VIRGINIA ELECTRIC AND POWER COMPANY

RICHMOND, VIRGINIA 23261

April 3, 1992

United States Nuclear Regulatory Commission
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Washington, D.C. 20555

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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(b)(2), enclosed is a summary description of facility changes, tests and experiments, including a summary of the safety evaluations, that were conducted at North Anna Power Station during 1991.

If you have any questions, please contact us.

Very truly yours,

W L Stewart

W. L. Stewart
Senior Vice President - Nuclear

Enclosure

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IE47
1/1

1991 50.59 SAFETY EVALUATION REPORTABLE TO NRC

JCO'S

91-SE-JCO-001
-002
-003
-004
-005

1991 50.59 SAFETY EVALUATIONS REPORTABLE TO NRC

JUMPERS

91-SE-JMP-002
-003
-004
-005
-006
-007
-008
-009
-011
-013
-015
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91-SE-JMP-035
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-061

1991 50.59 SAFETY EVALUATIONS REPORTABLE TO NRC

MODIFICATIONS

91-SE-MOD-001	91-SE-MOD-026	91-SE-MOD-050
-002	-027	-051
-003	028	-052
-004	-029	-053
-005	-030	-054
-006	-031	-055
-007	-032	-056
-008	-033	-057
-009	-034	-058
-010	-035	-059
-011	-036	-060
-012	-037	-061
-013	-038	-062
-014	-039	-063
-015	-040	-064
-016	-041	-065
-017	-042	-066
-018	-043	-067
-019	-044	-068
-020	-045	-069
-021	-046	-070
-023	-047	-071
-024	-048	-072
-025	-049	-073
		-074
		-075
		-076
		-077
		-078

1991 50.59 SAFETY EVALUATIONS REPORTABLE TO NRC

OTHERS

91-SE-OT-001
-002
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-012
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-023

91-SE-OT-025
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-041
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-047

91-SE-OT-048
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1991 50.59 SAFETY EVALUATIONS REPORTABLE TO NRC

PRIOR TO USE (PTU)

91-SE-MOD-001
-002
-003
-004
-005
-006
-010
-011
-012
-013

SAFETY EVALUATION NUMBER 91-SE-JCO-001

DESCRIPTION

A link from the refueling transfer cart drive chain was dropped and fell into the refueling purification system piping.

The purpose for this change is to allow operation with the link within the piping of the refueling purification system. The section of piping where the link fell is only used during fuel transfer operations. If the link migrated to the RP filters it will be trapped there.

SAFETY ANALYSIS SUMMARY

This issue is acceptable because the link cannot cause damage to any safety related component. There is no unreviewed safety question because the location and size of the link prohibit it from damaging or inhibiting the operation of any safety related component. The high flow rates experienced in the system during performance of the HHSI flow balance should have been sufficient to carry the link out to the RP filter.

SAFETY EVALUATION NUMBER 91-SE-JCO-002

DESCRIPTION

Several Recirculation Spray Heat Exchanger's Isolation MOV's (Units 1 & 2) were found to have high dynamic torque. JCO 91-02 was written to justify that the valves operated during a CDA event.

The JCO evaluated the concerns and determined the affected valves can be operated properly during a CDA event.

SAFETY ANALYSIS SUMMARY

This safety analysis assumes that during post-CDA recovery actions, an operator requires isolating or unisolating a Recirculation Spray Heat Exchanger. A guideline to operate the Recirculation Spray Heat Exchanger isolation MOV's will be in place.

The Recirculation Spray Heat Exchanger isolation MOV's will operate as required during the initial phase of a CDA event. The ability to isolate a leaking Heat Exchanger or to unisolate a Heat Exchanger which was isolated by mistake still exists.

SAFETY EVALUATION NUMBER 91-SE-JCO-003

DESCRIPTION

Justification for continued operation 91-03 evaluates the minor leakage of service water due to pitting corrosion in the encased in concrete portion of the Service Water piping to/from the Unit 2 CR chillers.

The purpose of the JCO is to specify the required compensatory action for the continued operation of Units 1 & 2 with the breach of piping integrity of the encased in concrete Service Water lines to CR chillers for Unit . Also, an action plan correcting the breach of Service Water piping integrity will be established.

SAFETY ANALYSIS SUMMARY

With the leakage described, the service water system components remain capable of providing the required flow to safety related equipment under design basis accident conditions. The seismic integrity of the piping is preserved. Safety related structures in the area have been evaluated under the conditions outlined and are unaffected.

Operability of safety related components (as defined by the ability to perform the intended safety function) remains unaffected. The monitoring of minor SW leakage does not create the possibility for a different type of accident nor is the minor leakage of sufficient magnitude to have any impact on the functional capability of safety related components.

The redundancy in the SW system has not been compromised and therefore, the ability of either header to perform the required safety function under design basis conditions (assuming complete loss of one header has been maintained). The capacity of the ultimate heat sink to maintain a 30 day supply of water following the DBA with no allowance for make-up water is preserved. The ability of the control room compressed air system to perform its safety function has not been degraded. The ability to monitor radioactive contamination of the SW system is maintained.

SAFETY EVALUATION NUMBER 91-SE-JCO-004

DESCRIPTION

Missing plastic ear protector of a sound powered phone headset and a high rad door key was lost in the containment.

The JCO was prepared to justify safe operation of safety related equipment in the containment due to debris left in the containment.

SAFETY ANALYSIS SUMMARY

A plastic ear protector of a sound powered phone headset and a high rad door key was lost in the containment. Should these pieces be washed down to the containment sump, they would have no effect on the operability of the safety injection system or the recirculation spray system because of the pump screen design. The screen system, as detailed on UFSAR Figure 6.2-79, is designed to prevent passage of particles larger in size than the smallest restriction in the Recirculation Spray system (i.e., spray nozzles). Per UFSAR section 6.2.2.2, particles of this size would have no effect on Low Head Safety Injection or Recirculation Spray pump operation. The amount of hydrogen that may be generated by the decomposition of the ear piece or the key is not significant.

SAFETY EVALUATION NUMBER 91-SE-JCO-005

DESCRIPTION

Justification for continued operation 91-03, Rev. 1 evaluates the minor leakage due to pitting corrosion in the concrete encased 4" and possibly 24" SW and Aux. Service Water piping. The minor leakages observed in Instrument Rack Room Unit 2, Service Building, Unit 1 Turbine Building Condensate Pump Pit, and Auxiliary Building Piping Tunnel, South of Manways.

The purpose of this JCO is to specify the required compensatory action for the continued operation of Units 1 & 2 with the breach of piping integrity of the concrete encased 4" SW lines to CR chillers Unit 2 and possibly 24" diameter concrete encased portion of SW and Aux. Service Water lines. Also, an action plan correcting the breach of SW piping integrity will be established.

SAFETY ANALYSIS SUMMARY

With the leakage described, the service water system components remain capable of providing the required flow to safety related equipment under design basis accident conditions. The seismic integrity of the piping is preserved. Safety related structures in the area have been evaluated under the conditions outlined and are unaffected.

Operability of safety related components (as defined by the ability to perform the intended safety function) remains unaffected. The monitoring of minor SW leakage does not create the possibility for a different type of accident nor is the minor leakage of sufficient magnitude to have any impact on the functional capability of safety related components.

The redundancy in the SW system has not been compromised and therefore, the ability of either header to perform the required safety function under design basis conditions (assuming complete loss of one header has been maintained). The capacity of the ultimate heat sink to maintain a 30 day supply of water following the Design Basis Accident with no allowance for make-up water is preserved. The ability of the control room compressed air system to perform its safety function has not been degraded. The ability to monitor radioactive contamination of the SW system is maintained.

SAFETY EVALUATION NUMBER 91-SE-JMP-002

DESCRIPTION

Install temporary Gaitronics in the Auxiliary Building Penetration Area.

To provide communications via Gaitronics to and from the Auxiliary Building Penetration Area. This will eliminate the current practice of having to exit a contaminated area to use the Gaitronics.

SAFETY ANALYSIS SUMMARY

A temporary Gaitronics will be installed in the auxiliary building penetration area to provide communications via Gaitronics to the area. The purpose is to eliminate the need to exit a contaminated area to use the Gaitronics.

The temporary Gaitronics will apply a negligible load to the Vital Bus.

Fuses will prevent feedback of an electrical fault into the permanent Gaitronics or the Vital Bus.

The temporary Gaitronics will be mounted adequately so that it is not a seismic concern.

SAFETY EVALUATION NUMBER 91-SE-JMP-003

DESCRIPTION

Temporarily install a hose from the primary drain transfer pump discharge to a) hot legs via HHSI, b) normal charging, or c) the Reactor Purification System via Cavity suction.

To conserve reactor coolant system (RCS) inventory by recovering leakage past the Loop Stop Valves and returning it to the RCS or the Refueling Cavity.

SAFETY ANALYSIS SUMMARY

The jumper is adequately rated for service conditions.

Check Valve will prevent backflow into the primary drain transfer tank (PDTT) or the primary drain system to the Stripper.

The jumper location and arrangement are such that the RHR System and RHR Pump operation are not adversely affected.

SAFETY EVALUATION NUMBER 91-SE-JMP-004

DESCRIPTION

Jumper to defeat the unit 2 annunciator because of the load shed switch being in the defeat position. (Annunciator H panel, window E8)

Annunciator to be cleared in accordance with the "black board" concept. Redundant indication is already provided by the corresponding annunciator on the unit 1 panel.

SAFETY ANALYSIS SUMMARY

[illegible] only defeat a single input to a single redundant annunciator. The operati[illegible] plant systems and components is otherwise unaffected. Should an [illegible] coincident with a loss of offsite power, the ability of the Emergency [illegible]ors to respond to the event is unaffected. The protection afforded by [illegible] scheme is not required while unit 1 is shutdown, but if it were to be [illegible]ce, even with the jumper in place, the system would function just as [illegible] the inputs to the annunciator from an actual overload condition on a reserve station service bus are not defeated, and the annunciator will still be capable of alarming should such a condition occur. Since the change involves only a redundant annunciator, there is no potential for causing a different type of accident or malfunction, nor is there any increase in the potential of a previously analyzed accident or malfunction to occur.

SAFETY EVALUATION NUMBER 91-SE-JMP-005

DESCRIPTION

Attach a portable generator to provide power to the vacuum primary system level control valve, 2-LCV-VP-201.

Bus 1A1-3 is the normal power supply for valve 2-LCV-VP-201. However, bus 1A1-3 is temporarily de-energized for maintenance.

SAFETY ANALYSIS SUMMARY

The proposed jumper provides a temporary power source to supply the vacuum primary system level control valve 2-LCV-VP-201. The normal power supply for this valve is 1A1-3, however this bus was removed from service for maintenance. The proposed jumper is simple and does not affect any other station power supplies. No safety systems are affected by operation of 2-LCV-VP-201. Therefore, no unreviewed safety question can exist and the jumper should be allowed.

SAFETY EVALUATION NUMBER 91-SE-JMP-006

DESCRIPTION

Electrical jumper to bypass the bearing oil and bearing lift pressure switches in the start circuit of the turning gear motor.

The turbine shaft needs to be jogged over one or two revolutions to allow NDE inspections of various components.

SAFETY ANALYSIS SUMMARY

Operation of the turning gear motor to jog over the turbine for inspections has no bearing on the capability of safety systems to respond to any accident. While the unit is shutdown, the turbine cannot contribute to causing such an accident since it has no mechanism for removing energy from the RCS. The inspections are necessary to ensure that the turbine is structurally sound and that no excessive potential exists for the generation of turbine missiles.

SAFETY EVALUATION NUMBER 91-SE-JMP-007

DESCRIPTION

Jumpers are to be installed to temporarily provide power to certain 480 volt loads from the opposite emergency bus during 480 volt bus outages for maintenance. The loads are 1) EDG battery charger and room lighting, 2) the semi-vital bus, and 3)4) both vital busses. When the respective 480 volt bus is de-energized, power will be supplied to these loads from the opposite emergency bus via the jumpers and breakers which are either not in service or have enough reserve capacity to handle the additional loads.

To maintain a power supply for the essential loads listed above while the remainder of the 480 volt bus is de-energized for maintenance.

SAFETY ANALYSIS SUMMARY

Technical Specifications require certain equipment powered by the emergency bus to remain OPERABLE during shutdown. Breakers being used to supply power to the alternate bus are sized to afford equivalent protection as the original power supply. This ensures that any faults are isolated by that breaker such that the OPERABLE (other units') power supply remains unaffected. Controls are adequate to ensure jumper removal is performed prior to MODE 4.

SAFETY EVALUATION NUMBER 91-SE-JMP-008

DESCRIPTION

Installation of temporary (non-seismic) pump and associated piping and valves for removing service water (SW) from a SW header which is out of service.

To facilitate SW header outages.

SAFETY ANALYSIS SUMMARY

Use of a temporary pump arrangement with flexible connections to facilitate SW header outages is acceptable.

The arrangement will be leak-tested after installation to ensure integrity. Flexible connections will be used to ensure no adverse effect on the SW system as result of a seismic event. Installation of pipe caps will be verified after removal to ensure SW system integrity.

SAFETY EVALUATION NUMBER 91-SE-JMP-009

DESCRIPTION

Temporarily install shorting screws in the Main Generator Current transfer blocks.

To prevent protective relay actuation during Main Generator testing.

SAFETY ANALYSIS SUMMARY

The reactor will be shutdown during the Main Generator Testing.

The Main Generator will not be connected to the Grid, nor will it be powered by the Main Turbine during this testing. Without steam flow, there is no reactivity feedback.

High Potential Testing is performed on the Main Generator to verify the integrity of the insulation.

Installation of shorting screws in Main Generator current transformer blocks places the current transformers at the same potential to prevent protective relay actuation during Main Generator high potential testing.

This action enables the testing to proceed without interruption by the protective circuitry.

The bypassing of Main Generator protective circuitry during testing while the reactor is shut down and not providing thermal power for steam generation cannot possibly cause, contribute to, or increase the consequences of a Design Basis Accident or any new accident.

The Main Generator is non-safety related. No other equipment's protective circuitry will be altered. Technical Specifications do not reference this circuitry, equipment, or testing.

SAFETY EVALUATION NUMBER 91-SE-JMP-011

DESCRIPTION

Installation of a temporary sump pump in the Unit 1 rack room.

Normal sump pumps, 1-DA-P-9A and 9B, are inoperable.

SAFETY ANALYSIS SUMMARY

The temporary sump pump installed in the Unit 1 rack room sump will provide the same function as the normal sump pumps. The temporary sump will make it easier on operations by not requiring manual pumping of sumps on a regular basis.

No unreviewed safety question exists because the temporary sump pump does not interface with any safety systems, high energy systems, or safety electrical systems.

SAFETY EVALUATION NUMBER 91-SE-JMP-013

DESCRIPTION

Mechanically block the containment sump pump discharge line trip valve, 1-DA-TV-100B, in the open position.

The inside containment isolation valve for the containment sump pump discharge line is broken. Parts will not be available for a few days.

SAFETY ANALYSIS SUMMARY

The jumper is to install a mechanical block to keep 1-DA-TV-100B open while parts are ordered for valve repair. The jumper is routine in nature and provides a critical function for operations during the outage.

No unreviewed safety question exists because containment isolation is not required below Mode 4. In addition, 1-DA-TV-100A will still provide isolation of the penetration as needed.

SAFETY EVALUATION NUMBER 91-SE-JMP-015

DESCRIPTION

Remove suction strainer for lube oil pump, 1-LO-P-1, and install jumper to Westinghouse Temporary Lube Oil Conditioning Unit.

To facilitate flushing of Lube Oil System through the Westinghouse Temporary Lube Oil Conditioning Unit.

SAFETY ANALYSIS SUMMARY

The Main Lube Oil System is non-safety related.

This is a simple mechanical jumper to allow flushing and cleaning of the Lube Oil System with a Westinghouse Temporary Lube Oil Purification Unit.

The jumper is rated for this application and is verified leak-tight before and after removal/installation.

Oil spill contingencies and precautions will be observed.

SAFETY EVALUATION NUMBER 91-SE-JMP-016

DESCRIPTION

Temporarily jumper around the instrument air solenoid operated valve (SOV) which maintains safety injection trip valve, 1-SI-TV-100, open. This is to ensure that 1-SI-TV-100 stays open at all times to ensure that N_2 is supplied to containment.

The SOV on 1-SI-TV-100 has failed.

SAFETY ANALYSIS SUMMARY

The affected penetration will be declared INOPERABLE in accordance with Tech Spec 3.6.1.1. If containment integrity is required to ensure that an ACTION STATEMENT is completed, the jumper will be removed. The jumper will be removed prior to Mode 4. Therefore, there is no unreviewed safety question since containment integrity is not required for the period while the jumper is installed. The jumper will increase the overall safety of the plant by ensuring that N_2 is available to containment.

SAFETY EVALUATION NUMBER 91-SE-JMP-018

DESCRIPTION

Temporarily jumper out the pump motor permissives on the high pressure heater drain pumps, 1-SD-P-1A, B, & C.

Temporarily jumper out the pump motor permissives to run motor uncoupled for post maintenance testing.

SAFETY ANALYSIS SUMMARY

All permissives jumpered out will be returned to operable status prior to returning the equipment to service. Running the motor uncoupled from the pump will not affect system performance and will not adversely affect electrical bus parameters. Required surveillances will be performed prior to declaring equipment operable.

Because the system (both electrical and mechanical) will not be adversely affected, no unreviewed safety question exists.

SAFETY EVALUATION NUMBER 91-SE-JMP-019

DESCRIPTION

Temporarily install 1" OD stainless steel tubing from moisture separator/reheater 1/2" Hydro Vent Valve to the steam dump system vent valve 3/4"-1-SD-827.

To determine whether or not the level column is being adequately vented, thus being the cause of the high level indication alarm.

SAFETY ANALYSIS SUMMARY

The jumper will be installed between the 1/2" Hydro Vent Valve and level column vent valve 1-SD-827. The purpose is to see if the level column is being adequately vented, which could cause the high-level indication/alarm. The jumper will not alter the function of the MSR, nor will its failure cause the MSR to malfunction. Should failure occur, jumper can be isolated with existing valves. Jumper does not require changes to Tech Specs or UFSAR.

SAFETY EVALUATION NUMBER 91-SE-JMP-020

DESCRIPTION

Jumper Out Pump/Motor Permissives for pump 1-SD-P-2B.

The purpose for this change is to run the motor uncoupled for post maintenance testing.

SAFETY ANALYSIS SUMMARY

All permissives jumpered out will be returned to oper[illegible] status prior to returning the equipment to service. Running the motor uncoupled from the pump will not affect system performance and will not adversely affect electrical bus parameters. Required surveillances will be performed prior to declaring equipment operable.

Because the system (both electrical and mechanical) will not be adversely affected, no unreviewed safety question exists.

SAFETY EVALUATION NUMBER 91-SE-JMP-021

DESCRIPTION

Disabling the low tank level switch for the 1H diesel day tank. The level switch is malfunctioning and causing a continuous alarm when tank level is adequate.

The switch is not operating properly. A Work Order has been submitted to correct the switch problem, but at present, no replacement parts are available. The malfunctioning switch is causing a continuous alarm when the actual tank level is adequate.

SAFETY ANALYSIS SUMMARY

The automatic makeup control system to the tank is unaffected.

TS requires monthly surveillance of level, not continuous, operators verify level once every 12 hours on logs.

By clearing this annunciator, operators will not be distracted from other potential problems associated with this EDG

The EDG's ability to perform its intended safety function is unaffected.

SAFETY EVALUATION NUMBER 91-SE-JMP-022

DESCRIPTION

Installation of a temporary video camera in containment to monitor 'A' RCP lower lube oil reservoir.

The lower lube oil reservoir for 'A' RCP is low due to a minor oil leak.

SAFETY ANALYSIS SUMMARY

Use of the camera will not result in any performance characteristics being changed. The camera is small (<5 lbs) and can easily be restrained. In the event of a CDA during use of this camera, the design of the sump screens is such that LHSI/RS performance will not be adversely affected. Material consideration have been adequately evaluated for use of the camera in containment.

SAFETY EVALUATION NUMBER 91-SE-JMP-023

DESCRIPTION

Defeat alternator circuit on fuel building sump pumps.

The purpose for this change is to allow maintenance on pump 1-DA-P-2A while pump 1-DA-P-2B is allowed to function normally in "auto."

SAFETY ANALYSIS SUMMARY

Section 9.3.3.5 of the UFSAR states that the fuel building sump pumps are full sized pumps, and therefore, are designed to handle 100% of expected sump inleakage.

The alternator circuit merely provides equal wear on both pumps. Bypassing the alternator circuit while maintenance is being performed on one pump allows the other pump to perform its design function in "auto", thereby reducing control room nuisance alarms and eliminating unnecessary operator action to manually operate the pumps.

Chapter 15 accidents or malfunctions are unaffected by bypassing the alternator circuit.

SAFETY EVALUATION NUMBER 91-SE-JMP-024

DESCRIPTION

Domestic Water will be used as a water source for a temporary shower to be used by asbestos workers. The temporary shower is located in a trailer outside the Unit 2 Turbine Building (west side).

To provide a shower for asbestos workers.

SAFETY ANALYSIS SUMMARY

No Unreviewed Safety Question exists since Domestic Water is not Safety Related. The Domestic Water jumper runs through a hose to a trailer for the purposes of providing a shower for asbestos workers. This is a normal load for the Domestic Water system. The shower drain is passed through filters in the trailer specially designed to capture asbestos wastes. By installing this jumper, the health and safety of the asbestos workers is ensured.

SAFETY EVALUATION NUMBER 91-SE-JMP-025

DESCRIPTION

Removal of Alarm Card in "Q" Amplifier for Main Turbine Generator #8 Bearing Vibration Sensor.

To defeat the input from the #8 Bearing to the Main Control Board Annunciator For Main Turbine vibration.

SAFETY ANALYSIS SUMMARY

No unreviewed safety question exists since:

1. The Main Turbine Vibration Monitoring System is not Safety-Related.
2. The #8 Bearing vibration sensor is one of many in the system. Defeating its input will not alter performance characteristics of the turbine generator or any of the other bearing vibration sensors.
3. The #8 Bearing vibration sensor is not functioning.
4. Main Turbine vibrations are monitored regularly by predictive analysis locally at the bearings.
5. If an actual vibration condition arose, other bearing vibration sensors would sense the common shaft's vibration.
6. Bearing temperature monitoring is functional on the #8 bearing.

The change should be allowed because:

1. Failure to defeat the #8 Bearing input to the Main Control Board Annunciator associated with Main Turbine Vibration will only result in sporadic, invalid alarms.

2. While the Main Turbine Vibration Monitoring System is not safety-related, an invalid alarm is distracting to the operators and should be defeated to allow the valid sensors to annunciate.

3. The alarm caused by the #8 Bearing prevents other turbine bearings' vibrations from alarming on the main control board.

4. Operator awareness and sensitivity to valid alarms is enhanced by the defeat of the #8 Bearing Vibration input to the Main Control Board Annunciator.

SAFETY EVALUATION NUMBER 91-SE-JMP-026

DESCRIPTION

Install temporary support to maintain the main turbine turning gear latched.

The turbine sometimes jumps off of the turning gear due to tolerance problems in the gear teeth. This jumper will allow the turbine to maintain on the gear as required to ensure even cooldown and to prevent bowing of the turbine shaft.

SAFETY ANALYSIS SUMMARY

This jumper is a simple mechanical device used to maintain the turbine turning gear lever engaged. Maintaining the turbine on the turning gear while the turbine is warm will ensure even cooldown and prevent turbine shaft bowing. The turbine is off-line and isolated from the steam supply system whenever the turning gear system would be placed in service. Therefore, the possibility of an accident occurring due to the jumper does not exist. The turning gear is not safety related (although the turning gear oil pump is powered from the 'H' emergency bus) and is not required to function to mitigate the consequences of any accidents or malfunctions of equipment important to safety. The loading on the emergency bus is not affected by the proposed jumper.

SAFETY EVALUATION NUMBER 91-SE-JMP-027

DESCRIPTION

Remove cover from the stator cooler for reactor coolant pump (RCP) 1-RC-P-1B to increase the RCP stator cooling flow.

To provide adequate air flow through the 1-RC-P-1B stator. Temperatures are currently elevated.

SAFETY ANALYSIS SUMMARY

This jumper increases the stator cooling flow for 1-RC-P-1B, thus ensuring the continued safe operation of the pump. The only safety function that the RCP motor provides is inertia for RCP coastdown. This jumper does not affect the mass or inertia of the RCP. The jumper does not alter the component cooling system in any way. The jumper will reduce the cooling to the RCP motor cubicle room. However, fan 1-HV-F-92B mixes the air in the cubicle with the air in the containment dome area. Because the air is thoroughly mixed with the containment atmosphere, the weighted average containment temperature will be affected. However, this parameter is closely monitored by the Control Room Operator, and actions may be taken to adjust this temperature. Furthermore, the Containment Air Recirculation Fans provide the bulk of the cooling to the containment environs, so the actual change to the containment ambient will be small and can be controlled by the Control Room Operator. As such this jumper does not constitute an unreviewed safety question.

SAFETY EVALUATION NUMBER 91-SE-JMP-028

DESCRIPTION

Energize control power for instrument air compressor 1-IA-C-1 for a technical representative while compressor is tagged out. See jumper form for details.

Allow for post maintenance testing/troubleshooting.

SAFETY ANALYSIS SUMMARY

The jumper will only energize the control circuit for 1-IA-C-1. Power for the compressor is tagged out and the jumper will verify no voltage present in backfeed to the compressor.

SAFETY EVALUATION NUMBER 91-SE-JMP-029

DESCRIPTION

Jumper IA to liquid waste 1-LW-PCV-109 and 1-LW-TCV-111 to close valves.

Controllers for valves are not operable and the valves fail open. The valves are desired to be shut to prevent draining unnecessary amounts of component cooling from the system to work on component cooling valve 1-CC-RV-119.

SAFETY ANALYSIS SUMMARY

The valves will be the second isolation for the tagout boundary and high flow stop valves will be installed in the IA line to prevent loss of the IA system if the jumpers fail.

These valves provide component cooling flow to 1-LW-E-2 and 3 (waste evaporator overhead condensers), which will be tagged out (component cooling side) during this evolution. Therefore, the valves are not needed to be in service.

The UFSAR states that the waste evaporator is not used.

SAFETY EVALUATION NUMBER 91-SE-JMP-030

DESCRIPTION

Tie wrap has been utilized on the turning gear for the main turbine to hold the handle to the engage position.

The turning gear is not staying engaged, and this mechanism is required to ensure that the turbine continues to roll in order to prevent wrapping the rotor.

