers provide the DC voltage input to a recorder in the main control room that actuates the alarm. These transducers are calibrated periodically to ensure accurate voltage readings are sent to the main control room. Additionally, the EDG battery systems are routinely tested to ensure operability.

The Generic Letter response also indicates that annunciator response procedures are provided for each of the conditions listed above. However, annunciator response procedures are only provided for "Station Battery Voltage Trouble", "Battery Charger Trouble" (for each division of battery chargers), "Vital Bus Inverter Trouble" (for each division), and "Emergency Diesel Generator Battery Voltage Trouble."

In addition, the response to the Generic Letter implies that testing is performed on the DC systems. This statement of testing could imply that the trouble alarms are also tested. At the time of the Generic Letter response, the alarm functions associated with the trouble alarms were not tested. However, procedures have now been developed to test the alarm functions associated with the station DC system and EDG DC system trouble alarms.

Summarize Justification for Change: The Virginia Power response to NRC Generic Letter 91-06 requires clarification associated with the control room alarms for the station DC system and EDG DC system. The clarification is acceptable because:

1) The UFSAR describes that DC system status is continuously displayed in the control room and a periodic visual check is made of the equipment. Each battery distribution switchboard is provided with an isolating transducer that feeds a battery voltage recorder in the control room. In addition, voltmeters are provided on the main board to indicate switchboard voltage. A condition of high output voltage, low output current, or ground fault of a particular battery charger will activate an alarm on the control room board annunciator. Also, the UFSAR describes several conditions that will render the EDGs incapable of responding to an emergency start signal. These conditions are annunciated either directly based on the situation or indirectly via a start failure alarm in the diesel generator room and via an EDG trouble alarm in the control room. Transducers provide the EDG DC voltage input to a recorder in the main control room that actuates the alarm. These transducers are calibrated periodically to ensure accurate voltage readings are sent to the main control room. Additionally, the EDG battery systems are routinely tested to ensure operability.

2) The trouble alarms are also acceptable because they inform the control room operators that there is a DC system problem. Upon receipt of a trouble alarm, the control room operators would implement the necessary actions to resolve an alarm

condition in accordance with approved annunciator response procedures. In addition, operator rounds would detect any local alarms in the EDG rooms.

3) Operation with only DC system trouble alarms in lieu of separate, independently annunciated alarms in the control room has occurred for over 20 years for North Anna Units 1 and 2. No detrimental effects have been observed with this method of operation.

4) Procedures have now been developed to test the alarm functions associated with the station DC system and EDG DC system trouble alarms.