SAFETY ANALYSIS SUMMARY

Operation of the turbine on the turning gear does not create the potential for any accident or major malfunction of any safety related equipment. It is desirable to keep the turbine on the turning gear to ensure that the rotor does not experience bowing. This will further ensure that the turbine rolls true when the unit is returned to power, and minimizes the chance of creating a turbine missile. The turning gear is not safety related, is not a Tech Spec consideration, and is not used to respond to any accident or malfunction of safety related equipment. The turning gear motor rotates the turbine at such a slow speed as to be inconsequential.

SAFETY EVALUATION NUMBER 91-SE-JMP-031

DESCRIPTION

Use an air jumper to bypass the solenoid operated valves (SOVs) that control the supply dampers for the 28B and 28C turbine building supply fans.

The supply dampers for the 28B and 28C turbine building supply fans are full closed and the controlling SOVs are missing or inoperable. Temperatures in the turbine building are extremely high and subsequently, the bearing temperatures of the secondary pumps and motors are approaching unacceptable levels.

SAFETY ANALYSIS SUMMARY

The major issue considered was the effect of high temperatures on secondary plant equipment. This jumper should be allowed because it will help alleviate those high temperatures by supplying more outside air to the turbine building for cooling.

No unreviewed safety question exists because the turbine building supply fans are not safety related and operability of these fans is not required for any designed accidents. The jumper will help ensure that the Feedwater Pump motor and pump bearings are within acceptable operating temperatures. This will help reduce the possibility of a loss of normal feedwater accident.

SAFETY EVALUATION NUMBER 91-SE-JMP-032

DESCRIPTION

The lifting of two leads at the penetration area to defeat alarm 1A-A3 "CONT RECIRC FAN 1A, B, C AOD CLOSED"

The lifting of the leads will defeat a control room nuisance alarm due to the switch being inaccessible at this time.

SAFETY ANALYSIS SUMMARY

The major issues considered were whether the defeating of the alarm would cause the control room personnel to lose all indication of the air operated damper (AOD) position. With the indicating lamp on the ventilation panel illuminated, it can be shown that the damper is in fact open.

The reason that the change should be allowed is that with the alternate indication available it is obvious that the AODs are open and that the fans will not be harmed by the defeating of the alarm.

An unreviewed safety question does not exist because:

- The containment recirculation fans will not be degraded due to the fact that the AODs are open and the alarm is the result of a spurious signal from a malfunctioning switch.
- The defeating of the alarm from one of the 3 containment recirculation fans will in no way alter the ability of the fan or the AOD to perform its intended function.
- The purpose of the jumper is only to eliminate a nuisance alarm and in no way performs any control function of either the fan or the AOD.

SAFETY EVALUATION NUMBER 91-SE-JMP-034

DESCRIPTION

Jumper out the start permissives for feedwater pump 2-FW-P-1C1 for an uncoupled run to allow post maintenance testing.

Maintenance has been performed on the C main feed pump and it is necessary to test run the pump uncoupled.

SAFETY ANALYSIS SUMMARY

The major issues considered are:

1. Maintenance has been performed on the C main feed pump and it is necessary to test run the pump uncoupled.
2. The jumper involves jumpering two sets of electrical contacts and lifting one lead.

The reasons the change should be allowed are:

1. The pump motor that is to be run uncoupled is isolated from the remainder of the system and the operable equipment is not affected. The autostart of the standby pump will be disabled during the period of the run, but this is allowable.
2. The safety related control and protection features of feedwater are unaffected since the uncoupled pump is isolated from the system.
3. No system containing radioactive material is involved or affected.
4. The pump is currently unavailable (tagged out) and this test run does not affect the emergency plan in any way.
5. Any mechanical failure of the motor during the run would be less severe than a failure during normal operation since the pump is isolated from the system and the motor is uncoupled from the pump.
6. The jumper does not involve any instrument channels and only applies to a non-safety related power supply.
7. The jumper does not affect any electrical loads.
8. No code requirements are affected.

An unreviewed safety question does not exist because:

- The feedwater pump motor is to be test run while the pump is isolated from the remainder of the system. The motor is to be tested uncoupled and there will be no unusual loads or conditions on the pump.
- The reactor protective circuitry is not affected in any way.
- No part of the jumper is related to any safety related system.

- No Tech Specs are involved since the main feedwater system is the only system affected.

SAFETY EVALUATION NUMBER 91-SE-JMP-035

DESCRIPTION

The jumper will defeat the 5B hydro unit hydraulic valve from auto-closing in the event of hydro unit 5A tripping off line with skimmer gate #2 inoperable due to maintenance. To ensure that minimum flow requirements (40 cfs) over the dam to river are met. A minimum flow will ensure no environmental impact to river.

SAFETY ANALYSIS SUMMARY

The jumper will ensure that the minimum required discharge flow from the main dam is maintained. The seismic integrity of the main dam will not be affected. Control of lake level will not be affected.

Analysis for all design basis accidents will not be affected.

SAFETY EVALUATION NUMBER 91-SE-JMP-036

DESCRIPTION

Open Breaker 8s in the G-12 cabinet to defeat the low cooling flow pump trip circuitry.

SAFETY ANALYSIS SUMMARY

The operation of the G-12 breaker is unchanged. An actual cooling flow problem will still be annunciated. Only the pump trip circuitry is defeated to prevent spurious low flow actuations from tripping the running pump. G-12 will still perform its intended safety functions.

SAFETY EVALUATION NUMBER 91-SE-JMP-037

DESCRIPTION

Jumper contacts were placed in the C loop hot leg isolation valve circuit to allow the hot leg isolation valve to be opened. The C loop hot leg isolation valve logic is not operable. This jumper will allow the valve to be opened.

SAFETY ANALYSIS SUMMARY

This jumper allows opening the C loop hot leg isolation valve due to malfunction of the interlock logic. Since the intent of the interlocks will be met, and this is a permissive to open rather than an automatic safety function, administrative control is sufficient as a substitute for the interlock.

SAFETY EVALUATION NUMBER 91-SE-JMP-038

DESCRIPTION

Jumper out the low temperature cutout for the Casing Cooling Tank chillers due to a faulty relay. Faulty relays are not picking up which enable the Casing Cooling Tank chillers to run. The relays, which appear to be inoperable, must energize to allow the chillers to run. The deenergized state of the relay is indicative of a Casing Cooling Tank low temperature, in which case the chillers should not run. However, the temperature circuit was verified to be operable by I&C personnel, and the low temperature switch is in the position which should energize the relays. Power to the relays was also verified.

SAFETY ANALYSIS SUMMARY

The jumper affects a control circuit only. The Casing Cooling Tank will be periodically monitored by the Operations Department to verify that the temperature of the tank contents remains within the allowable values. The Operations Department has the ability to turn on/off the chillers locally to ensure that the temperature requirements are met. There is no credit taken in any Accident Analysis for auto control of the Casing Cooling Tank Chillers.

SAFETY EVALUATION NUMBER 91-SE-JMP-039

DESCRIPTION

Jumper instrument air to trip valves 1-HRS-TV-1613 and 2-HRS-TV-1612 to maintain valves open.

Control power for the valves is not operable due to a transformer failure in the High Range Sample System (HRSS). The valves fail closed to divert water from the containment sump pump discharge to the HRSS. The valves need to be jumpered open to allow the normal periodic pumpdown of the Unit 1 and Unit 2 Reactor Containment Sumps.

SAFETY ANALYSIS SUMMARY

Control power for the valves is not operable due to a transformer failure in the HRSS. The valves fail closed to divert water from the containment sump pump discharge to the HRSS. The valves need to be jumpered open to allow the normal periodic pumpdown of the Unit 1 and Unit 2 Reactor Containment Sumps. The HRSS is only used after a Design Basis Accident. Since it is already out of commission due to the Transformer failure, its operation is unaffected. The containment sump pumps normally pump the containment sumps down periodically, this jumper is necessary to allow this to happen. Containment Isolation trip valves will still function as designed. A small diameter piece of tubing will be used to bypass the SOV and supply normal operating pressure air from the valve's own regulator. The containment sumps pumps are not required to operate in an accident. The HRSS is out of commission due to the transformer failure and is not prevented from operating due to this SOV bypass jumper, since the SOVs use the same transformer. No environmental impact is expected. Containment isolation is still functional.

SAFETY EVALUATION NUMBER 91-SE-JMP-040

DESCRIPTION

Lift Lead # 1FPMN01P00 (Black) on terminal 7-3 in penetration cabinet # RCPC-4B.

The purpose is to clear the Radiant Heat Detector Alarm for reactor coolant pump 1-RC-P-1B caused by a closed circuit on detector in order to support the "Black Board" concept.

SAFETY ANALYSIS SUMMARY

The radiant heat detector for the Unit One "B" Reactor Coolant Pump has failed to the alarm condition. Jumpering out its alarm will not alter its nonexis[illegible] capability to provide further useful information. The jumper will be installed outside reactor containment in a penetration cabinet by lifting one lead. The reactor coolant system temperature, boron concentration, steam demand, or control rod position cannot be affected by this one wire, nor can the levels of radiation or airborne activity. The UFSAR allows operation for extended periods with the substitution of an RCP bearing or motor temperature for an inoperable RCP heat detector provided the bearing or motor temperature is monitored at least once per hour when the RCP is in operation. (UFSAR Section 16.2,* at bottom of page 16.2.-8). By eliminating the locked in trouble alarm on the Main Control Board Annunciator Panel for the Fire Protection System, the operators' monitoring ability will be enhanced, according to the "Black Board" concept. Other detectors will still provide alarms, and will be annunciated normally.

SAFETY EVALUATION NUMBER 91-SE-JMP-041

DESCRIPTION

Jumper N1-1474 is used to block closed boron recovery system valves 1-BR-TCV-111A and 1-BR-PCV-109A to support tagout N1-91-CC-0109.

1-BR-TCV-111A and 1-BR-PCV-109A fail open and must be closed to support tagout N1-91-CC-0109.

SAFETY ANALYSIS SUMMARY

The boron recovery system temperature valves 1-BR-TCV-111A and 1-BR-PCV-109A fail open and must be closed to support tagout N1-91-CC-0109. This is not an unreviewed Safety Question since the "A" Evaporator will be tagged out during this evolution. Blocks to ensure that these valves remain closed are permitted, as specified in operating procedure OPAP-10.

SAFETY EVALUATION NUMBER 91-SE-JMP-042

DESCRIPTION

Jumper instrument air to bypass solenoid operated valves 1-CC-SOV-104A-1 and 104A-2 for trip valve 1-CC-TV-104A, the component cooling water supply to the reactor coolant pump 1-RC-P-1A.

This change is for the purpose of conducting maintenance on air leaks.

SAFETY ANALYSIS SUMMARY

During the time that the jumper is installed, the valve will be inoperable and Tech Spec LCO 3.6.3.1 will be complied with. During that time, a dedicated operator will be available to remove the jumper upon receipt of a Phase B signal. It should be possible to accomplish this action within the 60 second period that is the basis of the Tech Spec. Since the maintenance involved is only to repair leaks in the instrument air piping, operability of the trip valve after jumper removal should not be affected.

Tech Spec LCO 3.6.3.1 ACTION requires the valve to be fixed within 4 hours or else isolate the penetration.

SAFETY EVALUATION NUMBER 91-SE-JMP-044

DESCRIPTION

Two 3 ft. lengths of 1/2" stainless steel tubing with a capped stainless steel "T" between them was installed between the local Pressure Indicators downstream of 2-PCV-MS-223 and 2-PCV-MS-224, the Gland Steam Pressure Regulators for the Unit 2 #1 LP Turbine rear gland and #1 LP Turbine front gland.

The strainer upstream of 2-PCV-MS-224 is clogging, restricting steam flow to the #2 LP Turbine front gland. The strainer has been blown down several times to clear it, but the gland pressure and temperature continue to drop. The strainer is unisolable from the Gland Steam Header. This jumper is an attempt to provide additional steam flow to the #2 LP Turbine front gland. The "T" fitting will be used for a future jumper (if necessary) to provide flow from upstream of the PCV.

SAFETY ANALYSIS SUMMARY

The operators will be able to monitor pressure as before. The control system is being assisted by providing additional steam flow to makeup for the reduced flow caused by the clogged strainer. Installation will be in accordance with the Accident Prevention Manual policies. Equipment reliability will be maintained by supplying gland sealing steam to the #2 LP Turbine front gland. If the gland flow continues to drop, a loss of vacuum could result. The Gland Steam System is non-safety related. Failure of the jumper to provide additional steam flow will merely result in the degraded flow conditions that already exist. The supplying PCV would open further to supply the #1 LP Turbine rear gland in the event of a leak. The activity is designed to prevent exposure of the #2 LP Turbine to air inleakage through its forward gland. No protective circuitry, redundant instrument trains, or 1E power is involved with this mechanical jumper. The Gland Steam system is non Tech Spec related.

SAFETY EVALUATION NUMBER 91-SE-JMP-045

DESCRIPTION

Installation of a stainless steel tubing jumper between the local Pressure Indicators downstream of 2-PCV-MS-223 and 2-PCV-MS-224, the Gland Steam Pressure Regulators for the Unit #2 1 LP Turbine rear gland and #2 LP Turbine forward gland. In addition, stainless steel tubing will also be installed from the strainer blowdown line upstream of 2-PCV-MS-224 to a "T" in the line between the local Pressure Indicators.

A previously injected pipe flange downstream of 2-PCV-MS-224 is apparently clogged, restricting steam flow to the #2 LP Turbine forward gland. This jumper is an attempt to provide additional sealing steam flow to the #2 LP Turbine forward gland.

SAFETY ANALYSIS SUMMARY

The jumper will not hinder operators ability to monitor or control sealing steam to the #1 LP Turbine rear gland or the #2 LP Turbine forward gland. Operators will be able to monitor pressure as before. Installation will be in accordance with the Accident Prevention Manual policies. Equipment reliability will be maintained by supplying gland sealing steam to the #2 LP Turbine front gland. If the gland flow continues to drop, a loss of main condenser vacuum could result. The Gland Steam System is non-safety related. Failure of the jumper to provide additional steam flow will merely result in the degraded flow conditions that already exist. No protective circuitry, redundant instrument trains, or 1E power is involved with this mechanical jumper. The Gland Steam system is non Tech Spec related.

SAFETY EVALUATION NUMBER 91-SE-JMP-046

DESCRIPTION

Jumper relay contacts to simulate a 90% degraded voltage condition on the C phase of the 2J Emergency Bus.

The 27XC-2J1 relay contacts do not operate properly when the relay is de-energized, so the channel must be placed in trip in accordance with Tech Spec 3.3.2.1, Item 7.b.

SAFETY ANALYSIS SUMMARY

Relay 27XC-2J1 has failed. This relay is normally energized, and deenergizes on a C-phase 90% degraded voltage signal. in order to comply with Technical Specification 3.3.2.1, Item 7.b, the channel must be placed in trip. The jumper accomplishes this by placing jumpers across relay contacts to simulate a 90% degraded voltage condition on the C phase of the 2J Emergency Bus. This is conservative since it now will require only a degraded voltage on either A or B phases to actuate the degraded voltage circuitry. The 2 out of 3 logic for the 90% degraded voltage condition is still maintained.

SAFETY EVALUATION NUMBER 91-SE-JMP-047

DESCRIPTION

A jumper will be installed on Terminal Board 604 contacts 7 and 8 of 2-EI-CB-47E to provide a simulated Safety Injection (SI) signal to Relay 2-RPS-RLY-604XA.

Following the Unit 2 trip from full power and subsequent SI on 9/20/91, the other contacts on TB 604 were verified as closed; however, those on 2-RPS-RLY-604XA were not verified as having closed. This jumper will be installed to observe one of the automatic actuations which follow the closure of TB 604 contacts 7 and 8. This is done to verify proper operation of the Train A H_2 Analyzer Heat Trace System.

SAFETY ANALYSIS SUMMARY

Following the Unit 2 trip and subsequent SI on the morning of 9/20/91, the automatic actuations of Terminal Block 604 in 2-EI-CB-47E were verified with the exception of those which should have occurred via 2-RPS-RLY-604XA. These latter actuations may have occurred; however, whether they did so is unknown.

Four actuations should have occurred. The first two are blocks of any closure signals to 2-SI-MOV-2865B (the "B" accumulator discharge valve) and 2-SI-MOV-2867A (the "A" BIT inlet valve). The third actuation is the energizing of the hydrogen analyzer heat tracing following a five minute delay. The fourth and final actuation is an SI/CDA load shed.

Tech Spec 3.6.4.1 (hydrogen analyzer operability) cannot be verified unless its heat tracing can be shown to energize on the SI/CDA. The jumper will allow this demonstration.

Jumper N2-983 will lift lead 2ENSH08P01 on TB 902 contact 4 and short TB 604 contacts 7 and 8 in cabinet 2-EI-CB-47E. The lifted lead will disable the SI/CDA load shed on relay 604XA; the landed lead will block closure of the SI valves (as noted above) and energize the hydrogen analyzer heat tracing, the latter after a minute time delay.

The consequences of the jumper are acceptable because the affected components are moved into a safe condition, with the exception of the SI/CDA load shed disabling; the disabling will be brief and, even in the cause of an SI actuation while the jumper is installed, will not result in an unacceptable low-voltage profile on the emergency buses, in conjunction with the limiting conditions of Item #7.

SAFETY EVALUATION NUMBER 91-SE-JMP-048

DESCRIPTION

Defeat speed sensing trip to reactor coolant pumps 2-RC-P-1B and 2-RC-P-1C during pump start by removing "connect/disconnect" plug from relay 25B3 C2 (2-RC-P-1B) and relay 25C3 C2 (2-RC-P-1C).

Speed sensing relay could actuate prematurely due to setpoint drift and result in an unnecessary tripping of RCP during pump start.

SAFETY ANALYSIS SUMMARY

With speed sensing relay defeated, the RCP will not automatically trip in the event of a locked rotor condition. Operating Procedures OP-5.2 requires the operator to manually trip RCP being started if loop flow does not increase or starting current does not decay within 30 seconds of closing breaker. This manual action is adequate to prevent RCP motor and/or penetration damage should a locked rotor condition exist. No other protection is being defeated by this jumper. All required RCS and/or RHR loops will be maintained operable. Design RCS flowrates are unaffected.

SAFETY EVALUATION NUMBER 91-SE-JMP-049

DESCRIPTION

Jumper N2-986 documents the removal of the driver card for the automatic functions of the moisture separator reheater (MSR) reheat system flow control valves (FCVs).

The driver card to the MSR reheat steam FCVs is providing a signal to partially open the valves, even when the "reset" pushbutton is depressed. By removing the card, only the manual function and the "reset" functions are available, which is the preferred mode of operation.

SAFETY ANALYSIS SUMMARY

The automatic control portion of the MSR reheat system FCVs is being removed by removing the driver card for the automatic portion of the control circuit. However, the manual control and the "reset" pushbutton portion of the circuit will remain active. The automatic functions are not used (and are specifically not allowed to be used), and since the only functions that will be available after the jumper is installed are those functions that are used for normal and upset plant conditions.

SAFETY EVALUATION NUMBER 91-SE-JMP-050

DESCRIPTION

Tie wrap is to be utilized on the turning gear for the main turbine to hold the handle in the engaged position.

The turning gear is not staying engaged, and this mechanism is required to ensure that the turbine continues to roll in order to prevent warping the rotor.

SAFETY ANALYSIS SUMMARY

It is desirable to keep the turbine on the turning gear to ensure that the rotor does not experience bowing. This will further ensure that the turbine rolls true when the unit is returned to power, and minimizes the chance of creating a turbine missile. The turning gear is not safety related, is not a Technical Specification consideration, and is not used to respond to any accident or malfunction of safety related equipment. The turning gear motor rotates the turbine at such slow speed as to be inconsequential.

SAFETY EVALUATION NUMBER 91-SE-JMP-051

DESCRIPTION

Defeat the lower lube oil reservoir, high level alarm for reactor coolant pump 2-RC-P-1A.

The reservoir level is currently at the high level value and will not lock in either the high or normal value. A containment entry at power would be required to adjust the actual level.

SAFETY ANALYSIS SUMMARY

It was determined in Safety Evaluation 89-SE-JMP-034 that there was no unreviewed safety concern if the high and normal level alarms were reversed. This action was taken to "darken" a lit annunciator. In that case, if the level remained high, the alarm would be clear and if and when the level decreased to the normal range, the alarm would be received. This method of "black board" maintained the level of information available to the operators, but in a slightly different method. In current plant conditions, the lower reservoir level on the A pump is just at its normal/high setpoint. The constant alarming detracts from the operators awareness of the actual conditions of the pump. Since the annunciator is a common high/low level alarm, it is necessary to periodically determine if the alarm is a high or a low value.

A valid low level alarm in the reservoir is a condition that requires prompt action in order to protect the pump. It has been determined by Westinghouse that a high level in the reservoir is acceptable for long term plant operation. Operation with an alarm that constantly changes state is not acceptable from the standpoint of equipment safety. If the lube oil reservoir were to suddenly fail, there would be an increased likelihood that the alarm would be dismissed as unreliable until major damage occurred to the pump.

SAFETY EVALUATION NUMBER 91-SE-JMP-052

DESCRIPTION

Jumper out the Hathaway alarm for bearing cooling fan 1-BC-F-1B loss of power.

1-BC-F-1B is tagged out for several weeks to allow maintenance. The bearing cooling fan loss of power alarm is common to both the A and B fans. Jumpering out the field input from the B fan will not only clear the unnecessary alarm, but it will ensure that if the A fan were to loose power, the alarm will be noticed.

SAFETY ANALYSIS SUMMARY

The alarm for bearing cooling fan loss of power is a common annunciator to both the A and B fans. The alarm is mentioned in the UFSAR as being available to the operators to aid in overall system monitoring. With the field input from the B fan jumpered out, the alarm will be fully operable for the A fan. This will result in a "black board" and will ensure that any loss of power on the A fan will be noticed.

The alarm is still valid for the operable fan.

SAFETY EVALUATION NUMBER 91-SE-JMP-053

DESCRIPTION

Jumper out the 'B' moisture separator reheater (MSR) HI LEVEL annunciator.

The HI level is locked in on the 'B' MSR and is no longer an unusual condition. This jumper will remove this "nuisance" annunciator from displaying.

SAFETY ANALYSIS SUMMARY

It is desired to jumper out the 'B' MSR Hi Level alarm in order to comply with the blackboard policy. A high level exists in the 'B' MSR continuously, so the alarm is locked in and no longer indicating an unusual condition. The Hi Level alarm is backed up by a Hi-Hi Level alarm. No automatic functions are defeated by the proposed jumper. The consequences of a turbine trip would not be affected by this jumper because additional indication is available to the operator to warn of an impending turbine water induction event (e.g., MSR HI-HI Level alarm, Turbine Vibration alarm). The MSR Hi Level alarm has no protective functions and is backed up by the Hi-Hi Level alarm. Operator response to the Hi-Hi Level alarm is sufficient to prevent equipment damage or other accidents or malfunctions. Operation of the MSR Hi Level alarm is not addressed in the Tech Specs. No protective or control functions are provided by this alarm. The ability of the turbine overspeed protection system is not affected.

SAFETY EVALUATION NUMBER 91-SE-JMP-054

DESCRIPTION

Install jumper between points terminal blocks TB 105-5 and TB105-6 in auxiliary relay cabinet No. 1, 2-EI-CB-48A.

To allow opening of level control valve 2-CH-LCV-2460A to restore letdown even though 2-CH-HCV-2200A shows intermediate position.

SAFETY ANALYSIS SUMMARY

Low Pressurizer level letdown isolation is not discussed in the CVCS or Small Break LOCA discussions in the UFSAR.

Operation of the Chemical and Volume Control System will be allowed by this jumper. RCS Chemistry and Boron Concentration can be controlled more readily with letdown in service.

2-CH-LCV-2460A will be verified as being able to open after installation. Double verification of jumper removal and verification that 2-CH-LCV-2460A remains open is adequate after removal.

Opening 2-CH-LCV-2460A will not affect RCS temperature or Boron concentration. The systems will perform as required once the letdown isolation valve is opened.

Personnel injury is not likely because the jumper will be installed in the rack on a 25V DC circuit rather than at the valve. Normal caution will be used during jumper installation and removal. Equipment damage is not likely since RCS pressure and temperature are reduced. The letdown orifice isolation valve is a fail-closed valve and is most likely in a full closed position with only limit switches malfunctioning.

CVCS piping will contain the RCS liquid as designed. RCS activity levels will be reduced by allowing letdown purification.

Fuses serve as adequate protection for the 25V DC circuitry. Normal caution will be used during jumper installation and removal

Letdown isolation circuitry has no effect on EDG logics.

No additional surveillance is necessary. Containment isolation is still available via the 2-CH-TV-2204 isolation valves.

SAFETY EVALUATION NUMBER 91-SE-JMP-055

DESCRIPTION

Installation of pipe plug in the bearing cooling (BC) tower cell "1B" sensing line downstream of the isolation valve and pressure switch. This will remove the fire protection (FP) deluge actuation signal for the "1B" cell only.

To prevent the bearing cooling fans from tripping due to degraded "1B" cell sensing line piping.

SAFETY ANALYSIS SUMMARY

The Bearing Cooling tower and FP deluge systems are not required by the Technical Specifications nor are they Safety Related. A fire watch will be posted while this jumper is installed to provide fire detection capability. The fire protection system for the remaining BC tower cells are unaffected and will remain in service.

SAFETY EVALUATION NUMBER 91-SE-JMP-056

DESCRIPTION

Pull relay RSR in Panel 1-EP-CB-96B; this action will deactivate the Bearing Cooling tower trouble alarm (fire protection).

Eliminate alarm until fan repairs are completed.

SAFETY ANALYSIS SUMMARY

Fire Protection deluge to the Bearing Cooling tower 1B cell has already been deactivated for repairs to the 1B fan. Deluge to the other three fans remains active and will not be affected by the jumper. The jumper will only deactivate the alarm which is sent from the 1B cell and is presently locked in. By jumpering out that signal, the alarm may come in for valid signals originating from the other three cells or any other source. In general, however, Bearing Cooling and Fire Protection deluge to the BC tower are not safety-related and are not required to mitigate the consequences of a design bases accident. The ability to deluge the BC tower is not affected by the proposed jumper.

This safety evaluation is very similar to a previously approved SE; see Part A, Box 21 ("Other References").

SAFETY EVALUATION NUMBER 91-SE-JMP-057

DESCRIPTION

Install a loop seal off the drain valve located on the High Level Liquid Waste Tanks (HLLWTs') vent line to the Process Vent System.

To remove condensation from HLLWTs' vent line to the Process Vent System and monitor the rate of condensation accumulation.

SAFETY ANALYSIS SUMMARY

This safety evaluation evaluates the temporary installation of a loop seal off of a drain valve in the vent line from the HLLWTs to the Process Vent System. Due to piping configuration, an inadvertent "loop seal" exists in this vent line which prevents the Process Vent System from sweeping gases from the HLLWTs as designed. This temporary loop seal off the drain valve will remove condensation from the vent line and allow the system to operate as designed.

In the event of a seismic event, the tubing may become disconnected from the drain valve or the "drain can." In this event, an accumulation of condensation on the AB floor could occur. However, the Process Vent System operator would not be adversely affected by the failure of the tubing.

SAFETY EVALUATION NUMBER 91-SE-JMP-058

DESCRIPTION

Install a point to point jumper (red rubber hose) from boron evaporator 1-BR-406 to process vent line drain 1-DA-59 to allow venting the overhead condenser to the process vent equipment line drain.

It is desired to vent the boron evaporator to process vents. However, there is not enough pressure in the overhead condenser to lift the spring loaded discharge check valve. This jumper will bypass the check valve and allow for venting the boron evaporator.

SAFETY ANALYSIS SUMMARY

It is desired to vent the Boron Evaporator to Process Vents, however, there is not enough pressure in the overhead vent condenser tank to lift the spring loaded discharge check valve 1-BR-173. This jumper will temporarily install a rubber hose from 1-BR-406 to 1-DA-59 and bypass the check valve. This will allow venting the Boron Evaporator. Operation of the Boron Recovery System will otherwise remain unchanged

The Boron Recovery System is nonsafety related and is not required for mitigation of any design basis accident. Also, operation of the Boron Recovery System is not addressed in the Technical Specifications.

SAFETY EVALUATION NUMBER 91-SE-JMP-059

DESCRIPTION

Install a pressure regulator, rotometer, and tubing on the Boron Evaporator Reboiler Vent to supply service air as a regulated, measured pressure source to purge the Boron Evaporator to process vents. The previous point to point jumper (red rubber hose) from 1-BR-406 to 1-DA-59 to allow venting the overhead condenser to the process vent equipment line drain will be left installed, so that lower pressures may be used. (will not have to unseat the spring-loaded check valve in the overhead condenser discharge).

It is desired to vent the boron evaporator to process vents. However, there is not enough pressure in the overhead condenser to lift the spring loaded discharge check valve. This jumper will supply service air as a regulated, measured pressure source to purge the Boron Evaporator to process vents.

The previous jumper will bypass the check valve and allow for purging the boron evaporator at lower pressures.

SAFETY ANALYSIS SUMMARY

The Boron Recovery System is nonsafety related and is not required for mitigation of any design basis accident. Also, operation of the Boron Recovery System is not addressed in the Technical Specifications.

SAFETY EVALUATION NUMBER 91-SE-JMP-060

DESCRIPTION

Operating procedure 1-OP-5.4 will be modified to allow the Rx Head to be vented directly to the gas stripper when at least one pressurizer safety valve has been removed.

This feature will protect against radioactive releases to containment or to the environment. The Waste Gas Decay Tank (WGDT) oxygen level must be closely monitored during the venting and for several days thereafter because additional oxygen may enter the primary system via the missing safety valve(s). Further, the rate of venting should not be allowed to exceed the capacity of the gas stripper.

SAFETY ANALYSIS SUMMARY

Previously, the RCS head was vented to the pressurizer, which in turn could be vented either to the gas stripper or to the process vents. With a pressurizer safety valve removed, however, the pressurizer is exposed to containment atmosphere and any gases therein will be released to containment. This jumper will send RCS gases directly to the stripper, so that they will not be released to containment nor to the environment via the process vents.

This jumper should be installed because it retains any radioactive gases from the vessel head within systems which are designed to handle them, rather than release them either to containment atmosphere or to the environment via the process vents.

The RCS head gases will remain confined to systems which are designed to handle them.

SAFETY EVALUATION NUMBER 91-SE-JMP-061

DESCRIPTION

Install hose from primary drains transfer tank (PDTT) pump discharge to normal charging.

To conserve RCS Inventory by recovering leakage past the Loop Stop Valves and returning it to the RCS.

SAFETY ANALYSIS SUMMARY

The jumper is adequately rated for service conditions. A check valve will prevent backflow into the PDTT or the DG System to the Stripper. The jumper location and arrangement are such that the boration flowpath is not adversely affected.

SAFETY EVALUATION NUMBER 91-SE-MOD-001

DESCRIPTION

This EWR provided instructions for the replacement of cable 1EGPA0C020, removal of temporary wiring modification and like for like replacement of N and P terminal blocks in panel 1-EI-CB-202 associated with the 1H EDG differential relay circuit.

SAFETY ANALYSIS SUMMARY

The cable and terminal block replacements are one for one replacements having the same form, fit and function as the originally installed cable and terminal blocks. The operation and function of the 1H EDG differential relay circuit is not changed and special post installation testing will verify component and system operation.

SAFETY EVALUATION NUMBER 91-SE-MOD-002

DESCRIPTION

Engineering Work Request Number 90-296 installed flanges in the 1/2" leakoff line of the relief valves in the Safety Injection System. It also provided instructions for mounting the 1/2" and 1" leakoff lines to new supports coming off the Safeguards Building wall. In some instances the 1/2" line requires additional support. The existing vertical/lateral restraints on the 1/2" and 1" lines were modified to vertical restraints. Although the lines are non-safety, they require seismic supports to insure the integrity of other safety related equipment in the area.

Installation of the flanges was completed in accordance with the Virginia Power Corporate Weld Manual. No testing was required as the lines are 1" or smaller in diameter and are not safety-related.

SAFETY ANALYSIS SUMMARY

The modification did not increase the probability or consequences of an accident already evaluated in the UFSAR.

No unreviewed safety question was involved with this modification.

The margin of safety as set up in the Technical Specifications was not affected.

SAFETY EVALUATION NUMBER 91-SE-MOD-003

DESCRIPTION

Flanges were installed on each of the two stuffing box leak off lines on the main steam trip valves on Units 1 and 2. This was done to make it easier to take the valves apart. The flanges were installed 6 inches (± 2 inches) downstream of the trip valve. This EWR evaluated the installation of flanges on all valves, but installed the flanges on main steam trip valve 1-MS-TV-101A.

SAFETY ANALYSIS SUMMARY

The flanges were in accordance with the original piping Specification NAS 1009 (Installation of Piping and Mechanical Equipment), and do not interfere with the function or operation of the leak off lines.

A flange in the leak off line does not affect the ability of the main steam trip valves to operate as designed.

Installation of flanges does not increase the probability or consequences of an accident. The leak off lines function exactly as originally designed.

SAFETY EVALUATION NUMBER 91-SE-MOD-004

DESCRIPTION

Replacement of 01-CC-RV-115B&C Farris Type 2740 relief valves with 2740-OL Type relief valves.

Farris type 2740 relief valves have been discontinued and parts are no longer available.

SAFETY ANALYSIS SUMMARY

The relief valve replacement does not constitute an unreviewed safety question or require a modification to the Technical Specifications.

Modification will not increase the probability, consequence or possibility of an accident.

Margin of safety as set forth in the Technical Specifications will not be affected.

Accidents of a different type than was previously analyzed would not be possible.

SAFETY EVALUATION NUMBER 91-SE-MOD-005

DESCRIPTION

The alignment lug welds on Charging Pumps 1-CH-P-1B and 2-CH-P-1C and the fin block welds (1-CH-P-1B only) were determined to be undersized. The welds were repaired to provide the required weld size. The associated end plates on the pump cradle were modified to provide for welding.

SAFETY ANALYSIS SUMMARY

The operability of the charging pumps is not adversely affected. The weld repair returned the welds to the design standard. The modification to the end plates does not adversely affect the structural integrity of the pump cradle.

SAFETY EVALUATION NUMBER 91-SE-MOD-006

DESCRIPTION

This EWR provided a temperature permissive for loops TE-BR103A and TE-BR103B; stripper steam heater A & B outlet temperature lo. annunciator BR-A-G1 alarms when steam is not supplied to the stripper. The permissive will resolve nuisance alarming, in support of blackboard. Modification has been performed in the 7300 process racks, cabinet G. No control or indication functions have been altered, only alarm logic.

SAFETY ANALYSIS SUMMARY

A safety evaluation was performed. Modification/activity does not impact operation of gas stripper system. Ability of operator to perform safety functions was not affected. Wiring modification changes logic to annunciator only. Annunciator does not perform control functions. Plant S.R. systems and Tech Specs were not impacted by this activity.

SAFETY EVALUATION NUMBER 91-SE-MOD-007

DESCRIPTION

The manifold associated with 1-CC-FT-128 was replaced due to excessive leakage. The replacement manifold, Whitey model number SS-M3NBF86MPLTPW20-3022, is more reliable. The tube stubs off the manifold have an isolation valve and a cap installed which allows for the instrument to be calibrated without disassembly.

SAFETY ANALYSIS SUMMARY

The Whitey manifold exceeds the design requirements and was pressure tested to ensure it's integrity. It was seismically supported using existing supports. This does not create or increase the consequences of an accident.

SAFETY EVALUATION NUMBER 91-SE-MOD-008

DESCRIPTION

The gas purge system of the Incore Flux Detector System is to be removed.

To reduce radiation exposure associated with replacing the carbon dioxide (CO_2) bottle after depletion.

SAFETY ANALYSIS SUMMARY

The CO2 gas purge bottle should be removed because the system does not significantly reduce corrosion in the thimbles, but does involve periodic maintenance (replacing CO2 bottle) and associated personnel exposure.

Possible corrosion of thimbles was considered as part of an unreviewed safety question determination. Westinghouse and North Anna Power Station Engineering concur that this corrosion is not a major concern.

SAFETY EVALUATION NUMBER 91-SE-MOD-009

DESCRIPTION

Existing instrument manifolds are to be replaced with a similar Whitey instrument manifold.

The Whitey instrument manifolds have been determined to be more reliable than many existing manifolds.

SAFETY ANALYSIS SUMMARY

Some existing instrument manifolds are to be replaced with similar Whitey manifolds. The Whitey manifolds have been determined to be more reliable than some existing models.

The replacement instrument manifolds exceed design requirements. The manifolds will be seismically mounted and have been seismically qualified in Virginia Power Calculation SEO-1655. The function of the new manifolds has not been changed and the associated instrumentation will not be affected.

Therefore, the probability and consequences of any accident or malfunctions has not been changed. Adequate post maintenance testing will verify leak tightness and ensure the possibility for new accidents or malfunctions does not exist.

SAFETY EVALUATION NUMBER 91-SE-MOD-010

DESCRIPTION

This EWR replaced two sections of 4" pipe in the Air Conditioning Condenser Water System. These piping sections are downstream of strainers 2-HV-S-1A and 2-HV-S-1B, and were replaced with class 163 stainless steel piping.

The carbon steel piping was internally corroded. Buildup of corrosion products on the inside of the piping prevents accurate measurement of cooling water flow rate to the control room chillers. Accurate cooling water flow rates are necessary to verify the heat transfer capability of the chiller condensers as required by NRC Generic Letter 89-13.

The stainless steel pipe has the same thickness as the carbon steel and therefore has identical flow characteristics. Class 163 piping is Type 316L stainless steel. This material has been used successfully for replacement of piping in the Service Water System. The effect of this replacement on the seismic qualification of this piping has been evaluated, and the existing seismic pipe supports are adequate.

SAFETY ANALYSIS SUMMARY

This modification is acceptable for the following reasons: The flow characteristics and system operation remain unchanged. Class 163 piping material is compatible with the pumped fluid and the existing piping. The seismic evaluation indicates that the existing pipe supports are adequate, and installation and inspection requirements meet or exceed those of the original piping installation.

SAFETY EVALUATION NUMBER 91-SE-MOD-011

DESCRIPTION

Quench Spray Relief valve 1-QS-RV-100B was defective and required replacement. An exact replacement was not available from the manufacturer in time for the Unit 1 Outage so a suitable replacement was found from another manufacturer.

SAFETY ANALYSIS SUMMARY

The replacement valve, Consolidated Model 2990, was constructed of the same material and met the design requirements of the original valve and provides the same function as the original valve.

SAFETY EVALUATION NUMBER 91-SE-MOD-012

DESCRIPTION

Retraction of Incore Flux Thimble Guide tubes approximately 2 to 4 inches. Removing from service a limited number of guide tubes.

Eddy current testing of the incore flux thimble guide tubes determines that excessive wall thinning has occurred.

SAFETY ANALYSIS SUMMARY

This Safety Evaluation reviewed the safety implications of retracting flux thimble guide tubes and/or removing from service a limited number of guide tubes. The major issue considered was the ability of the incore flux monitoring system to perform its job with tubes retracted and/or removed from service.

Retracting tubes does not affect system performance at all and it enhances the structural integrity of the components by providing a new wear surface for future operation. Removing tubes from service can affect the ability of the system to work depending on how many and which tubes are affected. The Reactor Engineer and the SNSOC will approve all tubes prior to being removed from service. They will ensure that all Tech Spec requirements are met with regard to maintaining the incore flux monitoring system operable.

No unreviewed safety question exists because the proposed changes will not affect the ability of the incore monitoring system to respond to accidents (fuel assembly loaded in improper position, dropped rod, etc.). No new accidents are created and the potential for a small LOCA is reduced.

SAFETY EVALUATION NUMBER 91-SE-MOD-013

DESCRIPTION

The Emergency Diesel Generator over/under excitation relay will be replaced to eliminate contact deterioration. The existing D-3 relay will be replaced by two BBC type 76T solid state relays. The circuit function has been re-evaluated and found to be non-safety related.

The existing Westinghouse D-3 relay (40/76 has continuing contact deterioration problems, causing numerous station deviations, nuisance alarms and loss of protection. The modification is intended to resolve the problem.

SAFETY ANALYSIS SUMMARY

The Emergency Diesel Generator (EDG) System had been entirely classified as Safety-Related. This is a proper classification for many of the system components and sub-components which are required to support the system safety function. In some cases the safety-related classification has existed simply because a component-level evaluation was never conducted to identify those components which did not perform a safety function.

It has been determined that the EDG under/over excitation relay was, in the past, unnecessarily blanketed by the EDG Safety-Related Classification. A subcomponent classification evaluation (SCE number EG024) has been performed as part of EWR 88-273B. The evaluation determined that the 40/76 relay function was not a Safety Related application because it is overridden by an emergency start and it has sufficient isolation from Class 1E power supplies. Therefore, the relay has been classified non-safety.

The function of these relays (40/76) is to alarm an under-excited condition (40 relay) of the emergency diesel generator (EDG) or shutdown the EDG upon loss-of-excitation (76 relay) when in an non-emergency EDG start operating mode. These relays provide no protection, supervision, or trip function when the EDG is operating in its safety related mode. During an emergency EDG start, any relay contacts which could prevent operation of the EDG or shutdown the EDG are isolated from the EDG control circuits. Additionally, sufficient isolation has been provided between the relays and their Class 1E power supplies to prevent a malfunction in the relays from being transmitted back to the power supply. Any spurious operation of failure of either or both of the 40/76 relays or their auxiliary relays in the energized or de-energized mode with their associated contacts closed or open, respectively, will not prevent or diminish the EDG's capability from performing its intended safety function, or compromise any Class 1E supplies.

SAFETY EVALUATION NUMBER 91-SE-MOD-014

DESCRIPTION

This EWR addendum provided justification and direction for the modification of a Feedwater MOV by changing out the motor pinion and worm shaft gears. The new gears have slowed the stroke time down in order to lessen the problems with inertia/momentum. The net result will be easier control of the valve operator and greater reliability.

SAFETY ANALYSIS SUMMARY

The Safety Evaluation found that there are no unreviewed safety questions because the modification would not violate the bases and LCOs of the Tech Specs. All postulated failures were bounded by analyses found in the UFSAR.

SAFETY EVALUATION NUMBER 91-SE-MOD-015 Rev. 1

DESCRIPTION

Guard plates will be installed over the limit switches on the refueling manipulator crane.

The purpose of this change is to avoid accidental jarring of the limit switches.

SAFETY ANALYSIS SUMMARY

A guard plate will be installed over the limit switches on the refueling manipulator crane to protect the limit switches from jarring. Installation of the plate will help ensure proper operation of the crane. The guard plate is small, 11 gauge x 6 inches x 1-2 inches (maximum) and weighs less than three pounds. The failure of the guard plate is bounded by the evaluations for failure of the crane.

SAFETY EVALUATION NUMBER 91-SE-MOD-016

DESCRIPTION

The change being made is to replace six indicating lights per diesel and replace them with indicating lights which are in series with a 2 kilohm resistor.

To eliminate the chance of rendering a diesel inoperable in the event that a light bulb "shorts out".

SAFETY ANALYSIS SUMMARY

The major issues considered were whether the replacement lights would function within the guidelines of the original design basis.

The reason(s) the change should be allowed is because the replacement lights have the same design requirements as the original and will be mounted in the same location. Additionally, with the resistor in series with the light, it will render the affected circuit more reliable.

An unreviewed safety question does not exist because:

- the replacement lights will function exactly the same as the existing lights,
- these lights are only used for indication and will not function any differently than the existing lights,
- the replacement lights meet or exceed all design requirements. They have been evaluated and found to be an acceptable replacement. The component/system will function as designed.

SAFETY EVALUATION NUMBER 91-SE-MOD-017

DESCRIPTION

Trap 02-MS-TD-421D has a pinhole leak in its body and had to be replaced. The only traps available have 1/2" NPS connections whereas the existing trap had 1" NPS connections. The two traps were equivalent in terms of construction, capacity, function, and pressure/temperature ratings, therefore the trap was used with reducers. Also, the drain valve was replaced with a Conval valve. The Conval valve met or exceeded all requirements of the original valve. Seismic integrity was maintained since weight change was less than 10%.

SAFETY ANALYSIS SUMMARY

The replacement valve provides the same function and meets the same design requirements as the original valve. Since the valve meets the original design criteria, the probability of failure is not increased.

SAFETY EVALUATION NUMBER 91-SE-MOD-018

DESCRIPTION

Flanges are to be installed on the auxiliary feedwater pumps (motor driven) seal cooling water line.

The flanges will allow upper pump casing to be easily removed.

SAFETY ANALYSIS SUMMARY

Flanges are being installed on the seal cooling water line of the auxiliary feedwater pumps (motor driven) to facilitate pump maintenance.

The possibility of pump failure was considered in this review. The installation of flanges would not affect pump operability, so no increase in possibility of pump failure exists.

SAFETY EVALUATION NUMBER 91-SE-MOD-019

DESCRIPTION

In order to restore the cold piston setting to the design range of 2 5/8" to 5 1/8", (2) 3/4" thick spacer plates were added between the pivot lug and the end plate of snubber 1-CH-HSS-803. This will increase the pin-to-pin length by 1 1/2" and result in a cold piston setting of about 4 7/8". The plates were cut to fit the contour of the snubber end plate (i.e. 4 1/4" x 2 1/2") with 7/16" diameter holes to pass the end plate bolts through. The plates were made from ASTM A36, SR material. This is suitable for the application since the plates are only loaded in compression and their additional deadload (i.e. 4.52 LBS) will have no impact upon existing analyses, but is hereby documented. The existing end plate bolts were replaced with 3/8-16UNC x 3" long, carbon steel, ASTM A449, GR 5, SR bolts The bolt material has a yield strength in the range of 55-95 KSI which is more than adequate for the application and do not significantly alter component stiffness. The plates were coated in accordance with Specification for Inside Containment Protective Coatings NAS-3000.

SAFETY ANALYSIS SUMMARY

The modification of the pin-to-pin dimension is a local modification to snubber 1-CH-HSS-803 and does not affect the associated piping or any other component or system. The increased pin-to-pin dimension will permit the snubber to support the pipe at the design cold piston setting. The local modification has been evaluated and found to be acceptable, enabling the snubber to perform its intended support function. This is in accordance with the original design basis. Therefore, the modification does not increase the consequences or probability of a previously analyzed accident nor does it create an accident not previously considered.

SAFETY EVALUATION NUMBER 91-SE-MOD-020

DESCRIPTION

Accumulator test line containment isolation valve, 1-SI-TV-1842 is a 3/4" Masoneilan Model 38-20721. Like many others of this type of valve, the stem has a tendency to rotate as the valve strokes. This causes problems with the mechanism that actuates the valve position indication limit switches.

An anti-rotation device was installed on the valve. The device is held in place by the limit switch mounting screws. it straddles the limit switch actuating arm, thereby preventing rotation of the valves' stem.

SAFETY ANALYSIS SUMMARY

The anti-rotation device is a light weight, simple, passive device. Reliability of the position indication is improved. Seismic qualification of the valve and piping is not affected. The potential for, or consequences of a small break LOCA are not affected.

SAFETY EVALUATION NUMBER 91-SE-MOD-021

DESCRIPTION

This EWR replaced the actuator motor on safety injection MOV with one having the same form, fit and function as the original except for a 5% reduction in motor weight and slightly more current consumption (.2 amps).

SAFETY ANALYSIS SUMMARY

The motor replacement has no effect on Nuclear Safety. The motor, with the exception of the weight change was essentially a replace in kind activity. The motor replacement did not create an unreviewed safety question because it does not increase the probability or consequences of an accident nor did it create the potential for a new accident. The operation of the component remains unchanged from the original design basis.

SAFETY EVALUATION NUMBER 91-SE-MOD-023

DESCRIPTION

The fluorescent lighting in the Emergency Diesel Generator Rooms was replaced with high pressure sodium fixtures and lamp. The reason for this being that the existing lighting was a low ballast cutoff temperature causing lights to turn off at unpredictable times. The new HPS lighting has a higher cutoff temperature and has provided better illumination.

SAFETY ANALYSIS SUMMARY

The major issues considered were the effects of a seismic event and loss of off site power. In the case of a seismic event, the new lighting has not had any effect on the diesel engine operation in any way. The lighting does not interface with the diesel and has been seismically evaluated and found to be acceptable. In the case of a power loss, the emergency lighting will self-activate, just as it would with the existing lighting. Replacement of the lights does not pose an unreviewed safety question. Seismic and electrical considerations have been made and the new lighting has been found to be acceptable.

SAFETY EVALUATION NUMBER 91-SE-MOD-0024

DESCRIPTION

The Engineering Work Request (EWR) places switch covers on the containment depressurization activation (CDA) switches in order to lessen the chances of mistaking them for adjacent switches. The switches in question are located on control room benchboard 1-1 and the adjacent switches are those for SI initiation and Phase A initiation. Additional labeling will be added if required.

The switch covers will act as mechanical barriers to prevent inadvertent operation of the control switches.

SAFETY ANALYSIS SUMMARY

The major issues considered were: the selection of a suitable switch cover to serve as a mechanical barrier to prevent inadvertent operation of the CDA actuation switch(es). Also considered was the need for additional labeling to compensate for legend (escutcheon) plate removal as well as additional labeling because of the switch cover profile. The clear plastic of the cover allows visual verification of switch position.

The reason(s) the change should be allowed: this switch cover should serve as a mechanical barrier to prevent operator error in inadvertent operation of the CDA actuation switch(es). This cover addition will not alter the design or circuitry associated with the CDA system. The cover will not hamper normal or emergency operator actions.

An unreviewed safety question does not exist because:

- No change to the CDA circuitry is involved. The switch cover will serve to prevent operator error while additional labeling will compensate for cover profile. Operators will still be able to operate the switch/system as designed.
- The switch cover will serve to prevent inadvertent operation of the switch only. The circuitry of the CDA system will not be altered. Use of the control room bench boards will be as designed.
- Equipment operability is not modified. Implementation will be in modes 5 or 6 only. No change will result in the component/system design or circuitry. The chances for inadvertent operation will be reduced.

SAFETY EVALUATION NUMBER 91-SE-MOD-025

DESCRIPTION

Extra dowel holes installed in Charging Pump 2-CH-P-1B served no apparent purpose. They had been used in the past by mistake during seal replacement, not allowing the out board bearing housing to be properly reassembled. This action in turn caused the pump to bind when put into use. Maintenance requested a method by which these holes could be plugged to ensure that they are not inadvertently used again in the future.

After considering the design temperature of the pumps and material characteristics of possible plugs, the easiest solution was to plug the holes with Belzona R metal. This solution was easy to implement and met all material requirements of the pump.

SAFETY ANALYSIS SUMMARY

No unreviewed safety question exists. The plugging of the hole does not interfere with the pumps operation. The modification only exists on a external pump surface and does not interfere with moving parts of the pump. Also, no adverse affects result from the plug coming loose. The plug can in no way hurt the pump.

SAFETY EVALUATION NUMBER 91-SE-MOD-026

DESCRIPTION

Thermostat temperature switches 01/02-HV-TS-601A, B, C/701A, B, C on Control and Relay Room Chillers 01/02-HV-E-4A, B, and C for both Unit 1 and 2 are currently calibrated at 300°F. The thermostat temperature switch is physically mounted remote from the actual area to be controlled at 300°F. The thermostat should be set at 239°F +/- 9° F, per East Coast Compressor Corporation.

The Chiller Safety circuit on Unit 2 did not include contacts from the chilled water outlet flow switch (02-HV-FS-2213A, B, and C). Without adequate chilled outlet flow the Control and Relay Room Chiller will fail to operate. The current configuration of the chiller safety circuit wiring had the chilled water flow outlet relay contacts wired to an indicator locally or in the control room. East Coast Compressor Corporation has recommended the chilled water outlet flow switch be wired in series with the existing safety circuit switches on the chiller. This configuration currently exists on Unit 1 and has been added to Unit 2 by this EWR. This switch ensured that there is adequate chilled water flow outlet (50 GPM or greater) and prevent the compressor from overheating and automatically shutting off.

SAFETY ANALYSIS SUMMARY

The wiring modification and high temperature switch replacement of the Control and Relay Room Chillers has been performed under approved maintenance procedures and has not affected the operation of the chillers. System design bases for the operation of the Control and Relay Room Chillers were unchanged. Thus, the safety of the Control and Relay Room Chillers was unchanged.

SAFETY EVALUATION NUMBER 91-SE-MOD-027

DESCRIPTION

As a result of a complete roofing condition and repair analysis conducted in April 1990, the following maintenance was performed. The report recommended and this EWR implemented the complete removal of existing built-up roofs (BUR) on U1 & U2 Main Steam Valve House, Unit 1 and 2 Rod Drive, Unit 2 Casing Cooling, H_2 Recombiners and Auxiliary Building Elevator Shaft; the replacement roofing includes new insulation covered by a single-ply EPDM membrane.

Additionally, the report and this EWR addressed U1 and U2 Quench Spray, Security Blockhouse, Waste Solids, Oil Storage Pumphouse roofs, calling for them to be retrofit; existing loose gravel ballast was removed, identified wet insulation areas were cut out, removed and replaced; existing BUR remained and was topped with a 1/2" layer of insulation before being covered with the single-ply EPDM membrane.

SAFETY ANALYSIS SUMMARY

Reroofing with the single-ply membrane system will not increase the probability of occurrence or the consequences of an accident or malfunction of a different type than previously considered on the original roofing.

Safety analysis required due to the necessity to drill 3/8" diameter holes, 1-3/4" deep, spaced 12" o.c. into safety related concrete walls. These holes received the anchoring fasteners used in association with the termination detail for the roofing membrane to vertical surfaces.

SAFETY EVALUATION NUMBER 91-SE-MOD-028

DESCRIPTION

Abandonment-in-place of liquid waste (L.W.) equipment identified in Type 1 study (NP-2154-13) as being suitable for retirement, because it is no longer required for plant operation.

To abandon-in-place several components of the L.W. system so that preventive maintenance and periodic tests will no longer be required, thereby saving on man-hours and man-rem expenditures.

SAFETY ANALYSIS SUMMARY

The major issues considered were the probability of some other accident or malfunction, resulting from the bandonment of several L.W. components, that was not previously considered. This also includes any possible resultant environmental impacts.

A Type 1 Study (NP-2154-13) was done which concluded that various portions of the L.W. system could be retired in place without affecting the operation of the plant. These components are no longer in use. Only two of the forty components are classified as Safety Related. The two components act as system pressure boundaries for the component cooling water which is supplied to them. These two condensers as well as the other components will be drained and isolated as appropriate. They will no longer function as a system pressure boundary.

These components currently exist in the plant. The abandonment-in-place of these components will not physically alter any equipment except to isolate them from the system. They will no longer be part of the system fluid boundary. The integrity of the system will be maintained and the L.W. effluent treatment will not be compromised.

SAFETY EVALUATION NUMBER 91-SE-MOD-029

DESCRIPTION

A service water valve (isolating the pressure indicator off of the line 4"-W-469-151-Q3) became plugged. The existing globe valve design did not allow associated piping to be unplugged. Also, tubing was welded to the existing valve requiring the tubing to be cut to clean out the plugged line. To correct these problems, a gate valve was installed to replace the existing valve and the piping configuration was modified. The gate design and new configuration allows the line to be easily unplugged.

This type of replacement was found acceptable by EWR 87-162 and has been used in the SW System when 1" or 3/4" globe valves have become plugged.

SAFETY ANALYSIS SUMMARY

No unreviewed safety question exists. The valve did not alter performance of the Service Water System under accident or normal conditions. The new valve serves the same function as the old one.

SAFETY EVALUATION NUMBER 91-SE-MOD-030

DESCRIPTION

The lube oil filter on charging pump 2-CH-P-1B was worn and in need of replacement. The vendor no longer made this filter but provided a different filter as an updated equivalent. This EWR evaluated the new filter for acceptability and seismic considerations. It also provided an installation procedure and information for document updates.

SAFETY ANALYSIS SUMMARY

The filters were similar and had the same performance characteristics. The new filters were the recommended replacement by the vendor and were supplied safety related. There will be no change in the probability of malfunction or accident due to this change.

SAFETY EVALUATION NUMBER 91-SE-MOD-031

DESCRIPTION

This modification replaces the isolation valves and associated piping to the auxiliary feedwater pump lubrication oil pressure indicators.

The new piping configuration is such that the overall height of the pressure indicator is reduced and the number of potential leakage points is reduced to one tubing connection.

SAFETY ANALYSIS SUMMARY

This modification replaces the isolation valves to the Turbine Driven Auxiliary Feedwater Pump Lube Oil Pressure Indicators. The new valve and piping configuration will decrease the number of potential leakage points to only one tubing connection.

The valves and fittings meet the system design requirements of the Auxiliary Feedwater Pump Lube Oil System. The seismic integrity on the Auxiliary Feedwater Pump and the lube oil piping is maintained. System function, operation and performance remains unchanged.

SAFETY EVALUATION NUMBER 91-SE-MOD-032

DESCRIPTION

This EWR provided instructions for installing a new 25 pair telephone cable and an additional duplex receptacle for the Alternate OSC Area in the Unit 1 Emergency Switchgear Room. The modification, although impacting a safety related structure, has been classified non-safety related. A seismic evaluation was provided. The modification was necessary to support additional communications requirements for the Alternate OSC Area.

SAFETY ANALYSIS SUMMARY

The major issues considered were the seismic impact to the I-IV Battery Room in the Unit 1 Emergency Swtichgear Room.

A seismic evaluation was performed to ensure that there would be no adverse impact to the existing qualification of the Battery Room. The conduit from the existing IDF Box to the Alternate OSC Area contains a 25 pair telephone cord. Existing, qualified supports have been used to run the conduit. The conduit from the lighting panel 1S12, to the new electrical outlet in the Alternate OSC Area has not been seismically mounted, however, based upon the weight and length of the conduit to be run, there would be minimal impact to equipment in the vicinity if the conduit were to fall.

An unreviewed safety question does not exist because the activity did not increase the probability or consequences of occurrence for an accident previously evaluated in the UFSAR. Additionally, the activity did not increase the probability or consequences of occurrence for any malfunction of equipment related to safety and previously evaluated in the UFSAR.

SAFETY EVALUATION NUMBER 91-SE-MOD-033

DESCRIPTION

This EWR provided engineering approval for ordering and installing lock-out plates for safety and non-safety related 4160V ITE/ABB breakers. The 4160V breakers were originally designed to accommodate these lock-out plates. The lock-out plates were ordered safety related and a new procedure was created to provide installation instructions. Installation of the devices was performed during convenient maintenance intervals.

SAFETY ANALYSIS SUMMARY

The major issues considered were the seismic impact to the existing breaker configuration, personnel safety and inadvertent racking in of the breaker during maintenance activities.

The reason the modification was allowed is that the original design of the 4160V ITE/ABB breakers provided a mounting bracket for the locking devices. However, the locking devices were not installed when the breakers were originally purchased. A seismic analysis has been performed to ensure that the locking devices have not adversely impacted the qualification of the breakers or impacted the breaker during a seismic event. These locking devices have enhanced personnel safety during breaker maintenance, as well as, ensure that the breaker is not inadvertently racked in.

An unreviewed safety question does not exist because the activity does not increase the probability or consequences of occurrence for an accident previously evaluated in the UFSAR. Additionally, the activity does not increase the probability or consequences of occurrence for any malfunction of equipment related to safety and previously evaluated in the UFSAR.

SAFETY EVALUATION NUMBER 91-SE-MOD-034

DESCRIPTION

Use of General Electric (G.E.) Breaker Locking Devices as tagging devices for 120 volt and select 480 volt breakers throughout the station. Devices will be permanently attached to the breakers.

To provide adequate tagging for 120 volt and 480 volt G.E. breakers and to further ensure that tags are attached to the breakers as per the Virginia Power Accident Prevention Manual.

SAFETY ANALYSIS SUMMARY

The major issues considered and the reason the change should be allowed are as follows:

In considering whether or not to use breaker locking devices for tagging purposes on G.E. 120 volt and 480 volt breakers, engineering reviewed the operational concerns with inadvertently energizing breakers which were tagged-out. By using these devices, both personnel injury and equipment damage will be reduced.

The above referenced device will not adversely impact any Tech Spec required equipment. Normal breaker operation is not altered by the addition of a tagging device.

The modification is not adversely impacting any equipment necessary for accident mitigation or safe shutdown.

The devices will help to ensure personnel and equipment safety by reducing the probability of lost or missing tags.

SAFETY EVALUATION NUMBER 91-SE-MOD-035

DESCRIPTION

The inspection cover on the Charging Pumps speed increaser gearbox was replaced with a similar one from the same vendor. This was done to prevent oil vapors from the increaser gearbox from being drawn into the motor.

SAFETY ANALYSIS SUMMARY

No unreviewed safety questions exists because this modification does not adversely affect the operation of the Charging Pumps. The cover has the same fit, form, and function as the existing one.

SAFETY EVALUATION NUMBER 91-SE-MOD-036

DESCRIPTION

The original EWR 90-411 provided a temporary service air connection (hose) to the mat sump pumps on both units. A more permanent connection was needed and these addenda allowed the connection to be hard piped with 3/4" copper lines. The pipe was attached to exterior containment walls in accordance with STD-CEN-0041 (Attachment to the Reactor Containment Exterior Concrete Wall).

SAFETY ANALYSIS SUMMARY

No unreviewed safety question exists. The referenced Civil Engineering standard STD-CEN-0041 (Attachment to the Reactor Containment Exterior Concrete Wall) provided guidance to ensure that the structural integrity of the containment structure was not affected by attaching pipe supports to containment. Also, because the structural integrity was not affected, the probabilities or consequences of any accident or malfunction was not increased. Finally, no accident or malfunction was created because the modification did not interfere with any safety related system or components.

SAFETY EVALUATION NUMBER 91-SE-MOD-037

DESCRIPTION

The existing heating and ventilation system for the Control Rod Drive Room is modified by the addition of a mechanical refrigeration system consisting of a condensing/compressor unit, cooling coil, filter and louver, and related components which will be installed on the Rod Drive Room roof at elevation 229'-6".

To lower the summertime ambient temperatures in the Rod Drive Room and Cable Vault areas, thereby extending the operating life and reliability of electrical equipment/components located therein.

SAFETY ANALYSIS SUMMARY

The accidents considered in the Unreviewed Safety Question evaluation are 1) Uncontrolled Rod Cluster Withdrawal and 2) Large Break LOCA. The equipment malfunctions related to safety that were considered are 1) Failure of Emergency Fans 1 & 2 HV-F-68A and B and 2) Malfunction of Rod Drive Control System.

This Design Change should be allowed to be implemented to North Anna Power Station, Units 1 and 2 since it improves the ambient temperature environment of the Rod Drive Room and Cable Vault and thereby extending the operating life and reliability of all electrical equipment and components located in these areas.

SAFETY EVALUATION NUMBER 91-SE-MOD-038

DESCRIPTION

This EWR provides documentation on the temporary installation of the Calgon Corrosion Test Station, Chemonitor TL, on the service water system. The station is installed in the area of motor operations for valves 1, 2-MOV-123A, B, 223A, B in the service water valve house, evaluation 326'-0". The station takes up to 10 gpm of the service water upstream of the spray nozzles and returns experimentally treated water to the service water reservoir. This installation does not affect the service water system performance and serves the purpose to investigate adequate service water chemical treatment to control the service water system corrosion rate. The test station has now been removed.

SAFETY ANALYSIS SUMMARY

The implementation of this EWR does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in Section 9.2.1 of the UFSAR, neither does it create possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR. The test station installation also does not reduce margin of safety as defined in the basis of 3/4.7.4 and 3/4.7.7 of the Technical Specifications. No functional changes were made to any safety related system or component.

SAFETY EVALUATION NUMBER 91-SE-MOD-039

DESCRIPTION

This Engineering Work Request (EWR) authorizes use of Swagelok compression fittings in lieu of welded fittings for replacement of containment isolation valve 1-SS-TV-102A. It also provides instructions for replacement of the valve.

The valve will be replaced with the containment in a vacuum. Use of compression fittings will expedite the job.

SAFETY ANALYSIS SUMMARY

The compression fittings should be as reliable as the welded fittings they replace. The fittings that will be used were purchased to Q2 requirements. Additional nondestructive examination (NDE) will be performed to upgrade them to Q1.

Failure of one of these fittings would result in a hole no larger than 0.245 inches. One charging pump is adequate to maintain pressurizer level with a hole this size.

Upon replacement of 1-SS-TV-102A, the ability to isolate penetration 56 will be ensured. With the exception of the compression fittings the system will maintain its original configuration. The systems pressure boundary and seismic qualification will not be compromised.

SAFETY EVALUATION NUMBER 91-SE-MOD-040

DESCRIPTION

Replacing the motor for 01-SW-MOV-113A service water system motor operated valve in a near like-for-like replacement.

The original motor was grounded and could not be repaired.

SAFETY ANALYSIS SUMMARY

The major issues considered were whether the replacement motor would function within the guidelines of the original design basis.

The reason(s) the change should be allowed is because the replacement motor has the same characteristics as the original except for minor differences in horsepower ratings and nominal amperage ratings, which will not affect functionality. These exceptions have been evaluated and have been found to be insignificant.

An unreviewed safety question does not exist because:

- The motor being installed will function as designed.
- These motors will be properly tested to ensure operability and will not function any differently than the existing motors.
- The replacement motor meets or exceeds all design requirements. It has been evaluated and found to be an acceptable replacement. The component/system will function as designed.

SAFETY EVALUATION NUMBER 91-SE-MOD-041

DESCRIPTION

This EWR was written for the purpose of reducing the alarm setpoint on the Heat Trace on the piping sections between the Blender and the VCT (U1 & U2). The controlling setpoint has not been affected. The modification has reduced/eliminated nuisance alarms on the local annunciator panel. The alarm setpoint has been maintained above the temperature required for solubility of the fluid.

SAFETY ANALYSIS SUMMARY

The Safety Analysis for this modification has shown that no unreviewed safety questions exists. The margin of safety has not been reduced. The consequences of failure of equipment or components important to safety and described in the UFSAR has not been increased.

SAFETY EVALUATION NUMBER 91-SE-MOD-042

DESCRIPTION

This Engineering Work Request (EWR) will install a 5-valve manifold on existing Unit 1 and Unit 2 Reactor Coolant System level transmitters 01/02-RC-LT-1000/2000. In addition, this EWR will remove existing tubing, calibration pot, and sealed reference, if as low as reasonably achievable (ALARA) dose conditions exist. The level transmitters provide pressurizer water level indication on the reactor coolant panel in the fuel building. This water level indication is utilized during startup, shutdown, and refueling.

The existing transmitters require a complicated valve line up, root valve manipulation, and filling during calibration. This presents an ALARA concern. Installing 5-valve manifolds will eliminate the complicated valve lineup, root valve manipulation, and filling. In addition, time spent in containment will be less and the dose received will be less.

SAFETY ANALYSIS SUMMARY

The installation of the 5-valve manifold will not alter the operation of level transmitters RC-1000//2000 and will not alter the operation of the RCS.

The 5-valve manifold installation will not increase the chances of accidental depressurization of the RCS as analyzed in section 15.2.12 of the UFSAR. The manifold will be installed and tested in accordance with approved specifications.

The design basis of the Reactor Coolant System (RCS) are preserved. The 5-valve manifold installation will not affect the operation of the RCS.

The major issue considered was how this modification would affect the pressure boundary of the RCS. The manifold is qualified to be used in the environment that it will be exposed to. The tubing will be connected to the manifold using approved fittings and welds. Thus, the RCS pressure boundary will not be affected.

This modification will lessen the time required to calibrate level transmitters 1000 and 2000, which will reduce dose. In addition, it will alleviate a complex valve line-up procedure required prior to calibrating the level transmitters. To reduce the dose rate and simplify the calibration procedure and not affect the operation of level transmitters 1000 and 2000 is the reason why this modification should be allowed.

SAFETY EVALUATION NUMBER 91-SE-MOD-043

DESCRIPTION

This Engineering Work Request (EWR) will change the field wiring to the contact output board of the inadequate core cooling monitor (ICCM) cabinet trains "A" and "B" from a normally open set of contacts to a normally closed set of contacts. In addition, the software of the multibus controller will be changed to energize this contact coil upon loss of power (AC or DC) in the ICCM cabinet.

The existing contact output board energizes a coil and opens a set of dry C contacts when power (AC or DC) is lost in the ICCM cabinet. The hathaway system is looking for an open set of contacts to provide an alarm annunciation in the control room. The current configuration of the power loss contacts will not provide an annunciation in the control room.

SAFETY ANALYSIS SUMMARY

The wiring modification will not affect the operation of the ICCM cabinet trains "A" and "B". This modification will only affect the annunciation of power loss (AC or DC) in the ICCM cabinet. This annunciation will still alarm when loss of plasma display is detected, which is shortly after loss of AC or DC power to the ICCM cabinet.

The wiring modification will only affect an annunciation alarm in the control room for a system that provides monitoring capability only. The annunciation will aid the operator in evaluating the condition of the reactor core cooling system.

The design basis of the system are preserved. The wiring modification will provide annunciation in the control room when loss of power (AC or DC) occurs in the ICCM cabinet.

The major issue considered was how this modification would affect the monitoring capability of the ICCM cabinet. Currently, an alarm annunciation does not occur when loss of power (AC or DC) to the ICCM cabinet is present. This alarm will eventually annunciate when loss of power to the plasma display occurs. This wiring modification will not affect the monitoring capability of the ICCM.

This modification will alert the operator when there is a problem in the ICCM cabinet. The alarms and indications that the ICCM provide are vital to the stability of the reactor core. By alerting the operator of possible problems in the ICCM cabinet and not affecting the operation of the ICCM is the reason why this modification should be allowed.

SAFETY EVALUATION NUMBER 91-SE-MOD-044

DESCRIPTION

Bump bars were installed across the front panel controls of all inverters in the switchgear (SWGR) rooms. A seismic evolution of the bump bars was performed and the installation was documented to ensure that no adverse consequences would result.

Existing bump bars on the SWGR room inverters were installed without any quality assurance documentation. While the installation does not appear to cause a problem, a seismic evaluation would require that the material and installation be documented. Hence, this Engineering Work Request (EWR) provides a vehicle for such documentation.

SAFETY ANALYSIS SUMMARY

The major issue involved with the addition of bump bars onto the face of inverter panels is whether or not the additional mass would alter the original seismic evaluation of the inverters and/or create a new safety significant issue by itself. Taking the issue of mass, first: the weight of the bump bars is less than 1% of the inverter weight which was shown to have negligible effect upon frequency, deflection and anchorage reactions. (See engineering evaluation in EWR #91-129). The concern for the bump bar creating its own safety significant issue can be rationalized by the engineering evaluation which demonstrates its ability to withstand the Operating Basis Earthquake (OBE)/Design Basis Earthquake (DBE) events without failing. Since this modification is a self-contained, structural only change, and does not alter original operation or design basis of the inverters an unreviewed safety question does not exist for this modification.

SAFETY EVALUATION NUMBER 91-SE-MOD-045

DESCRIPTION

Teflon sealing materials will be replaced in the air equalizing valves in the personnel air lock and equipment hatch emergency air lock. Also Teflon shaft seals will be replaced in the personnel air lock escape hatches.

Teflon has been documented to begin to degrade when exposed to a radiation environment in excess of 1.5×10^4 rads. Replacement of Teflon components will maintain original design factors of safety against containment leakage.

SAFETY ANALYSIS SUMMARY

The air equalizing valve seats and stem seals in the Containment Personnel Air Lock and Equipment Hatch Emergency Air Lock shall be rebuilt to replace Teflon sealing components with modified EPT which has a radiation resistance of 1 x 10E8 Rads. The teflon shaft seals in the Containment Personnel Air Lock Escape Hatches shall be replaced with tefzel shaft seals which have a radiation resistance of 2 x 10E8 Rads. The radiation resistance for the replacement seal components exceeds the worst case accident radiation environment or 6.79 x 10E6 as described for environmental zone RC-291B (outside crane wall). The replacement seal materials envelope all other parameters described by environmental zone RC-291B.

Operation, function and periodic testing of the containment air locks will not be changed by replacement of the teflon sealing components in the air equalizing valves and escape hatch shaft. The potential for the seals to completely fail is unlikely since they are confined by metal parts. The potential leakage path provided by a seal failure is very small due to the close clearance design. In addition, the air locks are designed with a double barrier (interior and exterior) pressure boundary to minimize the potential of any single leakage path compromising containment integrity.

Containment air lock leakage will be verified within the acceptance criteria of Tech. Spec. 3.6.1.3. No change to any Tech. Spec. Limiting Condition for Operation, Surveillance or Bases is required by this modification.

SAFETY EVALUATION NUMBER 91-SE-MOD-046

DESCRIPTION

This activity replaces the containment sump pumps 1-DA-P-4A&B, 2-DA-P-4A&B which will physically modify the sump pumps as described in UFSAR Table 9.3.3.

The sump pumps currently installed are failing at a high rate.

SAFETY ANALYSIS SUMMARY

The pump replacement should be allowed since the replacement pumps meet or exceed the material, performance and design criteria of the currently installed pumps. This EWR will revise the UFSAR to accurately describe the sump pump and account for the aluminum addition to the maximum allowable aluminum inventory inside containment. Though the currently installed pumps do contain aluminum components, this EWR will ensure that the aluminum mass and surface area are administratively accounted for. The additional aluminum will not increase the aluminum inventory so as to exceed the maximum allowable inventory of aluminum inside containment.

SAFETY EVALUATION NUMBER 91-SE-MOD-047

DESCRIPTION

This modification installed suction pressure gauges for each Boric Acid Transfer Pump (1-CH-P-2A, 2B, 2C, and 2D). These gauges are required to meet the pump test requirements of ASME XI. Although the gauges are for ASME XI Testing only, the Root Valves will be open for ALARA concerns and equipment operability reasons. The Root Valves are considered accessible isolation valves and the installation is seismically qualified.

SAFETY ANALYSIS SUMMARY

A Safety Evaluation (ADM 3.9) was completed and revised as part of the EWR series. The modification was intended for pump testing purposes only and will not impact normal system operation. The installation meets the requirements of ISA Standard ANSI/ISA-567.02-1980 entitled "Nuclear Safety Related Instrument Sensing Line Piping and Tubing Standards for use in Nuclear Power plants." This standard is accepted by the NRC as described in Technical Report No. EE-0012 Rev. 1. This modification does not involve an unreviewed safety question.

SAFETY EVALUATION NUMBER 91-SE-MOD-048

DESCRIPTION

The sensing line for the Emergency Diesel Generator low starting air pressure switches was moved from downstream of the header check valve to upstream of the check valves. This provided a more accurate indication of the pressure in the air receiver. This modification was performed for all four of the Emergency Diesel Generators.

SAFETY ANALYSIS SUMMARY

The Air Start System and consequently the Emergency Diesel Generator were not adversely affected. The Air Start System functions as before, however, by relocating the sensing line the pressure switch provides more accurate information. The Air Start System has two redundant trains. The EDG function is unchanged by the modification and the likelihood or consequences of failure are not increased.

SAFETY EVALUATION NUMBER 91-SE-MOD-049

DESCRIPTION

This design change will modify the control circuits for the Main Steam Trip Valves 1-MS-TV-111A/B and 2-MS-TV-211A/B such that it will be necessary to provide a manual "Close" command to the valves in order to stop the steam driven AFW Pumps 1, 2-FW-P-2.

This change is intended to provide control response for the AFW Pumps 1, 2-FW-P-2 that is similar to the motor driven pumps in that a manual command is required to stop them after an automatic start. This is to provide a more controlled recovery of plant conditions after an event is brought under control.

SAFETY ANALYSIS SUMMARY

The major issues considered were: On November 2, 1990, while returning to power following a refueling outage, Unit No. 2 experienced an automatic reactor trip from about 13 percent reactor power. This generated RCE 90-0006 which recommended control circuitry be modified so that operator action is required to secure the Turbine Driven Auxiliary Feedwater Pumps. See DCP Appendix 8.3 for LER 90-10-00 and Root Cause Evaluation 90-0006. The reason the change should be allowed: Changing the controls for the 1,2-FW-P-2 Pumps to require manual stop commands will provide greater standardization in that it will make their operation similar to that of the Motor Driven AFW Pumps. This simplifies the operators environment during an event. Also, requiring a manual shut down of the Turbine Driven Pumps allows the operator more positive control of the systems while recovering from an event. The modification to the Main Steam Trip Valve control circuits will add an auxiliary relay. The controls will operate as they did before with the exception that there will be no automatic circuit reset after initiation of an automatic start. The operator will have to make a manual close command. This will enhance the availability of the pumps. The Systems will perform their Safety Related functions as they did before the modification. During the modification, the affected Main Steam Trip Valve will be unavailable. Only one train per Unit will be affected at a time. The addition of this auxiliary relay will be made "in kind" with the philosophy used during the original design. This circuit is a "fail safe" circuit that requires components to be available to prevent the operation of the steam driven feed pump. The steam turbine will start for the following conditions: loss of 125VDC control power, loss of air to the Main Steam Trip Valves and/or the failure of a required coil. This will continue to be true after the modification. Should the auxiliary relay coil fail, the steam driven feed pump will start. In the unlikely event that the auxiliary relay fails to open its contact, the circuit will operate exactly as it has in the past. The control, function and operating conditions of the Auxiliary Feedwater System will not be affected by the addition of this auxiliary relay. The modification in no way affects the availability of the system.

SAFETY EVALUATION NUMBER 91-SE-MOD-050
91-SE-MOD-066

DESCRIPTION

These modifications have installed a Rosemount transmitter, 3-valve manifold, and associated tubing for flow transmitter 02-SW-FT-205. This flow transmitter is utilized to monitor flow in the auxiliary Service Water supply piping. This indication is supplied to the Control Room. In addition, these EWRs downgraded the flow transmitter from SR to NSQ.

SAFETY ANALYSIS SUMMARY

The Rosemount transmitter, 3-valve manifold, and associated tubing installation has not affected the operation of auxiliary Service Water supply piping or the Service Water system. System design bases for the operation of the Service Water and flow transmitter 02-SW-FT-205 remain unchanged. Thus the safety of the Service Water system is unchanged.

SAFETY EVALUATION NUMBER 91-SE-MOD-051

DESCRIPTION

Replacement of an older model solenoid operated valve (SOV) which is no longer manufactured, with a new model.

The existing SOV has failed and needs to be replaced.

SAFETY ANALYSIS SUMMARY

The major issues considered were whether the replacement SOV would function within the guidelines of the original design basis.

The reason(s) the change should be allowed is because the replacement SOV has the same characteristics as the original except for wattage rating which will not affect functionality and a lower MOPD which has been found to be acceptable. The lower MOPD is acceptable because it still exceeds the system operating pressure. These exceptions have been evaluated and have been found insignificant.

An unreviewed safety question does not exist because:

- The SOV being installed will perform in the same manner as the existing and will perform its intended design function.

- The system will not be reconfigured nor will its operation be altered.

- The replacement SOV meets or exceeds all design requirements. It has been evaluated and found to be an acceptable replacement. The component/system will function as designed.

SAFETY EVALUATION NUMBER 91-SE-MOD-052

DESCRIPTION

Containment isolation check valve 1-FP-272 was replaced with a valve composed of a different material. The original valve was a TRW/Mission 4 inch Duo-Check valve Fig 155EF-X0 carbon steel body and plate. The new valve is a TWR/Mission 4 inch Duo-check valve Fig KISCMF-X20 stainless steel body and plate. Both valves weigh the same.

1-FP-272 failed Type C leakage test requiring replacement.

SAFETY ANALYSIS SUMMARY

Containment isolation check valve 1-FP-272 was replaced during the 1991 Unit 1 outage. A non-safety related replacement was procured from Virginia Power stock at another station and at the time was believed to be an identical replacement. However, it was later discovered that the original valve utilized a carbon steel body and plate while the replacement valve was composed of a stainless steel body and plate. The replacement should be allowed because of the following:

1. Both valves are identical except for material composition. Each valve is a 4 inch TRW/Mission Duo-Check valve.
2. The weight of each valve is 17 pounds. Therefore no new seismic concerns have been introduced by the replacement.
3. The replacement valve is composed of material which meets or exceeds the original design specifications.

The pressure rating on the valve should be lowered to 150 psig design since the valve was tested to 225 psig and system design piping pressure is rated for 150 psig.

SAFETY EVALUATION NUMBER 91-SE-MOD-053

DESCRIPTION

During the Self-Assessment prior to the NRC-Electrical Distribution System Functional Inspection (EDSFI), Item No. 120 was written reporting that the North Anna Setpoint Document (NASD) was not correct with regard to power protective relaying setpoints. These EWRs provide the methodology used to revise the NASD. The power protective relay settings are applied to the relays by the System Protection - Control Operations Group; these settings are taken from "White Sheets" initiated by System Protection. These EWRs immediately authorized System Protection "White Sheets" as the only source of power protective relay set points and revised the NASD. The NASD has been revised in two steps. The first step has been Addendum A to review the setpoints in Section G1 with an approved date of November 26, 1991 and the second step was Addendum B for Section G2 with an approved date of February 26, 1992.

SAFETY ANALYSIS SUMMARY

Updating the Setpoint Document to reflect the existing settings does not increase the probability or the consequences of an accident or malfunction of the equipment important to safety. In accordance with STD-GN-0030, calculations are required to update the Setpoint Document. Since preparing calculations for all setpoints in the subject sections of the document will require extensive time, the Setpoint Document has been updated by this change to correct the errors and the calculations will continue to be prepared. The changes have not adversely impact the Technical Specifications required equipment nor caused revision of the UFSAR. No known problems exist with the setpoints shown in the UFSAR or Technical Specifications.

Updating the Setpoint Document does not create the probability of an accident or malfunction of a different type than any evaluated previously in the UFSAR because no setpoints are being changed, and therefore, no equipment necessary for accident mitigation or safe shutdown is being modified.

The updating of the Setpoint Document to reflect existing settings does not reduce the margin of safety as defined in the bases of any Technical Specification.

SAFETY EVALUATION NUMBER 91-SE-MOD-054

DESCRIPTION

This Engineering Work Request controls the removal of Conductivity Sampling equipment associated with the Condensate Polishing System. The piping, valves, conductivity cells, etc. will be physically removed. Associated cables, recorders and annunicators will be abandoned in place and labeled as such.

The conductivity sampling equipment is outdated and is no longer being used. The equipment now used takes a sample at the local condensate polishing sampling sink and provides more accurate information.

SAFETY ANALYSIS SUMMARY

The Condensate Polishing System and associated conductivity sampling equipment are non safety related. The removal/abandonment of the conductivity sampling equipment does not alter the function of the Condensate Polishing System.

SAFETY EVALUATION NUMBER 91-SE-MOD-055

DESCRIPTION

Replacement components for Klockner-Moeller (K-M) motor control centers (MCC's) required a seismic review to establish the appropriate guidelines for mounting such replacement components onto existing mounting plates of K-M MCC's. Technical guidance was required for this effort since the mounting holes in the replacement components do not line up with the mounting holes of the obsolete components. It was necessary to drill new mounting holes, adjust component location within the cabinet slightly, install cover plates and cover existing openings in the MCC cabinet as adjustments to the new K-M components. This EWR was part of a Procurement Technical Review which approved the new K-M components as approved replacement parts.

SAFETY ANALYSIS SUMMARY

There were no unreviewed safety questions as a result of this EWR evaluation since no modification resulted via this document. The new K-M components became one-for-one replacement parts, except for the minor mounting adjustments that were mentioned above. These minor mounting adjustments have been shown to provide a mounting surface equal to or better than the original surface, without affecting the structural or electrical performance of the K-M MCC.

This EWR evaluation is unrelated to the probability of occurrence for any NAPS UFSAR Chapter 15 accident. Failure of the activities, governed by this evaluation, cannot increase the severity of consequences of previously analyzed accidents since they are bounded by NAPS UFSAR Chapter 15 accident analyses. The routine nature of the work associated with this evaluation, and fact that the new part is approved as a replacement part will ensure that no unreviewed safety questions remain.

SAFETY EVALUATION NUMBER 91-SE-MOD-056

DESCRIPTION

Ventilation for Demineralizer Alley sumps were provided during the reactor coolant and letdown filter changeouts. Vented covers were placed over the sumps with duct running to a 1000 cubic feet per minute (CFM) Flanders ventilation unit with prefilter and High Efficiency Postulate Absorber (HEPA) filter. The exhaust is then ducted to the auxiliary building ventilation system via polyvinye chloride (PVC) has hung from conduit/pipe supports.

Without ventilation, the Demineraliz[illegible]y sumps contain contaminated particles and gases that can escape during reactor coolant and letdown filter changeouts. Personnel are required to visually verify that flow to the sumps has ceased during these changeouts. This Engineering Work Request (EWR) reduced the risk of a personnel contamination event (PCE) during filter changeouts.

SAFETY ANALYSIS SUMMARY

The major issue involved in this EWR is providing adequate ventilation for the (5) Demineralizer Alley sumps during reactor coolant and letdown filter changeouts. The particular resolution, chosen in this EWR, has minimal impact upon neighboring systems, structures and components. The plexiglass sump covers are simple, light weight and positioned such that they don't interfere with any safety related equipment. The PVC duct is flexible, light weight and can be tye-wrapped to conduit/pipe supports along the route without adding significant deadload to the pipe, conduit or conduit/pipe supports. The motive force of ventilation is a modified 1000 CFM Flanders ventilation unit. This unit has been used in similar fashion before where its use has been met with success. The unit has a prefilter and HEPA filter which virtually eliminate all contaminants. The exhaust is still routed through the auxiliary building ventilation system charcoal and HEPA filters for added protection. All exhaust is monitored for contamination according to 10 CFR, Part 100 and Technical Specifications. Since this modification is self-contained, imparts insignificant loads to safety related equipment, and does not affect the margins of existing safety related systems, structures or components, an unreviewed safety question does not appear to exist for this EWR.

SAFETY EVALUATION NUMBER 91-SE-MOD-057

DESCRIPTION

The settings for the Emergency Bus 72% Undervoltage Timers for both Units 1 & 2 are shown in the Setpoint Document as 2.0 seconds. These settings are also shown as 2.0 seconds on Drawing 11715/12050-FE-21T/21U. This did not agree with either Units' Technical Specifications which states that the setting should be 2.2+/-0.03 seconds. These EWRs revised the System Protection "White Sheets", authorized by EWR 91-112, to reflect the Tech. Spec. setpoint. It also revised the drawings and other pertinent documents.

SAFETY ANALYSIS SUMMARY

Revision of the 72% Undervoltage relay time settings from 2.0 to 2.2 seconds provides agreement between the Technical Specifications and the Setpoint Document. The total EDG response time required in Tech Specs (13.3 sec.) was not changed; only one time delay was changed which contributes to the total EDG response time.

There are no unreviewed safety questions because:

> This setpoint change did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety since this setpoint change provides agreement with Tech Specs.
>
> The revision of the setpoint did not create the possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR. The revised setpoint leaves the relay configured such that single failure criterion still bounds any postulated malfunctions.
>
> These EWRs have not reduced the margin of safety as defined in the basis of the Tech Specs because the setpoint change provides agreement with the Tech Spec settings.

SAFETY EVALUATION NUMBER 91-SE-MOD-058

DESCRIPTION

Removal of a 120 volt alternating current (VAC) circuit from semi-vital bus 1A; Panel 1-EP-CB-16A, Circuit Breaker #24.

Above circuit is presently feeding an unmarked receptacle in the service building hallway, as well as outlet strip in the Chemistry Office area. Receptacle and outlet strip are not required to be on the semi-vital bus.

SAFETY ANALYSIS SUMMARY

The major issues considered were the possibility of adverse impact to the semi-vital bus 1A, and other breakers located in panel 1-EP-CB-16A.

The reason the modification should be allowed is that there is no existing equipment in the Chemistry Lab, secondary side sample sink area which requires power from an emergency bus. Additionally, the 120 VAC receptacle in the service building hallway is not identified as being powered from a semi-vital bus. Removal of this circuit from the semi-vital panel will decrease the potential for any adverse impact to the 1A semi-vital bus. A new power supply for the Chemistry area will be obtained from a lighting panel in the Chemistry/Health Physics Office area.

SAFETY EVALUATION NUMBER 91-SE-MOD-059

DESCRIPTION

[illegible] settings for the Emergency Bus 90% Degraded Voltage Timers, 62, were shown in [illegible] Setpoint Document for Unit 1 as 63 +/-5% Seconds and were not shown for Unit 2. This did not agree with either Units' Technical Specifications which state that the setting should be 60+/-3 Seconds. This EWR revised the G1 and G2 Sections of the [illegible] point Document to 60+/-1 Seconds in order to insure compliance with the Tech. [illegible] setpoint. These setting were verified to be in compliance with Tech. Specs. by [illegible] This timer setting is included in Calculation EE-0036.

[illegible] timer for the 90% Degraded Voltage with Safety Injection was added to the [illegible] Document. This time is 7.5 Seconds +/-0.75 as stated in Calculation EE-[illegible] To provide better control of the possible adjustments of the 62S timers, their [illegible] rification was added to the tests presently included in 1/2-PT-36.11. In this way [illegible] nsion of the time setting is directly related to the overall EDG response times which are Tech. Spec. requirements.

SAFETY ANALYSIS SUMMARY

Revision of the Setpoint Document to show 90% Degraded Voltage relay time settings as 60 +/-1 seconds provides agreement with the Technical Specifications. The total EDG response time required did not change.

There are no unreviewed safety questions because this Setpoint Document change does not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety since this setpoint change provided agreement with Tech. Specs.

The revision of the setpoint does not create the possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR. The revised setpoint leaves the relay configured such that single failure criterion still bounds any postulated malfunctions.

This EWR does not reduce the margin of safety as defined in the basis of the Tech. Specs. because the setpoint change provided agreement with the Tech. Spec. settings.

SAFETY EVALUATION NUMBER 91-SE-MOD-060

DESCRIPTION

Replacing the air drive system for moving the Fuel Transfer car with a winch and cable system to improve the Fuel Transfer System Reliability.

SAFETY ANALYSIS SUMMARY

This safety evaluation was performed because a UFSAR change is required where it refers to the means for Fuel Transfer Car movement as being an "air motor". The air motor is being replaced with a winch/cable system and new above water limit switches which are more sensitive and reliable. These features are improvements in design and therefore do not constitute an unreviewed safety question.

SAFETY EVALUATION NUMBER 91-SE-MOD-061

DESCRIPTION

Main Steam Valve 2-MS-95 will be replaced with a similar valve.

The valve has been repaired numerous times and continues to leak.

SAFETY ANALYSIS SUMMARY

Valve 2-MS-95 is being replaced with a similar valve due to numerous repairs and leaks with the existing valve. Both the new and existing valves are 600 pound Class 2 Gate valves made of carbon steel. The only real difference is that the new valve is an Anchor/Darling valve and the existing one is a Walworth. The new valve will function the same as the existing one. The new valve has been acceptable from a seismic concern.

SAFETY EVALUATION NUMBER 91-SE-MOD-062

DESCRIPTION

Sixteen one foot by three foot lead blankets are to be tye-wrapped and stainless steel (S.S.) banded around line 8".51.40.153A.Q2 in the reactor purification (RP) pump cubicle of the auxiliary building at 244 foot level. The lead blankets are to be installed in 2 layers on the 8 foot run of pipe and fastened together with tye-wraps and secured with 2 S.S. bands per blanket.

Based on as low as reasonably achievable (ALARA) studies, an overall 35% reduction in radiation levels within the RP pump cubicle can be expected.

SAFETY ANALYSIS SUMMARY

The major issue to be considered here is the impact of attaching (16) 1 foot x 3 foot lead blankets to 8 feet of line 8".51.40.153A.Q2 in the RP pump cubicle of the auxiliary building, elevation 244 feet. The lead blankets will wrap around the pipe and be tye-wrapped together to keep them from shifting. To prevent any long-term degradation, two S.S. bands shall circumferentially secure the blankets to the pipe. As such, the blankets become rigidly attached to the pipe. Calculation CE-0874 has verified that line 8".51.40.153A.Q2 can withstand operating basis earthquake (OBE)/design basis earthquake (DBE) events, within code stress limits, with the shielding in place. With the blankets rigidly attached to the pipe, the shielding becomes passive with respect to other equipment, hence there are no other unreviewed safety questions to be created.

SAFETY EVALUATION NUMBER 91-SE-MOD-063

DESCRIPTION

Permanent lead blanket shielding weighing 60 pounds per foot is being placed around line 2"-CH-264-602-Q2 in the Unit 1 pipe penetration area, auxiliary building, elevation 244 feet 6 inches.

To reduce the general area dose rates for routine jobs in the Unit 1 pipe penetration area, auxiliary building, elevation 244 feet 6 inches.

SAFETY ANALYSIS SUMMARY

The lead blanket shielding will be rigidly attached to the pipe creating no additional harmonic motion. The pipe may remain in service as dead load and seismic conditions were evaluated. General area dose rates will be reduced at no sacrifice to margin of safety. Also, minor amount of combustibles due to the installation of permanent lead blanket, shielding are added which do not change the area's (Area 11) fire severity classification and no impact to Appendix "R" equipment or program due to shielding. Changes to Chapter 8, combustile loading analysis initiated in accordance with STD-GN-0021.

SAFETY EVALUATION NUMBER 91-SE-MOD-064

DESCRIPTION

A drain line will be added to the existing fume hood located in the Drumming Room. As this drain line enters the 259'-6" floor elevation of the auxiliary building it is seismically supported to prevent seismic interaction with any safety related equipment.

To prevent seismic interaction (two over one seismic criteria) with safety related equipment within the collapsible envelope of the drain line route @ 259'6" floor slab elevation in the auxiliary building by seismically supporting it.

SAFETY ANALYSIS SUMMARY

The Drain Line Route on the 259'-6" elevation floor of the auxiliary building installed for the existing fume hood located in the drumming room is non-safety but supported seismically. The only reason that this drain line is supported seismically at this elevation is because of its interaction with any safety related component within its collapse envelope.

SAFETY EVALUATION NUMBER 91-SE-MOD-065

DESCRIPTION

The heating ventilation and air conditioning (HVAC) purge restraints, 1-SM-264 & 265 and 2-SM-167 are to be revised to eliminate any linkages that may exist between the Rod Drive Control rooms, where they are located, and the containment walls. External support legs are eliminated, member sizes are increased (1-SM-264 & 2-SM-167 only) and greater clearance provided.

HVAC purge restraints, 1-SM-264 & 265 and 2-SM-167 are linking the Rod Drive Control rooms of the Auxiliary Building to the external side of containment walls. These buildings are built on separate foundations with a 2" rattle space to accommodate relative building seismic displacement. This design change will release the linkage.

SAFETY ANALYSIS SUMMARY

The main technical issue to be addressed in this design change is the potential for adverse effects to the Containment Air Supply and Purge system due to the existing HVAC support arrangement. The Containments and Rod Drive Control rooms are supported on separate foundations with a 2 inch rattle space to permit relative building displacements during a seismic event. Any linkage between the two buildings must be evaluated to ensure that it does not rigidly link the buildings together, nor create a condition of overstress, with the linkage, due to postulated relative displacements between the two buildings. It is obvious that the linkage caused by these existing HVAC supports will not rigidly link the buildings together, however; the linkage does create a condition of overstress, within in all three HVAC supports, during relative seismic displacements of the two buildings (reference Calculation Nos. CE-0876, Rev. 0 and CE-0842, Rev. 0). To eliminate the conditions of overstress, support modifications are required.

HVAC support member and steel duct stresses were analyzed and maintained within the AISC, 8th Edition allowable limits. Piping and pipe penetration stresses were limited to meet the requirements of AISI B31.7-1969 with addenda through 1970. For details of these analyses, see Calculation Nos. CE-0842, Rev. 0 and CE-0876, Rev. 0.

To eliminate the linkage between the Containment wall and the Rod Drive Control rooms, the short angle lateral brace was removed from supports 1-SH-265 and 2-SH-167. The removal of this lateral brace created higher member loads for these supports and in the supports directly above them (i.e. 1-SH-266 and 2-SH-168, reference drawings 11715-FB-7G and 12050-FB-7G, respectively). Accordingly, member sizes need to be increased for supports 1-SH-265 and 2-SH-167 in order to accommodate the increased loads at the same margin of safety. See sketches N-91125-3-S-001, SH 1 & 2 of 2 for details.

The angle member of 1-SH-264, which is adjacent to the column in the Unit 1 Rod Drive Control room, shall be removed to allow the mating bracket to be coped. The existing holes in the mating bracket shall be elongated to provide sufficient clearance (i.e. 7/16" minimum) between the angle member and the column. A minimum clearance of 7/16" will prevent the support from hitting the column during a seismic event. For details of this modification, see sketches N-91125-3-S-002, SH 1-3 of 3.

SAFETY EVALUATION NUMBER 91-SE-MOD-067

DESCRIPTION

This EWR allowed the interchanging of the Diesel Generator (EDG) output voltage adjust switch handle with the EDG voltmeter selector switch handle. Only the switch handles were involved in this change, the switch mechanism and wiring was not impacted. The output voltage adjust handle is currently a "pistol grip" and the voltmeter selector handle is a knurled knob. This was a human performance enhancement to the diesel generator control panels in the main control room.

SAFETY ANALYSIS SUMMARY

An unreviewed safety question does not exist because interchanging the EDG output voltage adjust and voltmeter selector switches did 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. The switches themselves, and any associated wiring circuits have not been altered when the handles were interchanged. The handles did not affect the diesel start circuits or loading sequencing.

The possibility of an accident or a malfunction of a different type than any evaluated previously in the UFSAR was not created because only the handles were interchanged. No circuitry important to the auto or manual start capability was altered. The switch function was not changed, and no additional modifications to the handles (to ensure that they will be securely attached to the switch mechanism) were necessary. Both handle types (pistol grip and knurled knob) are qualified for the SBM type switch mechanism. All accidents involving use of emergency power supplies were considered during the engineering evaluation.

SAFETY EVALUATION NUMBER 91-SE-MOD-068

DESCRIPTION

Welding receptacles will be mounted in the Main Steam Valve houses. This will involve seismic conduit run through each of the Rod Drive rooms and in the Main Steam Valve houses (MSVHs).

Extension lords are presently used every time a welding machine is used in the MSVHs.

SAFETY ANALYSIS SUMMARY

The major issues considered are the following:

1) The ability of the recept. or conduit to harm surrounding equipment in the case of a seismic event, and

2) The breach of a fire barrier.

It was decided that the seismically mounted conduit and receptacles will have no potential to harm any surrounding equipment because seismic mounting is designed to withstand a design basis seismic event. In the case of the fire barrier, a fire watch will be posted while the barrier is opened, then it will be resealed. This presents no safety threat.

In conclusion, this modification does not constitute an unreviewed safety question. The ability of the mounted equipment to withstand a seismic event and the opening and resealing of a fire barrier have both been considered.

SAFETY EVALUATION NUMBER 91-SE-MOD-069

DESCRIPTION

Replace the existing undersized pressurizer heater cables with larger cables. Replace the existing 70A fuses and 90A current limiting fuses in the pressurizer heater circuits.

Unit 2 has been experiencing problems with fuses and circuit breakers failing in the pressurizer heater circuits. Inspections show excessive heat being generated by the cables. The heat is being transferred through the conduits into the fuse cabinets and breaker panels. The insulation on the cables in some places is stiff and brittle from the heat. Calculations show that the cables are undersized.

SAFETY ANALYSIS SUMMARY

The major issues and reasons for the change are as follows:

The inadequate cable size should be repaired to provide for a safe reliable and operable Pressurizer Heater System. After implementation of this change the nuisance trips of the circuit breakers and the failure of the fuses should be eliminated. This will be accomplished without jeopardizing the safety or reliability of the plants systems, components or structures.

The above referenced devices will not adversely impact on the Tech. Spec. required equipment nor cause revision to the UFSAR.

The modification is not adversely impacting any equipment necessary for accident mitigation or safe shutdown.

An unreviewed safety question does not exist.

SAFETY EVALUATION NUMBER 91-SE-MOD-070

DESCRIPTION

A jail bar type cover will be installed in the opening below the platform leading into the spent resin dewatering tank cubicle. A seismic evaluation of this change was performed.

This change will preclude access to an area which is normally accessed through a locked, high radiation area door above the platform.

SAFETY ANALYSIS SUMMARY

A jail bar type cover will be installed over the opening below the platform to the spent resin dewatering tank. The cover will consist of two angles bolted with Hilti Kwik Bolt II's to the concrete walls on either side of the opening. Four bars will be welded between the angles and run horizontally. One bar will be welded in the middle connecting all four horizontal bars and run vertically.

A seismic evaluation of the jail bar cover shows that the cover will not be damaged, fall, or affect Safety Related systems or components in the event of an Operating Basis Earthquake or Design Basis Earthquake.

SAFETY EVALUATION NUMBER 91-SE-MOD-071

DESCRIPTION

Installation of new station UHF radio antennas and associated cables in the Turbine, Auxiliary, and Containment Buildings. This is non-operational system and is support for a future DCP that installs a new station UHF transmitting/receiver system to support an upgraded Station UHF radio system in the future.

SAFETY ANALYSIS SUMMARY

This change will provide for the installation of antennas and associated cable to support an upgrade to the Station Radio System via a future DCP (91-14-3). The concern of attaching the antennas to safety related walls and the containment crane access platform has been addressed in Calculation CE-0866.

An unreviewed safety question does not exist.

SAFETY EVALUATION NUMBER 91-SE-MOD-072

DESCRIPTION

Instrument Air Volume Tanks for Service Water Trip Valves, 1-SW-TV-101A, B and 2-SW-TV-201A, B will be eliminated.

The Instrument Air volume tanks for 1,2-SW-TV-101, 201A & B are not ASME VIII certified nor are they part of any maintenance or testing program. Due to the reliability of the Instrument Air system and that the containment air recirculation system has no ESF functions, the tanks are not required.

SAFETY ANALYSIS SUMMARY

The volume tanks associated with the Service Water trip valves 1,2-SW-TV-101,201 A&B are not required and should be eliminated because: the Instrument Air system is a highly reliable system (power supply from the H emergency bus) to provide sufficient air for the valves, loss of a cooling medium to the containment air recirculation cooling coils would require a simultaneous loss of both the chilled water system (station blackout) and the Instrument Air to prevent the trip valves from opening to introduce service water to the cooling coils, containment air recirculation system is used during normal unit operation and after Condition II and III accidents to remove heat from the containment structure. Since this system is not used after Condition IV accidents, it is not considered an Engineered Safety Feature (ESF). Quench Spray and Recirc. Spray are the only ESF that are used after a LOCA or Main Steamline Break inside containment to remove heat from the containment structure, the Instrument Air volume tank's reliability remains uncertain since the tanks are not included in the stations maintenance, test or ISI programs, the highly unlikely event of a total loss of the air recirculation system will not prevent safe shutdown of the plant, and the containment isolation return trip valves for the recirculating air cooling coils (1,2-CC-TV-100,200 A,B,C and 1,2-CC-TV-105,205 A,B,C) will fail close on a loss of Instrument Air. These valves are not provided with a backup air supply volume tank so that on a loss of Instrument Air, the valves will close and prevent flow through the cooling coils. In the event that the service water trip valves were to remain open on a loss of Instrument Air due to the volume tank backup air supply, flow through the recirc air cooling coils would not be possible due to the fact that the containment isolation valves for the cooling coil return piping would fail close.

SAFETY EVALUATION NUMBER 91-SE-MOD-073

DESCRIPTION

Key-lock switches will be added in series with each of seven turbine trip input channels to the Reactor Protection System. Three key-lock switches will be added in series with the three auto stop oil pressure indication channel, and four will be added in series with the four stop valve indication channels.

This change will allow any of the seven channels discussed above to be easily put into "trip" condition. This will be done because Technical Specification 3.3.1 1 requires that if these input channels fail or require maintenance, they shall be placed in "trip" condition. Currently the only way to accomplish this is by jumper.

SAFETY ANALYSIS SUMMARY

The major issue considered is that of the turbine being in "trip" condition without the reactor tripping while at 30% power or above.

The installation of key-lock switches cannot cause such an accident to occur. Failure of key-lock switch could, at the worst, cause a "trip" signal to be received from the particular channel involved. A key-lock failure could not prevent a turbine trip from registering a "trip" signal on the corresponding channel.

The system involved consists of low-voltage, isolated circuits that could not cause harm to any of the surrounding systems. An unreviewed safety question does no exist.

SAFETY EVALUATION NUMBER 91-SE-MOD-074

DESCRIPTION

Changing the Unit 2 4160/480V Load Center Transformer Tap setting from 4160V to 4056V to increase output voltage to be in accordance with the GDC-17 Analysis and Calculation EE-0008.

Correct field conditions to match the analysis of GDC-17. Transformer setting discrepancies were found during the NRC EDSFI inspection and noted in Deviation Report N-91-1243. Unit 2's settings were review and found acceptable, but Corporate Electrical Engineering recommended changing the settings to provide additional margin for equipment operation.

SAFETY ANALYSIS SUMMARY

The Emergency Bus Transformer Tap Setting Changes will provide a 2.5% boost in the voltage of the 480V busses supplied by the transformers. These tap changes are being completed to provide transformer tap setting that agree with settings used in the calculations for the GDC-17 analysis. These transformer tap changes do not create the possibility of a new or different kind of accident from any accident previously evaluated since the tap settings are enveloped by the electrical design analysis. There is no increase in the probability of an accident since transformer tap settings do not impact the probability of accidents evaluated. Consequences of accident are reduced since the transformer tap settings provide additional margin for operation of the equipment is provided by the additional voltage. Fault analysis is not impacted since the proposed new tap settings were are in the existing fault calculations.

SAFETY EVALUATION NUMBER 91-SE-MOD-075

DESCRIPTION

The modification will enhance the reliability and operation of the Service Water radiation monitor RM-SW-109. Activities will consist of rerouting the detector discharge piping to the east side of the service water valve house, installation of a power supply regulator for the control room ratemeter and permatizing the jumper out of the antijamming device. The elimination of the antijamming device involves analysis of the overall Service Water and Radiation Monitor systems design.

The purpose of the discharge piping reroute is to enable easy verification of flow through RM-SW-109. Presently, this flow can only be observed from the Service Water Valve House roof or from the other side of the Service Water pond. The former is impractical and the latter is only possible during daylight and good weather. The power supply regulator is installed to make the ratemeter less vulnerable to Electro-Magnetic Interference, lightning and power supply noise. This vulnerability has contributed to nuisance alarms chronically since its initial installation. The permatization of the jumper out of the antijamming circuit will minimize the recovery time from any nuisance alarms because no fuse will be blown.

SAFETY ANALYSIS SUMMARY

The major issues considered were: The modification to enhance the reliability and operation of the Service Water radiation monitor RM-SW-109. Activities considered were: rerouting the detector discharge piping to the east side of the SWVH. This is done to provide a means for verifying desired flow through RM-SW-109, installation of a power supply regulator for the control room ratemeter. This is done to eliminate power supply noise induced false alarms on the ratemeter, permatizing the jumper out of the antijamming device. The elimination of the antijamming device involves analysis of the change with respect to the operation of the ratemeter under different inputs.

The three elements considered will tend to make the RM-SW-109 subsystem more reliable without compromising any of the necessary design features. The measures taken to improve reliability and provide flow indication will enhance the existing design. RM-SW-109 is not evaluated in the UFSAR. Radiation monitoring of the Service Water system is done primarily by other monitors. Design of the flow indication will be using sound engineering practices. The power supply regulator is a common and prudent method for eliminating noise related problems with sensitive instrumentation. The elimination of the antijamming feature only removes nuisance alarms on the high end which is redundant to other detectors. These other detectors are not susceptible to the noise sources with plague RM-SW-109. High activity rates in the Service Water would not go undetected. In the event that the ratemeter went offscale the digital readout would show all "E"s and an LED would be lit to show "Range" (i.e. offscale high). These positive indications to the operators would direct the operators to enter the appropriate response procedures (AP 5.1).

SAFETY EVALUATION NUMBER 91-SE-MOD-076

DESCRIPTION

The design change will remote each of three Steam Generator Blowdown hand control valves' downstream piping and have it discharging to the Steam Generator Blowdown tank separately. A new sparger pipe will be provided for each Steam Generator Blowdown line in the tank. The purpose of the design change is to minimize the erosive effect of Steam Generator Blowdown flow by shortening the hand control valve downstream piping and adding sparger to divert the discharging energy of Steam Generator Blowdown flow.

SAFETY ANALYSIS SUMMARY

The only accident evaluated in the UFSAR, which may relate to the portion of the SG Blowdown system that is affected by this design change, is the requirement to manually terminate the steam-generator blowdown (Section 15.4.3.1). The design change will not violate this requirement.

The changes will be consistent with the affected system's design basis. The modifications made do not change the operation or ability of equipment important to safety to perform their safety function.

Based on the above, an unreviewed safety question does not exist.

SAFETY EVALUATION NUMBER 91-SE-MOD-077

DESCRIPTION

Replace the existing undersized pressurizer heater cables with larger cables. Replace the existing 70 ampere fuses and 90 ampere current limiting fuses in the pressurizer heater circuits. Change the existing circuit breakers with temperature compensating breakers (All new cable will be routed in tray).

Unit 1 has been experiencing problems with fuses and circuit breakers failing in the pressurizer heater circuits. An inspection showed excessive heat being generated by the cables. The heat is being transferred through the conduits into the fuse cabinets and breaker panels. The insulation on the cables was also identified to be stiff and brittle from the heat. Calculations showed that the cables are undersized.

SAFETY ANALYSIS SUMMARY

The major issues and reasons for the change are as follows:

The inadequate cable size should be repaired to provide for a safe, reliable and operable Pressurizer Heater System. After implementation of this change the nuisance trips of the circuit breakers and the failure of the fuses should be eliminated.

The replacement of cables with larger cables, use of current limiting fuses and temperature compensating circuit breakers will not adversely impact the Technical Specification required equipment nor cause revision to the UFSAR and is not adversely impacting any equipment necessary for accident mitigation or safe shutdown.

SAFETY EVALUATION NUMBER 91-SE-MOD-078

DESCRIPTION

The in-containment structural modifications are performed in preparation for the steam generator replacement project. The modifications consist of cutting the bioshield walls around each steam generator (SG), removal of a portion of the operating floor in front of the equipment hatch, and enlargement of the opening in the crane wall to provide clearance for the SG lower assemblies. The change also consists of drilling and installing anchor bolts on the operating floor. SG level transmitter tubing and tubing supports will be detached to allow for the biowall cutting and then reattached.

The equipment hatch platform will be permanently lengthened by 6 feet and widened by 5 feet. A permanent staircase will be erected to replace the existing ladder, and a jib crane will be added. The platform will also be temporarily lengthened to accommodate the SG lower assemblies. Following the steam generator repair (SGR), the temporary platform will be removed and the tugger cable will be relocated.

The equipment hatch barrel floor will be removed to provide the clearance needed for the removal and reinstallation of the SG lower assemblies. A new floor will be installed in the equipment hatch barrel to allow the movement of other equipment. The new floor will become a permanent part of the plant configuration.

An electrical outlet, conduit 1CK907NN3 and receptacle 124, will be relocated permanently from the section of the steam generator A biowall that is being removed to the wall adjacent to the polar crane wall opening. Conduit 1CK907NN2 and the outlet, receptacle 122 will also be relocated permanently from the portion of the steam generator C biowall that is being removed to the same wall, but closer to the pressurizer. A section of concrete at the top of the polar crane wall opening is being removed. A light fixture is located in the area above the polar crane wall opening. The light fixture will be relocated permanently above this opening by shortening the conduit that extends down to the fixture from a tee conduit. Design drawings that show the affected electrical components are N-9015-1-1FE46A, N-9015-1-1FE46B, and N-9015-1-1FE67D.

These modifications are performed to provide clearance for passage of the existing SG lower assemblies and for installation of the new lower assemblies during the SGRP outage.

SAFETY ANALYSIS SUMMARY

The major issues considered for the in-containment modifications follow:

° Could the cutting and removal of the bioshield wall, crane wall, and operating floor significantly affect the design performance of the containment?

- Could the wall replacement affect any of the results of the design basis accident analyses as discussed in the UFSAR?

- Could the cutting, reinstallation, and drilling of anchor bolts lead to a significant increase in radiation doses?

- Will the auxiliary crane and runway beam anchor bolts affect the seismic or structural integrity of the floor slab?

- Will the installed instrument tubing function in the same manner as previously and will it be as reliable?

Upon evaluation of these issues, it was concluded that the activities covered by this safety evaluation can be conducted without undue risk to the health and safety of the public and that this design changes does not create an unreviewed safety question as defined by 10 CFR 50.59. These conclusions rest on the following major points:

- The containment structure is designed to sustain, without loss of required integrity, all effects of gross equipment failures up to and including the rupture of the largest pipe in the reactor coolant system and any condition resulting from a LOCA. The in-containment structural modifications do not affect the performance or integrity of the containment. The as-left conditions after the biowall and operating floor cutting will be similar to the present conditions, and there will be no effect on the design performance. The replacement of the operating floor with a structural steel platform does not affect the seismic loads. An analysis was performed which showed the slab as accepting the same loads as prior to the cutting. The biowall sections will be restrained to ensure that they meet their structural and biological shielding design as well as all other design criteria. An analysis was performed showing no effect on seismic and wall loading. The permanent enlargement of the opening in the crane wall, which creates a reduction of the wall depth by 1 foot, is insignificant in terms of the wall structural capacity. Based on the stress distribution through the depth of the wall due to the loads on top, the bottom few feet of the crane will make no contribution to the wall's structural capacity. An analysis shows no effect on seismic and crane loading.

- The in-containment modifications do not affect any results of the design basis accident analyses as discussed in the UFSAR. The reinforced-concrete walls and slabs are provided for biological shielding. Results of the design basis accident analyses remain valid.

- The cutting of the in-containment structures will not lead to a significant increase in radiation doses. The cutting will be accomplished using a diamond wire saw. This cutting technique is faster than conventional methods and creates no significant airborne dust or rubble.

- The modifications associated with the auxiliary crane and runway anchor bolts do not involve cutting rebar and, therefore, a seismic analysis was not necessary.

- The level transmitter will have the same configuration and will not affect the function. Reliability will not be affected because the connection process used to reinstall the tubing will be identical to the process previously used. Some tubing supports will be required to be welded to the steel plates in lieu of the original bolt-to-concrete configuration. The installation will meet plant specifications to ensure seismic installation.

The major issues considered for the equipment hatch platform modifications are:

- Whether the platform modifications would adversely affect the containment building, equipment hatch, or any other structures, systems or components important to safety; and

- Will the modified platform be capable of supporting the SG lower assembly load.

Upon evaluation of these issues, it was concluded that the activities covered by this safety evalaution can be conducted without undue risk to the health and safety of the public. It was also determined that this activity will not results in an unreviewed safety question as defined in 10 CFR 50.59. These conclusions were based on the following:

- The platform is not physically connected to the equipment hatch, containment structure, or any other structures, systems, or components. The platform does support the weight of the equipment hatch missile shields. The platform has been proven to withstand tornado loads up to 360 mph (Calculations 11715-Book BK-5AF and 11715-Book BK-5AH). Seismic design of the platform is not required. It is a self-supportive structure; thus, it will not adversely affect any structures, systems, or components. Support of the equipment hatch missile shield will not be altered by the modification.

- The jib crane will not have any seismic requirements. An evaluation shows that the effects of the crane as a missile will not result in a condition worse than the utility pole missile postulated in UFSAR Section 3.5.4. Therefore, the crane as a missile will not adversely affect the containment structure or the missile shield.

- Calculations C106-01 and C106-02 determined that the modified platform would be capable of supporting the SG lower assemoly load during the transfer process. The effect of wind loading on the platform is not significant.

- All the modifications in this activity will be conducted outside the containment building and will not affect any structures, systems, or components important to safety required to function at the time the platform is in the transfer process. Therefore, accidents and malfunctions analyses in the USFSAR will not be affected by this activity.

Upon completion of the SGR outage the temporary portion of the platform will be dismantled. The remainder of the platform will be acceptable as determined by Calculation C106-01.

The major issues considered for the equipment hatch barrel floor modification were the following:

- Whether the equipment hatch will be able to perform its intended function after the installation of a new floor.

- Whether the load-bearing capability of the equipment hatch barrel floor will be reduced.

- Whether containment integrity will be affected during the removal and replacement of the equipment hatch barrel floor.

- The movement of heavy loads during the removal and replacement of the equipment hatch barrel floor.

- Whether the equipment hatch barrel will sustain any damage during the modification process.

Upon evaluation of these issues, it was concluded that the activities covered by this safety evaluation can be conducted without undue risk to the health and safety of the public and that this design change does not create an unreviewed safety question as defined by 10 CFR 50.59. These conclusions are based on the following:

- The structural integrity of the equipment hatch barrel will be maintained.

- The load-bearing capability of the equipment hatch barrel floor will not be reduced because the new floor will have the same load-bearing capability as the original floor.

- Work regarding this design change will be done anytime containment integrity is not required to be maintained or when the LCOs of Tech. Spec. 3/4.9.4 are met. Therefore, containment integrity will not be affected.

- Heavy loads will be handled in accordance with 0-MCM-1303-01, "Moving Miscellaneous Heavy Loads and Çoncrete Floor Plugs In Çontainment during Unit Outage," to protect structures, systems, or components required to be functional during the removal and replacement processes.

- Detailed work procedures will be developed to ensure that the equipment hatch barrel is not damaged during removal of the existing floor. The equipment hatch barrel will be inspected after the work is completed to verify that it has not been damaged by the work, or that any incidental damage has been acceptably repaired.

The major issues for the electrical modifications are:

- Will the relocation of the conduits and receptacles introduce a new failure mode?

- Are the conduits and receptacles required for safe shutdown?

Upon evaluation of these issues, it was concluded that the activities covered by this safety evaluation can be conducted without undue risk to the health and safety of the public and that this design change does not create an unreviewed safety question as defined by 10 CFR 50.59. These conclusions rest on the following major points:

Electrical outlets, Conduits 1CK907NN2 and receptacles 124 and 122, are used by the reactor vessel stud tensioners. The light fixture is an overhead light that is being raised since the polar crane wall opening is being made higher. The conduits, receptacles, and light fixture are non-seismic, nonsafety-related electrical components that are not relied upon for accident mitigation and control, nor are they required for the safe shutdown of the facility. The relocation will be in accordance with specification NAS-2016 and NAS-3014. The proposed activity does not involve precursors to accidents described in the UFSAR. No new failure modes are introduced by relocating the electrical components. The relocation will use the same materials and no revision to the system and/or plant design basis documentation is required.

SAFETY EVALUATION NUMBER 91-SE-OT-001

DESCRIPTION

To add requirement for operable power source during movement of fuel when no fuel in core (not in any mode) and separate power supplies into two trains to shutdown operations.

This ensures available power in the event of a fuel handling accident and prevents confusion on what needs to be operable when shutdown.

SAFETY ANALYSIS SUMMARY

The proposed Tech Spec changes clarifies the Emergency Power Supplies which must be OPERABLE in Modes 5 and 6 and adds to the applicability the case of Mode 6 while moving fuel or heavy loads over fuel. This accommodates a fuel handling accident in the fuel building. The proposed Tech Spec ensures that equipment is available for this accident, and is therefore conservative.

Removal of the action requiring that CONTAINMENT INTEGRITY be established is acceptable for the following reasons:

1) When the unit is defueled, there is no need for containment integrity.

2) In Modes 5 and 6 there is no potential to pressurize the containment.

3) Action was added to suspend movement of irradiated fuel and movement of loads over irradiated fuel if minimum number of power sources or buses is inoperable. This Action will preclude any accident initiators that may require the containment to be sealed from the outside environment.

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

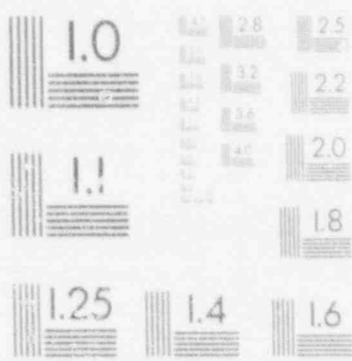
IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)



150mm

6"

SAFETY EVALUATION NUMBER 91-SE-OT-002

DESCRIPTION

Emergency use of Lake-to-Lake operation of the Service Water System to proceduralize requirements for conformance with Service Water Design Basis (Standing Order #177 "Service Water System Controls").

SAFETY ANALYSIS SUMMARY

The major issue considered with regards to Lake-to-Lake operation of the Service Water system was the effect Lake-to-Lake operation had on meeting the design flow requirements during a design basis accident. With the Service Water system aligned Lake-to-Lake, Service Water pumps 1/2-SW-P-4 are running which locks out Service Water pumps 1/2-SW-P-1A from auto starting. 1/2-SW-P-1B will auto start but because of the elevation difference between the Service Water reservoir and the discharge canal, the discharge pressure of service water pump 1/2-SW-P-1B will be lower and flow higher than expected. Therefore, when Service Water is lined up Lake-to-Lake and a Safety Injection/Containment Depressurization Accident (CDA) occurs, there is a good chance that the Service Water pumps located at the reservoir will run out. Service Water pump 1/2-SW-P-4 will continue to run, however single failure criteria requires the assumption that one will fail. In addition, no flow balance testing of the Service Water system using 1/2-SW-P-4 in Lake-to-Lake mode has been performed so credit should not be taken for these pumps. Adequate recirculation spray service water flow may not be available to return containment to subatmospheric conditions within one hour of a design basis accident and therefore Lake-to-Lake operation of the Service Water system should not be used for normal operations.

Note that 1-OP-49.1 (Service Water Operation) restricts Lake-to-Lake operation of the Service Water system to emergency situations or when both units are in Mode 5 or 6. During Mode 5 or 6, a Loss of Coolant Accident (LOCA) or Feed/Steam Line break requiring CDA are not credible accidents and Lake-to-Lake Service Water operation would therefore not increase the consequences of an accident or malfunction. In an emergency situation (such as loss of all normal service water pumps), both units would be making preparations for shutdown and the time spent aligned Lake -to-Lake above Mode 5 would be short. The chances that an accident requiring CDA initiation during this situation is negligible. Therefore, the increase in the consequences of an accident requiring CDA or of a malfunction is negligible. In no case would operation of the service water system in Lake-to-Lake mode increase the probability of an accident requiring CDA.

SAFETY EVALUATION NUMBER 91-SE-OT-003

DESCRIPTION

Section 15.2.4.2.3 will be revised to indicate that the reactor coolant system can be filled via the charging system as well as the refueling water storage tank.

Resolution of Deviation Report 90-1442.

SAFETY ANALYSIS SUMMARY

Section 15.2.4 of the UFSAR discusses uncontrolled boron dilution accidents. As part of this general section, 15.2.4.2.3 describes how the reactor coolant system is filled prior to startup, and what design parameters of our power plant prevent uncontrolled boron dilution during this evolution.

As currently written, Section 15.2.4.2.3 states that the reactor coolant system is filled with borated water from the refueling water storage tank. Deviation Report 90-1442 was written because we normally use the charging pumps to fill the reactor coolant system with water that has been borated in the boric acid blender.

Filling the reactor coolant system is controlled by 1/2-OP-5.1. This procedure gives instructions for filling from the refueling water storage tank or by use of the charging system. When filling via the charging system, this procedure prevents uncontrolled boron dilution by the following requirements:

1. The fill water must have a boron concentration equal to or greater than the water in the reactor coolant system.

2. The procedure limits total charging system flow to a maximum of 150 gpm.

Filling the reactor coolant system in accordance with 1/2-OP-5.1 does not violate the limitations of UFSAR Section 15.2.4.2.3, because the maximum dilution rate is still less than 300 gpm. Therefore, there is no increase in the probability of an uncontrolled boron dilution accident. Consequences of an uncontrolled boron dilution accident would not be increased by this method of filling, as the dilution could be stopped by increasing the concentration of boron or stopping the charging pump.

During the filling evolution, the charging system and the reactor coolant system will be operated within their design bases. This should prevent the malfunction of any safety related equipment or an accident of a type not previously evaluated.

Section 15.2.4.2.3 of the UFSAR should be revised to indicate that the reactor coolant system can be filled via the charging system as well as the refueling water storage tank.

SAFETY EVALUATION NUMBER 91-SE-OT-004

DESCRIPTION

The general (High Level) outage plan/schedule for the upcoming Unit 1 refueling outage is being evaluated. This is not a change, test, or experiment, but the outage involves numerous tests and abnormal plant configurations, and therefore warrants a safety evaluation.

The purpose of the outage plan/schedule is to provide a guideline for outage activities in order to ensure that all required testing and maintenance evolutions are accomplished. The plan must accomplish this while maintaining essential components, power supplies, and system configurations operable to the extent required by Technical Specifications, other license commitments, and operating limitations in general.

SAFETY ANALYSIS SUMMARY

The outage schedule is only a plan for the various activities which must occur in order to refuel the reactor, test various components, and to properly maintain the equipment so that it can perform design functions. These activities are planned with the intent of eliminating situations that could increase the probability of accidents or equipment malfunctions. This is accomplished by arranging the timing of potentially conflicting activities so that operability and availability of essential equipment is maintained at all times. Maintenance activities are scheduled such that boration flow paths and high volume make-up are available whenever fuel is in the reactor vessel. Complex, multi-departmental evolutions are routinely preceded by a pre-job briefing where all aspects of expected and potentially unexpected plant responses are discussed by key personnel. All equipment required by Technical Specifications while shutdown will be maintained operable. The shutdown and start-up sequence will be carried out using approved procedures and nuclear design parameters will be verified prior to power ascension via performance of start-up physics testing.

SAFETY EVALUATION NUMBER 91-SE-OT-005

DESCRIPTION

Reload of North Anna Unit 1 for Cycle 9 operation.

SAFETY ANALYSIS SUMMARY

Technical Report NE-822 Rev.0 presents a discussion of the analyses and evaluations supporting the conclusion that the North Anna 1 Cycle 9 reload core can be safely operated to its cycle burnup limit and that an unreviewed safety question does not exist.

SAFETY EVALUATION NUMBER 91-SE-OT-006

DESCRIPTION

To clarify the definitions of a cooling system as described in sec. 9.4.1.1 of the UFSAR. "C" chiller is not a "swing" chiller as it can only be powered from the H train. It can be mechanically aligned to the I train air conditioner as described in sec. 9.4.1.1.
To clarify the Service Water line-up to the air conditioning systems.

The purpose for this change:

Unclear explanation of air conditioning systems.
Provide the current equipment configuration of the cooling systems.
Clarify the Service Water line-ups.
Clarify the operating hours of the cooling system equipment.

SAFETY ANALYSIS SUMMARY

One 100% capacity cooling system which supplies the common control rooms and emergency switchgear rooms in order to meet the single failure criterion is installed for each reactor unit. The cooling systems cannot be cross connected between the two reactor units. Each cooling system consists of two independent 100% redundant air conditioning trains, one powered from the H train and the other powered from the J train. An air conditioning train consists of a control room air handling unit (AHU), an emergency switchgear room AHU, chilled-water piping and a water chiller. An additional water chiller (HV-E-4C) for each reactor unit is provided to prevent compressor failure from shutting down the H train air conditioning system for any appreciable time. This chiller has the capability of being mechanically aligned to provide chilled water to either air conditioning system of its respective reactor unit. Because the HV-E-4C chiller cannot be powered from two emergency power sources (H & J), it is not truly a "swing" unit. Each reactor unit has two (HV-E-4A & HV-E-4C) chillers which receive power from the H train and an HV-E-4B chiller which receives power from the J train. The air conditioning arrangement is such that no action, either automatic or manual, is required during an emergency, as the normal mode will continue. However manual action is always required during normal and emergency plant conditions for the respective reactor unit whenever any operating air conditioning system fails. Therefore, it is permissible to mechanically align the HV-E-4C chiller to provide chilled water to either air conditioning system of its respective reactor unit during normal and emergency plant conditions; however, if HV-E-4C is aligned to 'J' train AHU's, the 'B' air conditioning train will only be considered operable if HV-E-4B is available to be placed in service.

SAFETY EVALUATION NUMBER 91-SE-OT-007 and
91-SE-OT-007 (Revised)

DESCRIPTION

To add requirements to TS 3.7.4.1 to throttle flow to Component Cooling Heat Exchangers when less than 4 Service Water Pumps are operable to Action d of 3.7.4.1 that the third Service Water pump does not require auto-start capability and change 1/2 auxiliary service water pumps to 2/2 and add LCO 3/4.7.4.2 for OPERABILITY of Service Water System in Modes 5 and 6.

This will ensure that greater than or equal to design flows are achieved to the recirculation spray heat exchangers during a design basis accident and clarify the requirements of the 3rd service water pump and ensure that a complete backup system is available in case of a passive failure and that an adequate heat sink is maintained for the residual heat removal system.

SAFETY ANALYSIS SUMMARY

The Technical Specification change will enhance the availability of the service water pumps and ensure adequate flow to the recirculation spray heat exchangers. Operation of the service water system is not affected. Throttling service water to the component cooling heat exchangers has no significant affect on the component cooling system operation. Clarifying Action d, of TS 3.7.4.1, to not require auto-start capability for the 3rd service water pump, will protect the service water pumps from low flow conditions and still provide a backup in case of an active failure. Requiring 2/2 Aux service water pumps will ensure that a backup system is available in case of a passive failure. New Action e of TS 3.7.4.1 will place the units in a safe condition if heat sink is not available during Modes 1-4. The new LCO (3/4.7.4.2) ensures that an adequate heat sink will remain available when both units are in Modes 5 or 6.

Based upon the above statements, the design basis of the system involved will be assured. There is not an unreviewed safety question.

SAFETY EVALUATION NUMBER 91-SE-OT-010

DESCRIPTION

This Special Test was performed to determine the torque required (using MOVATS Equipment) to operate the Recirculation Spray Heat Exchanger (RSHX) inlet and outlet Motor Operated Valves (MOVs) by simulating design-basis conditions. This test was conducted as part of the GL 89-10 actions. This test applied to the following valves:

1. 1-SW-MOV-103A, SW Supply to "A" RSHX Isol Valve.
2. 1-SW-MOV-103B, SW Supply to "B" RSHX Isol Valve.
3. 1-SW-MOV-103C, SW Supply to "C" RSHX Isol Valve.
4. 1-SW-MOV-103D, SW Supply to "D" RSHX Isol Valve.
5. 1-SW-MOV-104A, SW Supply to "A" RSHX Isol. Valve.
6. 1-SW-MOV-104B,SW Supply to "B" RSHX Isol Valve.
7. 1-SW-MOV-104C, S\ Supply to "C" RSHX Isol Valve.
8. 1-SW-MOV-104D, SW Supply to "D" RSHX Isol Valve.

SAFETY ANALYSIS SUMMARY

This Special Test measured system performance, but did not change the system in any permanent way. Also, as specified by Safety Evaluation 91-SE-ST-010, the test could have been performed in Mode 5 or 6 when RSHX operability is not required. Adequate SW flow to opposite Unit during CDA was insured by requiring immediate isolation of Unit 1's RSHXs in the event of a Unit 2 CDA. Following the test, the LMCs were closed and capped.

Although performed for only Unit 1 at this time, this safety analysis is applicable to a nearly identical Unit 2 procedure.

SAFETY EVALUATION NUMBER 91-SE-OT-012

DESCRIPTION

Paragraphs in UFSAR Sections 4.2.1.1.2, 4.2.1.2.2, 4.2.1.3.2, 4.2.1.4.2, 4.2.1.4.3, 4.2.3.1, 4.2.3.2, 4.2.3.2.1, 4.2.3.3.1, and 4.2.3.4.1 were slightly modified. Paragraphs were deleted from Section 4.2.1.3.1.

These sections were modified or deleted to reflect current Westinghouse design and manufacturing methods.

SAFETY ANALYSIS SUMMARY

These sections were modified to make them consistent with current assembly design and manufacturing and design methods. These changes involve only format improvements and text clarification.

The changes to the text do not involve plant changes, tests or experiments. Therefore the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety is not increased.

The changes do not involve the identification of technical information other than that already considered in the safety analyses. Therefore the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR is not created.

The changes do not introduce any information relative to the performance or integrity of any fission product barrier which was not incorporated in the safety analyses. The conclusions of the approved safety analyses are unchanged. Therefore the margin of safety as defined in the basis for any technical specification is not reduced.

SAFETY EVALUATION NUMBER 91-SE-OT-013

DESCRIPTION

Updated Section 2.3 of NAPS UFSAR with more recent information.

More recent information is available.

SAFETY ANALYSIS SUMMARY

New meteorological information has become available (updated) incorporating recent years of climatological data gathering. An unreviewed safety question does not exist as no changes to the facility, no changes to the equipment, and no changes to testing resulted from including news meteorological data.

SAFETY EVALUATION NUMBER 91-SE-OT-014

DESCRIPTION

Correct a typographical error in the UFSAR.

These change(s) were identified by the UFSAR Verification Project in accordance with NDCM-3.18 and Electrical Engineering Implementing Procedure EE-021.

SAFETY ANALYSIS SUMMARY

The following changes were identified by the UFSAR Verification Project in accordance with NDCM-3.18 of Electrical Engineering Implementation Procedure EE-021.

North Anna Power Station (NAPS) UFSAR Section 9.2.1 in listing the service water system indicators and alarms which are monitored from the control room lists: "Low flow alarms on the recirculation spray heat exchanger radiation monitoring pump discharge, the charging pump, gearbox, and seal water lines, and the discharge to the service water spray system." Source documents (listed in References) indicate that the gearbox and seal water lines of the charging pump are the lines being monitored for low-flow. This UFSAR change will correct a typographical error by stating: "Low-flows alarms on the recirculation spray heat exchanger radiation monitoring pump discharge, the charging pump (gearbox and seal water lines), and the discharge to the service water spray system."

This change to the UFSAR is to correct a typographical error. Correcting this error will also add clarity to the statement. There will be no physical changes to the station.

SAFETY EVALUATION NUMBER 91-SE-OT-015

DESCRIPTION

Correct the discussion of control room displays in section 5.6.2 of the UFSAR.

These change(s) were identified by the UFSAR Verification Proj..ct in accordance with NDCM-3.18 and Electrical Engineering Implementing Procedure EE-021.

SAFETY ANALYSIS SUMMARY

The following cnanges were identified by the UFSAR Verification Project in accordance with NDCM-3.18 of Electrical Engineering Implementation Procedure EE-021.

North Anna Power Station (NAPS) UFSAR Section 5.6.2, in discussing the Resistance Temperature Detector Bypass Manifolds states: "The Tavg for each loop is indicated on the main control board." Source documents (listed in References) indicate that delta T is also indicated on the main control board. This UFSAR change will indicate this by stating: "The Tavg and delta T for each loop is indicated on the main control board."

North Anna Power Station (NAPS) UFSAR Section 5.4.2.3 states: "An additional signal is transmitted...in the control room. This separation meets 10CFR50 Appendix R Section III.G.2.d." Section 5.6.2.4 states: "A signal is also transmitted to the auxiliary monitoring panel...in the control room. This separation meets 10CFR50 Appendix R Section III.G.2.d." The attached source documents (listed in References) show that separation is actually accomplished by implementing the requirements of 10CFR50 Appendix R Section III.G.2.f. Since any of the means of separation identified in 10CFR50 Appendix R Section III.G.2 are acceptable, this UFSAR change will revise both of the impacted sections as follows: "This separation meets 10CFR50 Appendix R Section III.G.2."

These editorial changes to the UFSAR will , rovide consistency between the UFSAR and design documents. No physical changes to the station are being reported.

SAFETY EVALUATION NUMBER 91-SE-OT-016

DESCRIPTION

This UFSAR Change Request discusses full flow testing of the inside recirculation spray pumps (IRSPs) which is performed during refueling and "dry bump" testing which is performed quarterly.

Support/reflect proposed License Amendment (Serial NO. 90-596R2) to delete License Condition 2.C(15)(c) on Unit 2.

SAFETY ANALYSIS SUMMARY

Long term mechanical reliability of the Inside Recirculation Spray Pumps (IRSPs) is the major issued considered. Deleting the requirement (License Condition 2.C(15)(c)) to remove and inspect the Unit 2 ISRPs and notifying the NRC of our current practice of full flow testing the Unit 1 and 2. ISRPs on a refueling basis sufficiently provides verification of the ISRPs ability to perform their intended function. Continued reliability testing of the ISRP sufficiently justifies deleting the license condition.

SAFETY EVALUATION NUMBER 91-SE-OT-017

DESCRIPTION

Evaluation of Steam Generator Row 1 Explosive Tube Plugs and Tubes.

Potential for explosive plug failure or Row 1 tube failure.

SAFETY ANALYSIS SUMMARY

During the 1987 refueling outage for North Anna Unit 1, visual inspections of the primary side of the steam generator tubesheets were performed. A few explosively installed Row 1 plugs were identified as potential leakers. Several of these explosive plugs appeared to be dripping water, several other plugs in each steam generator appeared to be either damp or wet. All suspected leaking explosive plugs are located in the hot leg side of the affected tubes. All row 1 tubes in the North Anna Unit 1 steam generators which were plugged using explosive plugs were preventively plugged prior to indication of tube degradation.

A safety evaluation was previously performed for operation for one cycle with any potentially leaking plugs as found. This evaluation has been written to evaluate the effect on safety of operation of the steam generators with the previously identified or newly identified leaking explosive plugs for an indefinite number of fuel cycles. The evaluation is based on continued periodic observation of the row 1 tubes. Additionally this evaluation considers the potential for plug top release.

To date, no primary to secondary leakage has been attributed to any potential leaking row 1 explosive plugs. Since the tubes were plugged after a limited period of operation, through wall corrosion of the tubes would not be expected, therefore the focus of the previous evaluation was consideration of damage that a leaking explosive plug could cause to a plugged tube. The following is a summary of the 1987 evaluation, the 1989 flow slot inspection, and plug top release consideration.

1987 Evaluation Summary

Analysis of the potential for contained water in a plugged tube indicates that more than one cycle of pressurization and heatup is necessary to obtain the required net volume expansion of the tube until a break occurs. The number of cycles required to open a break in a tube depends on several factors including: the initial conditions of the trapped mass, water out-leakage if any during plant heatup, the dissolution of air in the water, etc. However, the actual number of cycles required is unknown.

Flow slot photographs for all three SG's were examined during the 1987 refueling outage to determine if any of the tubes exhibited signs of bulging. If the hydraulic cycling mechanism described above were occurring, the pictures would be expected to reveal some degree of relative bulging. A photographic inspection of the tubes and flow slots during each refueling outage is expected to provide sufficient monitoring of the potential for tube bulging. The validation that no tubes are bulged at the start of a fuel cycle will assure that the potential for a tube rupture has not been increased.

1989 Flow Slot Inspection

An evaluation of the flow shot photographs taken during the 1989 refueling outage showed none of the tubes visible in Row 1 exhibited any indications of volumetric tube expansion. The pattern formed by the intersections of the tubes with the tube support plate is determined to be linear except for some curvature attributable to lens parallax; the same curvature is discernible along the edges of the flow slots. The visibility of tubes away from the center of the photographs is diminished uniformly with distance (as would be expected). The distance from the flow slots to tubes adjacent to the slots appears to be uniform.

Plug to Release

The phenomenon of plug top release has also been considered as a possible failure mechanism. Unlike a mechanical plug configuration, explosive plugs have an installed configuration which is not expected to temporarily or intermittently seal primary to secondary leakage from a through wall crack above the primary contact area between the plug and the tube. The residual stress as a result of the explosive expansion process is not conducive to a uniform through wall circumferential oriented PWSCC process. Tube portions containing tubesheet expansion transitions also formed by an explosive process were removed from North Anna Unit 2 and destructively examined. Cracking which had occurred in these transitions was non-uniform and through-wall. Additionally, explosive tube pugs removed from another unit and examined did not show uniform circumferentially oriented cracking. These investigations support the conclusion that explosive plugs will not degrade in a manner that could result in a rapid plug top release similar to that postulated for mechanical tube plugs. Given the expected corrosion mechanism for a leaking explosive plug, any leak as a result of a hypothetical tube rupture in an explosively plugged tube would be less than that of the previously analyzed postulated tube rupture accident.

Conclusion

Since the secondary side photographic inspection of the visible row 1 tubes in all 3 steam generators did not indicate any volumetric tube expansion, the hydraulic cycling mechanism described above would not be expected to result in tube rupture of a previously plugged tube during the current fuel cycle of plant operation based on the previous safety evaluation. Verification that tube bulging has not occurred using visual inspection of the row 1 tubes during subsequent refueling outages will support operation of the steam generator during the following fuel cycle.

Based on the information outlined above and the previous evaluation, operation of the North Anna Unit 1 steam generators with potentially leaking explosive plugs during a fuel cycle following a visual tube inspection with no indication of tube bulging is acceptable. Based on the information evaluated, continued operation the steam generator North Anna Unit 1 will not result in an unreviewed safety question as defined in the criteria of 10CFR 50.59 (a) (2).

SAFETY EVALUATION NUMBER 91-SE-OT-018

DESCRIPTION

Action plan for corrective actions taken to mitigate plug top release.

A plug top release problem has been identified with Westinghouse designed mechanical plugs.

SAFETY ANALYSIS SUMMARY

The potential for rapid release of the top of some Westinghouse designed mechanical tube plugs has been identified. While this is considered a low probability event, under certain conditions the release of the plug top can occur with sufficient energy to puncture the tube in which it is installed. As one part of the program to address this phenomenon, an action plan has been developed to minimize the potential for plug top release from mechanical tube plugs installed in steam generators. The action plan is discussed in detail in the safety evaluation. This safety evaluation is written to address the program implemented at North Anna Unit 1 to determine the plugs which may be returned to service as is or modified by installation of a plug in plug (PIP) and the plugs to be removed and replaced. Additionally, the integrity of welded plugs is assessed.

Operation of the North Anna Unit 1 steam generators subsequent to the completion of the action plan has been evaluated using the criteria of 10 CFR 50.59 for an unreviewed safety question. The action plan included installation of PIPS in or removal and replacement of mechanical plugs judged to have the potential for cracking during the next fuel cycle. However, for part of the evaluation of the criteria of 10 CFR 50.59 for an unreviewed safety question, instantaneous plug top release is arbitrarily assumed, although not expected.

The maximum flow rate through a leaking plug in a tube with a postulated tube rupture is limited by the expander of the plug and would be less than the leak rate assumed for the previously completed accident analysis for tube rupture. The possibility of an analyzed accident, in particular a steam generator tube rupture, has not been increased since the maximum flow rate through the expander of a failed plug is less than the RCS makeup capability and does not represent a flow rate equal to a tube rupture. Thus a plug top release would not represent an accident.

Any hypothetical failure of a plug due to PWSCC, including simultaneous plug top release in several plugs, would be bounded by the analysis of a single tube rupture and the possibility of a different accident has not been created. Based on analysis and testing, the released tube plug top is not calculated to contain enough energy to perforate the tube walls of both the inactive tube and an adjacent active tube or tubes.

The action plan outlined in the evaluation is expected to minimize the potential for a plug top release event which could result in a tube perforation for the next fuel cycle. The tube plugs remaining in service either operate at a temperature, are of a material condition, or a design configuration not expected to experience cracking in the next cycle, have had a PIP installed, or are in tubes plugged with a sentinel plug in the cold leg end and therefore expected to be filled with water.

Based on the information outlined in the safety evaluation, operation subsequent to the implementation of the mechanical plug action plan for North Anna Unit 1 will not result in an unreviewed safety question as defined in the criteria of 10 CFR 50.59.

SAFETY EVALUATION NUMBER 91-SE-OT-019

DESCRIPTION

Evaluation of the use of "Plug-In-Plug" methodology to prevent steam generator mechanical tube plug failure.

Method used to repair steam generator mechanical tube plugs.

SAFETY ANALYSIS SUMMARY

This evaluation is written to address the effect on the safe operation of the plant of the installation of plug in plugs in selected steam generator mechanical tube plugs at North Anna Station Unit 1.

The potential for the rapid release of the top of some Westinghouse designed mechanical tube plugs has been identified. While this is considered a low probability event, under certain conditions the release of the plug top can occur with sufficient energy to puncture the tube in which it is installed. As one part of the program instituted to address this phenomenon, a device has been developed to minimize the energy which can be imparted to the plug top in the event of a sudden plug top release. This device restricts the rate at which reactor coolant can enter the interior of the tube by plugging the opening to the plug. This device is referred to as the plug in plug (PIP). The PIP is designed to be installed in plugs as determined by an evaluation of the corrosion propensity of plugs and in lieu of other actions such as removal and replacement of the plug. It is expected that the PIP may remain in place in the plug for the remaining operational life of the steam generator.

The evaluation of the corrosion propensity of tube plugs is based on considerations such as material conditions and operating temperatures. The evaluation of tube plug corrosion propensity and the determination of which plugs are to have PIP's installed are addressed elsewhere, separately and are not considered in this safety evaluation.

The PIP has a threaded shaft which is screwed into the internal threads of the expander of a mechanical plug. The PIP is screwed into the expander until a flange on the bottom of the PIP contacts the bottom of the tube plug shell and an pre-established torque limit is reached. The PIP does not contact the top of the tube plug and does not engage the threads in the bottom of the plug shell. The PIP is designed to be removable without damage to the pressure boundary integrity of the PIP and the bottom of the tube plug shell.

The function and integrity of the tube plug is not adversely affected by the installation of the PIP. Since the tube plug represents the pressure boundary, installation of the PIP will not increase the potential of a tube rupture or tube leak. The PIP is installed in a previously plugged tube and will not change the hydraulic or heat transfer characteristics of the steam generator for design transients or postulated accident analyses. The installation of a PIP will not increase the possibility or consequences of a previously analyzed accident. The effect of any failure as a result of the installation of a PIP, including hypothetical failure of a tube plug, would be bounded by the analysis for a steam generator tube rupture and the possibility of a previously unanalyzed accident has not been created. The margin to safety for the primary to secondary pressure boundary as defined in the basis of the Technical Specification is provided in part by the Technical Specification requirements for tube eddy current inspection and plugging limit and the provisions of the ASME Code used in the design of the tube plug and the PIP including inherent safety factors. The margin of safety is not reduced.

Based on the information outlined above, installation of the PIP into selected mechanical plugs in the steam generator tubes at the North Anna Power Station Unit 1 will not result in an unreviewed safety question as defined in the criteria of 10 CFR 50.59.

SAFETY EVALUATION NUMBER 91-SE-OT-020

DESCRIPTION

Operation with Steam Generator Mechanical Tube Plug Remnants remaining in the tubes.

The plug top remnants can not be removed with remote equipment and ALARA consequences are great.

SAFETY ANALYSIS SUMMARY

This evaluation assesses the potential safety impact of operation with steam generator mechanical plug remnants (i.e., a portion of the top of a tube plug), including in some cases portions of the sealing lands, remaining in steam generator tubes within the tubesheet region of steam generators. This evaluation is valid for any steam generator in which mechanical tube plugs are used. The evaluation is valid for any number of plugged tubes with plug remnants.

The subject condition can arise as a result of the removal of mechanical plugs from steam generator tubes. A portion of the plug above the expander may not be removed by the plug removal process and to be removed must be manually pulled from the tube following the machining operation. Since removing the plug remnant is typically difficult and may account for unnecessary occupational radiation exposure (ORE), the option is provided to leave the steam generator tube plug remnant in the tubesheet and reinstall plugs fabricated of Alloy 690 material. Hence, the plug remnants will remain in the tubes within the tubesheet region. The newly installed plug will perform the function of removing the tube from service and acting as the primary-to-secondary pressure boundary.

This evaluation demonstrates that operation of the steam generators with plugged tubes containing plug remnants will not have an adverse effect on the pressure boundary integrity of the steam generator and does not represent an unreviewed safety question per the criteria of 10 CFR 50.59.

The steam generator tube plug remnant is captured in an inactive steam generator tube by the installation of a new plug. The new plug represents the pressure boundary between the reactor coolant and the secondary side. There are no forces on the plug remnant as a result of normal operation or postulated accident conditions which would tend to move it from within the tubesheet region of the tube where it remains after the plug removal and installation operations. There are no fluid or dynamic structural forces on a remnant in a tube removed from service which could result in wear of the remnant on the tube or replacement plug. The presence of a plug remnant does not impact any structural considerations relative to the integrity of a tube.

Operation of the steam generators with tube plug remnants, is not expected to have an adverse impact on material of the tubes or tube plugs. The plug remnant material as well as the newly installed tube plugs are compatible with Alloy 600 and Alloy 690 steam generator tubing and tube plugs. The presence of a plug remnant in a tube removed from service does not provide a concentrating mechanism to induce corrosion of the tube or new plug.

A plug remnant inside the steam generator tubesheet region may have a potential adverse effect on a postulated tube plug top release. However, the tube plugs to be installed in tubes containing plug remnants are of Alloy 690 material. Cracking of the plugs is a necessary precursor to plug top release and cracking is not expected in thermally treated Alloy 690 tube plugs based on extensive corrosion testing.

SAFETY EVALUATION NUMBER 91-SE-OT-021

DESCRIPTION

Test data now exists that justifies removal of a majority of sentinel plugs installed in Unit 1 Steam Generators during the Fall 1987 Outage. Tubes that pass inspection will be returned to service. Those that do not pass inspection will be re-plugged.

Tubes recovered as a result of this analysis will offset the tubes required to be plugged in this Outage. This analysis still addresses requirements of NRC Bulletin 89-01, "Failure of Westinghouse SG Tube Mechanical Plugs."

SAFETY ANALYSIS SUMMARY

Subsequent to the North Anna Unit 1 tube rupture on July 15, 1987, a tube fatigue analysis was performed for the North Anna 1 plant, and several modifications were implemented. Downcomer flow resistance plates were installed in all steam generators, resulting in a nominal reduction of tube stability ratios. Sentinel plugs, which provide a leak rate less than the Technical Specification limit of 500 gpd and well below that of a ruptured tube, were installed. The plug locations were determined based upon criteria developed at the time

Testing since this evaluation has determined that single-sided support of the tubes is sufficient to limit fluidelastic excitation of the tubes. Non-uniform AVB insertion configurations have been tested to define AVB positions which, dependent upon local flow conditions, produce excessive tube vibration. The analysis document provides the justification for the removal of a majority of the sentinel plugs installed in North Anna Unit 1 during the July, 1987, outage. The justification is developed from a detailed AVB insertion mapping, updated thermal/hydraulic analysis, and vibration analysis. The fatigue analysis considered the effects of prior operating history on tube fatigue and of a postulated T_{hot} reduction. The report concludes that a few tubes should remain sentinelly plugged. The remainder of the sentinel plugs installed in response to the tube fatigue issue, with the exception of those "boxing" the plugged R9C51 tube, may be removed.

SAFETY EVALUATION NUMBER 91-SE-OT-022

DESCRIPTION

Evaluation of loose objects identified in secondary side of steam generators.

Loose parts have been identified in the steam generator secondary side.

SAFETY ANALYSIS SUMMARY

During a Foreign Object Search And Retrieval (FOSAR) conducted during the Cycle 8 refueling outage, several loose objects were found in he secondary side of the North Anna Unit 1 steam generators A and B. This safety evaluation addresses the operation of North Anna Unit 1 with the identified loose objects remaining the secondary side of the steam generators and demonstrates that they will not have an adverse effect on the primary pressure boundary integrity of these steam generators and does not represent an unreviewed safety question per the criteria of 10CFR 50.59.

BACKGROUND

The following foreign objects have been identified in the secondary side of the North Anna Unit 1 steam generators during a FOSAR conducted during the Cycle 8 refueling outage:

S/G	Object	Dimensions
A	Cylindrical Metallic Object	0.25"x1 1/2 inches in length
A	Flexitallic Gasket Strip	0.15" width x1 1/2 inches in length
B	Tube Segment (From Tube Pull)	0.125 inch in length

The cylindrical metallic object in Steam Generator A is located on the tubesheet near tube R40 C24 in the nozzle side hot leg of the tube bundle. The object could not be retrieved due to its position relative to the steam generator hand hole. It is not certain whether the object will remain fixed in place during subsequent plant operation. The object appears to be highly corroded and it appears to have threads The flexitallic gasket strip, also in Steam Generator A, was originally located on the tubesheet near tubes R33-34 C16 in the cold leg of the tube bundle. The object was moved almost to the hand hole then was "lost". The piece could not be relocated.

During the tube pull efforts in steam generator B, two attempts were made to perform the cut on the inner diameter of the tube. The drive system for the cutters broke while making the first cut in tube R11 C14 (hot leg). The second cut in the same location resulted in a small tube segment as the cutter was not positioned in the exact location as the first cut. This segment was observed during a Welsh Allyn inspection of the cut area. The tube segment is currently located on the first tube support plate; it is expected to either remain in place or migrate further into the bundle or potentially to the tubesheet during subsequent plant operation.

This evaluation demonstrates that operation with the identified foreign objects remaining in the secondary side steam generators A and B will not have adverse effect on the primary pressure boundary integrity of these steam generators and does not represent an unreviewed safety question.

No significant tube wear would be expected at this location as there is virtually no tube vibration at the top of the tubesheet elevation since the tube is expanded into the top of the tubesheet. No eddy current signal was identified in Tube R40 C24 at the top of the tubesheet region.

To address the potential for the objects in Steam Generator A and B to migrate during future plant operation, estimated wear calculations were conducted for each of these objects assuming that the objects may, in time, remain in contact with a tube in a worst case orientation and that sliding/impact wear occurs. The wear calculations performed take into account the flow velocities, fluid densities, and resultant drag forces for the North Anna Unit 1 Model 51 steam generators. The bounding wear calculation approximates the time expected for the impact sliding wear mechanism to reduce a tube form 0% wall loss to the technical specification plugging limit of 40% allowable wall loss. Eddy current inspection showed no evidence of wall thinning in the peripheral tubes in either Steam Generator A or B. The results of this evaluation indicate that tube wear times to allowable wall loss envelop Cycle 9 operation.

INSTRMENTATION AND CONTROL EVALUATION

The foreign parts are on (or near) the top of the tubesheet, therefore, it is highly unlikely that any of the foreign parts would be lifted by fluid flow from the top of the tube sheet. Even if some foreign parts would be lifted from the top of the tube sheet, it is highly improbable that these foreign parts would flow through the tube bundle, through the flow distribution baffle, through the tube support plates, and through the anti-vibration bars. In addition, the foreign parts would have to be oriented such that it would flow in the gap between the steam generator tubes and the support plates. Even if these loose parts would flow through the tube bundle, through the flow distribution baffle, through the tube support plates, and through the anti-vibration bars it is highly unlikely that the foreign parts would be lifted up through the primary moisture separators. In the highly unlikely event that the subject foreign parts could migrate upward through the steam generator bundle, through the flow distribution baffle, through the tube support plates, through the anti-vibration bars, and then through the primary separators then the loose part could migrate either upward into the steam region or downward into the downcomer region between the generator shell and wrapper.

If loose parts could flow through the tortuous paths indicated above, then a potential safety concern exists concerning loose parts in the steam generator secondary side in that the loose parts could migrate upward through the steam generator bundle and then either upward into the steam region or downward into the downcomer region between the generator shell and wrapper.

Safety-related instrument sensing lines are located in both of these areas. At North Anna Unit 1, there are safety-related level and delta P level sensing instrument taps on the shell of the steam generator. These taps are flush with the inside diameter of the steam generator shell. The sensing lines associated with level instrument taps have an inside diameter (I.D.) of approximately 1/4 inch.

Loose parts flow through the secondary moisture separators and into the steamline is not credible because of the additional tortuous path through the chevron type dryer vanes. In addition, a loose part would have to be oriented such that it would flow through a perforated plate located upstream of the secondary moisture separator. Therefore, the safety-related pressure taps in the steamline will not be impacted by any loose parts.

The operation at North Anna Unit 1 with the foreign objects present in the secondary side of the Steam Generators A and B has been evaluated using the guidance of NSAC-125 and does not involve an unreviewed safety question per the criteria of 10 CFR 50.59.

SAFETY EVALUATION NUMBER 91-SE-OT-023

DESCRIPTION

Demonstrate that the previous Safety Evaluation for cable stabilizers for S/G tubes exhibiting circumferential degradation is applicable to the implementation of the S/G tube cable stabilization program for the North Anna Unit 1 Feb. 1991 outage.

The North Anna Unit 1 cable stabilization program during the Feb., 1991 outage is slightly different than the previous Safety Evaluations. However, the current methodology is sound and remains bounded by the previous Safety Evaluations, this safety evaluation documents the slight changes in the program and the overall integrity of the S/G identification and tube stabilization program.

SAFETY ANALYSIS SUMMARY

Virginia Power requested Westinghouse to provide an evaluation assessing the potential safety impact of operating North Anna Unit 1 with thirteen (13) tubes plugged and stabilized due to eddy current inspection results indicating circumferentially oriented cracking near the top of the tubesheet.

In addition, Westinghouse has assessed the potential safety impact of operating the North Anna Unit 1 steam generators without stabilizing plugged tubes which had exhibited indications of circumferential cracking.

The crack indications are located within the tube explosive-expansion transition zone in the hot leg tube end at the top of the tubesheet. Typically, these indications are not readily detectable via the standard bobbin coil probe. The indications reported above were detected by an 8 x 1 probe and characterized by a rotating pancake coil (RPC) probe.

Steam generator secondary side loose objects are not judged to be the cause of the tube degradation. Confidence in eddy current signal characteristics and tube locations (essentially all non-peripheral) support the absence of a loose object at the tubesheet elevation as a localized wear mechanism.

The evaluation, completed in accordance with 10 CFR 50.59 criteria, is written to assess the potential safety impact of operation of North Anna Unit 1 with certain stabilized, plugged, and removed from service. These tubes had exhibited unacceptable indications of circumferential cracking.

In addition, this evaluation assesses the potential safety impact of operating the North Anna Unit 1 steam generators without stabilizing various plugged tubes which also had exhibited indications of circumferential cracking.

Furthermore, structural, i.e., thermal-mechanical and flow induced vibration considerations are addressed within this evaluation as is the potential for interaction of the afore referenced tubes with active tubes.

This evaluation also considered continued operation of the North Anna Unit 1 steam generators with the twenty nine (29) plugged tubes exhibiting moderate indications of circumferential cracking. This assessment addressed corrosion considerations and tube structural considerations. In addition, the potential interaction of inactive tubes with adjacent active neighboring tubes was evaluated and judged to be acceptable. It is concluded that operation of the North Anna Unit 1 steam generators after stabilizing the 13 subject degraded, plugged tubes will not result in a previously unanalyzed accident or increase the probability of an analyzed accident. Concomitantly, it is concluded that continued operation of the steam generators without stabilizing the 29 tubes is not expected to deleteriously impact the safe operation of the North Anna Unit 1 steam generators. Additionally, from the perspective of the probability and consequences of a steam generator tube rupture, the above evaluation supports the conclusions that steam generator tube integrity margins as provided in the Technical Specification bases are not reduced. Therefore, operation of North Anna Unit 1 through Cycle 18 with the 13 tubes plugged and stabilized and the 29 tubes plugged but not stabilized does not represent an unreviewed safety question per 10 CFR 50.59 (a) (2) criteria.

SAFETY EVALUATION NUMBER 91-SE-OT-025

DESCRIPTION

Operation at high power (> 30%) using the bypass feedwater regulating valves. This may be accomplished by any combination of manual or auto control of the bypass and main feed regulating valves.

Occasionally there are "rough" operating positions on the main feed. reg. valves. By utilizing the bypass feedwater valves, smoother feedwater control is maintained and thus better S/G level control is attained.

SAFETY ANALYSIS SUMMARY

The concern for operating at any power level with any combination of Main Feed Regulation Valve (MFRV) or Bypass Feed Regulation Valve, is an excessive feedwater addition due to a feedwater control problem. The feedwater control problem may be due to equipment failure or operator error.

UFSAR Chapter 15.2.10 evaluates this transient. The UFSAR assumes the following:

1) Initial S/G water level at the low-low limit
2) All FW injection into the S/G is assumed to be 70°F
3) All FW is injected at 200% of the normal full power feed rate
4) The FW is terminated when the S/G water level reaches the high-high setpoint
5) No credit is taken for the heat capacity of the S/G metal, or initial S/G water or steam heat capacities
6) Cases for Reactor power at 0% and full power are evaluated.

Using these extraordinary assumptions, this accident is still less severe than a excessive load increase or rod-withdrawal accident. Clearly, overfeeding a S/G because either a MFRV, or Bypass valve, is in manual is much less limiting than the assumptions used in the accident analysis. Furthermore, the most probable cause for an operator error induced accident is due to a valve being placed in manual (such as a FW Bypass valve) and a reactor trip occurring at full power. In this case, the operator error (not recognizing the RCS cooldown is being caused by the excessive FW) will be corrected by Emergency Operating Procedures E-0, ECA-0.0, or ES-0.1. These procedures ensure proper FW and S/G water level control.

The UFSAR accident remains bounding for any excessive FW addition accident. Furthermore, the probability of equipment malfunction causing the accident or plant transient is reduced, since the activity is utilized to stop FW transients/oscillations which may potentially increase the wear on the MFRV's and/or Bypass valves. There is no quantitative method for evaluating the frequency of operator errors. However, in the event that an operator errs in FW Control, the error will be revealed and resolved by the Station Procedure that is utilized to recover the unit after a transient. In addition, excessive overfeeding will be terminated @ 75% narrow range level by a trip of the Main Feedwater pumps. This will stop the flow of FW to the S/G's.

Nuclear Safety Analysis and Mech. Engineering have reviewed this condition. The maximum calculated flow that can be achieved through three (3) full open main feedwater reg. valves plus three (3) full open bypass FW reg. valves is 133% of nominal full-power FW flow. This is bounded by the current FW flow analyzer which assume a 200% of nominal FW flow at accident initiation.

<u>NOTE:</u> The current UFSAR description shows 200% to one generator. The latest Westinghouse analysis is 200% to all generators.

SAFETY EVALUATION NUMBER 91-SE-OT-027

DESCRIPTION

Evaluate of potential head deficit on the motor-driven auxiliary feedwater pumps 1/2-FW-P-3B reduced the design basis AFW flow for the main feedline break event from 340 gpm to 300 gpm.

The purpose is to support continued full power operation within current Technical Specifications limitations in conjunction with AFW flow requirements for the main feedline break event and Station Emergency Operating Procedures.

SAFETY ANALYSIS SUMMARY

Technical Report NE-827, Rev. 0 presents a discussion of the analysis and evaluations supporting the conclusion that the minimum motor-driven AFW pump flow can be reduced and still meet the most limiting acceptance criterion for the main feedline break event. No other UFSAR transients are affected by this change. The existing North Anna setpoints remain valid.

This evaluation does not address any change to the present AFW flow test acceptance criterion. It is expected that any future change to the flow test acceptance criterion will be addressed via a separate 50.59 evaluation in conjunction with associated procedure changes.

SAFETY EVALUATION NUMBER 91-SE-OT-028 and
91-SE-OT-028 Rev. 1

DESCRIPTION

The change involves extending biennial procedure reviews to every four years based on the guidance provided in ANSI N18.7. This change requires a modification to the QA Topical Report.

The purpose of this change is that our current programmatic procedure reviews and activities are meeting the intent of the Biennial procedure review requirement from ANSI N18.7-1976 and that given these programmatic reviews and activities should be extended to a 4 year review to allow for more focus to our programmatic review activities, such as the procedure upgrade program.

SAFETY ANALYSIS SUMMARY

The QA Topical commits us to follow ANSI N18.7. As currently written, the Topical does not allow us to revise the periodicity of procedure reviews. Therefore, to change the frequency of procedure reviews a QA Topical report change is necessary. ANSI N18.7, Section 5.2.15 contains the requirement for biennial procedure reviews. However, the ANSI standard is flexible enough to allow the modification or deletion of this requirement based on the following:

- ANSI N18.7 states that the frequency of reviews may vary depending on the type and complexity of the activity involved an may vary with time as a given plant reaches operational maturity.

- ANSI N18.7 also states that to ensure that the procedures in current use provide the best possible instructions for performance of the work involved, systematic review and feedback of information based on use is required.

Therefore, our interpretation of ANSI N18.7 is that it was structured to require new plants to review procedures more frequently than a plant which has reached operational maturity. Once a plant has reached operational maturity (as our plants have), the requirement then focus on maintaining the best possible procedures through systematic feedback. We are extending the current biennial procedure review requirement to a periodic procedure review requirement (the period will be every 4 years) based on the guidance provided in ANSI N18.7 and overall improved quality of the procedures now being used at the stations. Note that procedures that currently require an annual review will be unaffected by this change. In addition, to ensure continued high quality procedures we will periodically assess a sample of procedures to verify that more frequent reviews are not necessary.

SAFETY EVALUATION NUMBER 91-SE-OT-029

DESCRIPTION

The purpose for this change is to satisfy our requirement to submit an update to the QA Topical report annually.

SAFETY ANALYSIS SUMMARY

The changes to the Operational Quality Assurance Program Topical Report have been reviewed with respect to the criteria defined in 10CFR50.59. The changes to the Topical Report do not reflect or affect changes in the facility as described in the UFSAR or propose the conduct of tests or experiments not described in the UFSAR. The changes do reflect changes in the standards (procedures) and the organization implementing these standards which may be referenced in the UFSAR. These changes are deemed not to involve an unreviewed safety question. These changes are editorial and organizational changes and serve only to enhance administrative controls at the power stations and improve the effectiveness of the quality assurance program.

SAFETY EVALUATION NUMBER 91-SE-OT-030

DESCRIPTION

Revision of section 13.1, Organizational Structural, of the UFSAR. This re-write reflects the current operating organizational structure of the Station.

To reflect the current Station Operating Organization, to remove extraneous material, and to simplify the descriptions used.

SAFETY ANALYSIS SUMMARY

This change provides a re-write of the position descriptions and reflects the current Operating Organization of the North Anna Power Station. Changes to the UFSAR for the Operating organization and descriptions of this organization do not reflect the accidents analyzed in other sections of the Safety Analysis Report. Regulatory requirements, Station equipment, components, systems, operating procedures, and Technical Specifications remain unaffected by these internal organization changes Therefore, this change does not constitute an unreviewed safety question.

SAFETY EVALUATION NUMBER 91-SE-OT-031

DESCRIPTION

3/4.7.3.1 is being changed to require 3 operable component cooling subsystems, provide actions for 1 or 2 inoperable subsystems and provide for determining operability of the component cooling pumps for as per T.S. 4.0.5 if either unit is Modes 1-4. 3/4.7.3.2 is being added. The bases are being changed or added to reflect the revised specifications.

3/4.7.3.1 is being changed to reflect the design basis as described in the UFSAR. 3/4.7.3.2 is being added to ensure that an adequate complement of component cooling components is operable if both units are in Modes 5 or 6.

SAFETY ANALYSIS SUMMARY

This is a Technical Specification change and no modifications of hardware or systems is required. All previously evaluated component malfunctions associated with component cooling are still valid. In addition, no accidents associated with component cooling are identified in Chapter 15, Accident Analysis, of the UFSAR.

This change will ensure that the design basis as described in the UFSAR is met. In addition, the margin to safety will be increased by requiring an additional component cooling subsystem to be operable in Modes 1-4, and 2 component cooling subsystems to be operable in Modes 5 and 6.

SAFETY EVALUATION NUMBER 91-SE-OT-032

DESCRIPTION

Fuel rods will be removed from fuel assembly AM1 and tested for profilometry, length, and oxide thickness.

The rods to be tested have advanced cladding material. The tests will provide data on how the rods performed after two cycles of irradiation

SAFETY ANALYSIS SUMMARY

North Anna fuel assembly AM1 is an advanced materials demonstration assembly which recently acquired two full cycles of operation. Westinghouse is prepared, at their expense, to mobilize their test equipment and perform length, profilometry, and oxide measurement tests on selected fuel rods with advanced cladding materials. The Westinghouse equipment has been modified and fully tested prior to its arrival at North Anna. The Vantage-5 fuel rod handling tool whose rod g ripping device failed resulting in a dropped rod at North Anna in 1990 will not be used during these tests. Instead, the MFRS handling tool which successfully handled over 264 fuel rods during the same project when the Vantage-5 tool failed will be used. However, the Vantage-5 tool with a redesigned rod gripping mechanism will be brought as a backup to the MFRS handling tool. The tests that Westinghouse will perform are integral to the development of high performance advanced cladding materials which North Anna has a direct benefit in terms of the development of materials which reduce the amount of fuel rod oxidation. Only one fuel assembly or one individual fuel rod is handled at any given time. The lids to the fuel assembly inverting basket are checked to ensure they are locked any time a fuel assembly is inverted. The collect for the fuel rod handling tool has been load-tested to ensure adequate rod gripping force. All fuel will be handled in accordance with existing approved Station procedures. The fuel assembly inverting rig will be securely mounted to the fuel pool deck to prevent the equipment/fuel from dropping into the pool. This is the same equipment that was used to reconstitute eight fuel assemblies and recage one assembly at North Anna in early 1990. Upon completion of the tests, fuel assembly AM1 will have all removed rods replaced in their respective assembly locations. Thus, there will be no change to any system or component, nor will the operation of any system or component be changed as a result of performing the fuel rod examinations.

SAFETY EVALUATION NUMBER 91-SE-OT-033

DESCRIPTION

This procedure replaces the existing relay cover of device 94/P4 BU with a modified relay cover. The trip fuses for CB 242 will be removed during this cover replacement to prevent inadvertent trip of CB 242 and loss of "B" RSS.

The existing relay wiring prevents the relay cover from being installed properly. The cover is interfering with the relay armature such that the contacts are physically closer to making contact with the coil de-energized (non-actuating state).

SAFETY ANALYSIS SUMMARY

The relay cover replacement is required in order to prevent inadvertent loss of "B" RSS supply to the station due to vibration induced actuation of device 94/P4 BU which could occur in its existing condition. The trip fuses for CB 242 will be removed in order to prevent actuation of the relay during the replacement. The time required to replace this relay cover is relatively short, less than 1 hour. The probability of a fault occurring within this brief window is extremely unlikely. Protection of plant equipment from a fault on the primary side of the "B" RSS transformer is still provided by breakers L102 and/or 15E1 and 15E3.

SAFETY EVALUATION NUMBER 91-SE-OT-034

DESCRIPTION

Evaluate use of binder clips (such as 1DL Binder clip No. 10020) to attach tags to breakers which do not have an attaching device.

The purpose is to allow use of binder clips for breaker tagging.

SAFETY ANALYSIS SUMMARY

Use of binder clips does not impede breakers operation and, in fact, enhances the equipment and personnel safety aspects of tagging breakers by providing greater assurance that the tag will remain attached to the breaker. This will decrease the potential for equipment damage and/or personnel injury from inadvertently operating a "tagged out" breaker.

SAFETY EVALUATION NUMBER 91-SE-OT-035

DESCRIPTION

This TS change will modify tolerance values from the Undervoltage (UV) Trip Setpoints and Allowable Values for Items 7a and 7b of TS Table 3.3-4 and will change the values for Trip Setpoints and Allowable Values. This change also will modify the values for Trip Setpoints and Allowable Values for Item 6e from a percentage of bus voltage to actual voltage values.

Start Setpoints should be presented as fixed numbers rather than percentages of a bus voltage. UV setpoint and allowable values are changed to maximize protection to Class 1E components and to allow for future additions to the emergency busses.

SAFETY ANALYSIS SUMMARY

Technical Specification 3/4.3.2 Table 3.3-4 Item 6.e.
The Trip Setpoint, is changed from 57.5% of transfer bus voltage to 2392 volts on the transfer bus. The Allowable Value is changed from 52.5% of transfer bus voltage to 2184 volts on the transfer bus. 2392 and 2184 volts are the actual values represented by 57.5% and 52.5% of 4160 volts (nominal voltage of the transfer busses). Therefore, these changes are of an administrative nature intended to promote clarity and ease of use of the specification.

Technical Specification 3/4.3.2 Table 3.3-4 Items 7.a & 7.b.
The tolerance values listed in Table 3.3-4 are modified by this change. The Trip Setpoints and Allowable Values represent the minimum values for the actuation of the Loss of Power sensing relays. The appropriate tolerances for the Loss of Power sensing relays are developed according to the characteristics of the specific relays used in the application. The tolerances are applied to the specification limits to establish actual instrument setpoint for the actuation of the relays. Thus the setpoints insure that the operation of equipment remains consistent with the assumptions and limitations defined in the UFSAR.

The assumptions used in the accident analysis require undervoltage protection to be actuated before voltage on the busses drops below a given value. Technical Specification instrument trip setpoints are derived from this value with added conservatisms. The undervoltage Trip Setpoints and Allowable Values are changed to maximize UV protection for the Class 1E equipment and to provide additional operating margins for loads to be added to the emergency busses. The calculations supporting this Tech Spec Change uses GDC methodology and calculational methodologies from GN-0030 for single element setpoints. These setpoints are not considered as primary protection setpoints.

The undervoltage and degraded voltage allowable values are changed to ensure UV protection for the Class 1E equipment and to provide clarification of the relationship between the Trip Setpoints and the Allowable Values.

SAFETY EVALUATION NUMBER 91-SE-OT-036

DESCRIPTION

This evaluates the contingency actions which have been taken to ensure positive isolation of penetration 56C due to 1-SS-TV-102A (RCS Cold Leg Sample Line Isolation Valve-inside containment) being inoperable.

The purpose is to provide a technical justification for obtaining a waiver of compliance regarding Technical Specification 3.6.3.1 so that power escalation and mode change may occur since the penetration is isolated and there is no leakage potential through this penetration to the outside atmosphere.

SAFETY ANALYSIS SUMMARY

Containment Integrity for penetration 56C has already been established without relying on an ESF actuation, and the leakage limits are acceptable for this penetration. However, LCO 3.0.4 applies to LCO 3.6.3.1, so the trip valves must be demonstrated to be OPERABLE and capable of performing a stroke within the required time for any increase in mode. Even though the penetration has been isolated and the trip valves deenergized in their safe position, an increase in modes is not allowed unless a waiver of compliance is received from the NRC. The plant is in a safer condition with these trip valves closed and deenergized, since an ESF actuation is not required to establish Containment Integrity for penetration 56C, so the waiver of compliance in this case is justified.

SAFETY EVALUATION NUMBER 91-SE-OT-037

DESCRIPTION

Early draindown of the RCS to mid-loop in order to perform maintenance on Loop "B" stop valve disc pressurization line.

The line is leaking and drainage to mid-loop is required to perform repairs.

SAFETY ANALYSIS SUMMARY

Operational Constraints

The analyses above suggest the following operational constraints for midloop operation in the current situation.

1) At least two steam generators should be available. This means:

   - SG level > 5% narrow range
   - AFW makeup capability to these generators
   - PORV's for these generators blocked open

2) RCS makeup capability on each safeguards bus consisting of at least a HHSI charging pump or a LHSI pump.

3) Draindown should not proceed before 48 hours. Alternately, draindown can proceed as early as 33 hours if all 3 SG's are available as defined in 1) above.

4) Calibrated instrumentation must be available for monitoring RCS level in the drained condition.

5) One loop stop valve bypass line should be open to avoid a potential hot side pressurization and loss of crossover leg fluid out the RCS breach at the loop B cold led stop valve.

We do NOT recommend venting of the RCS to the containment in this situation. Reflux cooling is more effective if the pressure is allowed to rise naturally to a saturation temperature that is several degrees above the secondary side. Use of an open loop stop valve bypass line should prevent any loss of fluid through the RCS breach due to hot side overpressurization.

Based on a review of the currently applicable analyses for loss of RHR events at mid-loop, we concludes that early draindown of the RCS (as early as 33 hours) does not invalidate the generic analysis, does not place the unit in an unalyzed condition, and does not pose an unreviewed safety question as defined in 10 CFR 50.59, provided the operational constraints outlined herein are adhered to.

SAFETY EVALUATION NUMBER 91-SE-OT-038

DESCRIPTION

The Boron Recovery Heat Exchanger will be cleaned with a Citric Acid solution at 200 degrees, 150 psig, pumped by a temporarily installed skid with high pressure hoses and fittings.

The purpose is the decontamination of the Heat Exchangers.

SAFETY ANALYSIS SUMMARY

The process involved in this procedure is the chemical cleaning of the stripper feed heat exchangers 1-BR-E-6A and 6B with a sequential addition, recirculation, and ion exchange of several agents. The process will use a skid mounted arrangement of pumps, ion exchangers, heaters, lines, and associated equipment. The system design and operation is compatible with the boron recovery heat exchanger. The chemical reagents used will not adversely affect the piping or components of the boron recovery system and they will be neutralized and flushed out prior to returning the system to operation.

During the cleaning process, the heat exchanger will remain isolated from the remainder of the boron recovery system. The skid mounted equipment is located in a safe manner with consideration for shielding and leak detection. There is no interaction with safety related equipment, and the procedure guidance is clear and complete. Any chemicals spilled will be contained in the auxiliary building sump and processed as a normal liquid effluent. The procedure outlines methods to be used to determine any system leakage and actions to take in the event of a spill or leak-by into the boron recovery system.

The cleaning process is design to reduce the radiological source term in an effort to decrease personnel exposure. The boron recovery system is described in the UFSAR but its operation will not be affected in any way.

SAFETY EVALUATION NUMBER 91-SE-OT-040

DESCRIPTION

Remove reference to daily operation of the flash evaporator.

The flash evaporator is not used.

SAFETY ANALYSIS SUMMARY

The flash evaporator is no longer used in daily plant operation. Makeup water is provided for the plant by a reverse osmosis unit located near the intake structure. The flash evaporator served no safety function.

SAFETY EVALUATION NUMBER 91-SE-OT-041

DESCRIPTION

This is a UFSAR change to provide clear descriptions and to correct typographical and grammatical errors, to correct an incorrect description, and to correct a reference to a figure and the title on the figure.

Typographical and grammatical errors were found in the review process.
UFSAR contains an incorrect description of the purpose of a chiller pressure switch.
Several system drawings are referenced in section 9.4.10.2 which in fact only cover one of the systems.
The UFSAR does not specify that a manual response is needed for the Control Room chillers in an emergency if power is interrupted and exceeds the protective circuit time setting.
Descriptions for instrumentation application for air conditioning systems and bottled air systems were unclear and could be interpreted incorrectly.

SAFETY ANALYSIS SUMMARY

The impact that the chiller pressure switch description change, a description clarification of the operation of the chillers in an emergency, a description clarification for the instrumentation application section, and typographical and grammatical corrections have on the Control Room and Relay Room Air Conditioning Safety Related System were the major issues considered in the Unreviewed Safety Question Determination. This UFSAR change is justifiable because it does not physically alter any equipment in the plant. It only contains changes which are already existing but need to be reflected in the UFSAR to avoid confusion. An Unreviewed Safety Question does not exist because the changes do not impact the plant in any way. This is so because:

1) Clarifying the description of the operation of the chillers in an emergency only provided the reader with clearer information about the system.

2) Correcting the description of the purpose of the chiller pressure switch only helps the reader understand the chillers better as well as eliminating incorrect information in the UFSAR.

3) Clarifying the description of the instrumentation application section only eliminates the possibility for human error when interpreting the UFSAR.

4) Typographical and grammatical corrections only aid the reader when reading the UFSAR.

SAFETY EVALUATION NUMBER 91-SE-OT-044

DESCRIPTION

Manually place 1-SI-MOV-1865B, Accumulator Tank B Isolation Valve, on the backseat and adjust the external limit switch to allow this.

To place 1-SI-MOV-1865B on Backseat to stop a steady drip packing leak.

SAFETY ANALYSIS SUMMARY

1-SI-MOV-1865B is normal locked open during power operation to ensure that the "B" Safety Injection (SI) Accumulator, 1-SI-TK-1B, passively discharges into the Reactor Coolant System (RCS) in the event of a major primary or secondary loss of coolant accident. Under certain circumstances during accident recovery actions the accumulators may be isolated to prevent nitrogen discharge to the coolant system or discharge of the accumulator water inventory if control of RCS leakage has been established and controlled depressurization is occurring. Discharge of the accumulators at these particular times would exasperate but not prevent recovery actions.

1-SI-MOV-1865B currently is experiencing a minor packing leak which will very slowly degrade the carbon steel studs if allowed to continue. Either the leak must be fixed or periodic inspection of the valve established to assure continued integrity of the studs. The most benign fix is to place the valve on its back seat.

To fix the leakage it will be placed on the back seat and the external snap lock type position limit indicator adjusted so the operator in the control room can still remotely verify the valve is still full open on its back seat.

1-SI-MOV-1865B has a stellite No. 6 backseat with very limited seating area. The design of the valve is such that very limited wedging action will take place when the valve is placed on the backseat. The valve may experience a heatup of approximately 200 degrees F during a design basis accident; however, this change is not expected to significantly affect seating force. If placed on the back seat the valve should close with essentially the same reliability as other SI accumulator valves.

1-SI-MOV-1865B has an open and closed limit torque switch. The open torque switch is bypassed for the first 80 to 85% if travel is from the closed seat in the open direction. The closed torque switch is bypassed only at the 100% open or greater position. The closed limit bypass will be maintained as the valve is opened past the 100% open position. This helps assure automatic valve closing.

If 1-SI-MOV-1865B fails to close in post accident recovery the Emergency Operating Procedures specify a qualified vent path to depressurize the SI accumulator and prevent Nitrogen injection into the RCS.

No adverse consequences on valve reliability or operation are expected as a result of the planned backseating attempt.

SAFETY EVALUATION NUMBER 91-SE-OT-045

DESCRIPTION

Change required actions for Unit 1 nonfunctional penetration fire protection barriers to make them consistent with Unit 2.

To avoid misinterpretation of required actions between units.

SAFETY ANALYSIS SUMMARY

This change will alter the Unit 1 requirements for nonfunctional penetration fire protection barriers in the conservative direction. Protection requirements will in no way be degraded due to the change, but compliance with the requirements will be enhanced due to the simplification associated with commonality between units. Since this change to the UFSAR will preclude misinterpretation of the fire protection requirements while making the requirements more conservative, this change should be allowed.

SAFETY EVALUATION NUMBER 91-SE-OT-046

DESCRIPTION

This evaluation is being performed to assess the 1991 update of the North Anna Power Station Appendix R Report. It incorporates design changes completed in 1990 and information concerning Appendix A to APCSB 9.5-1, Fire Area Commitments.

This change incorporates modifications to the plant which impact the Appendix R program and updates the program to reflect correct plant configurations.

SAFETY ANALYSIS SUMMARY

The 1991 update of the North Anna 10 CFR 50 Appendix R Report incorporates changes and plant modifications made since revision 7. Changes made to the plant by either by Design Change Packages (DCPs) or Engineering Work Requests (EWRs) are controlled by Virginia Power General Engineering Nuclear Standard STD-GN-0021, "Appendix R Design Guidelines. If during a modification the Appendix R program documentation is affected, Attachment 5.3, "Appendix R Report Change Notification" form, of the standard is required to be completed. The 1991 update of the Appendix R Report complies and incorporates all submitted "Appendix R Report Change Notification" forms since revision 7 of the Report was issued. In addition, all DCPs and EWRs completed in this time period were researched to determine if any additional modifications may have been performed which could have an impact on the Appendix R Program and which were not identified as such. This additional review did not identify any generic concerns, and it was concluded that the procedures addressing Appendix R Report changes are adequate.

SAFETY EVALUATION NUMBER 91-SE-OT-047

DESCRIPTION

Power operation with low pressure turbine blades removed.

Blade #102 on LP1 governor end has an indication near its key to the turbine rotor. Operation with this indication is not acceptable, so the blade will be removed along with blade #11, its 178 degree counterpart (to keep the turbine balanced).

SAFETY ANALYSIS SUMMARY

As stated by Westinghouse, the removal of the two blades will not adversely impact turbine operation. Vibration levels may change, but will not exceed normal operating limits. Operators will still be required to trip the turbine if vibrations become excessive. The turbine will also be tripped if condenser backpressure exceeds the limits specified in CAL 86-02. The ability of the turbine to trip when required is unaffected. The probability and severity of a turbine damage resulting in a reactor trip or turbine missile damage is not significantly increased.

SAFETY EVALUATION NUMBER 91-SE-OT-048

DESCRIPTION

Change the text of UFSAR Section 9.5.8.2 to state that the surface of the ground outside the air intake is paved and that operations personnel take logs periodically.

Plant configuration and station practice (as noted above) do not concur with UFSAR descriptions. To correct these inconsistencies, a change in the text of the UFSAR is necessary.

SAFETY ANALYSIS SUMMARY

This change simply updates the UFSAR to conform with current practices and physical plant layout. Having a paved surface outside the Emergency Diesel Generator (EDG) intake loovers rather than a crushed stone surface further limits dust intake and is an improvement. Reducing log taking frequency in the EDG rooms does not introduce a higher probability of dirt/dust collection since this is a long term accumulation problem. At least once per 12 hours or twice a day is more than sufficient to detect dirt buildup and have appropriate actions taken before it becomes a significant problem.

SAFETY EVALUATION NUMBER 91-SE-OT-049

DESCRIPTION

Change the text of UFSAR section 9.5 to state that compressor relief valves are set to open at 275 psi and that the first (of two) tube-oil high-temperature switch sounds an alarm if lube oil temperature reaches 225°F.

This change is necessary to correct inaccuracies/inconsistencies between the UFSAR and other station documents observed during the 1991 EDSFA.

SAFETY ANALYSIS SUMMARY

The proposed UFSAR text change does not involve an unreviewed safety question as this change is necessary to create concurrence between this document and current station practices. Compressor relief valves have previously been set to open at 225 psi and the first (of two) lube-oil high-temperature switch sounds an alarm at a lube oil temperature of 225°F. Station documents and technical manuals reflect these current setpoints. (See Attachments). Since these design bases are in effect, there is no increased probability of accidents or malfunctions to previously evaluated safety related equipment created by these changes. To rectify inconsistencies between UFSAR descriptions and station documents, UFSAR section 9.5 needs to be changed.

SAFETY EVALUATION NUMBER 91-SE-OT-050

DESCRIPTION

The postulated primary to secondary leakage in the faulted steam generator for a main steam line break is assumed to increase the 10 gpm when the steam generator depressurizes. The previous assumed leakage for control room operator dose calculations was less than 1 gpm.

To support operation of North Anna Unit 1 Cycle 9 to end-of-cycle prior to performing the next steam generator inspection.

SAFETY ANALYSIS SUMMARY

A review of the current steam generator tube integrity status for North Anna Unit 1 has been performed, and an assessment made of the potential primary-to-secondary leak rates under accident conditions at the end of the current operating cycle. A Westinghouse assessment of the observed tube flaw characteristics led to the conclusion that an upper limit primary to secondary leak rate under accident (main steam line break) conditions is 9.5 gpm to the faulted steam generator. This limit was developed based on the assumption that the currently established administrative limit for leakage during normal operation of 50 gpd in any steam generator continues to be adhered to.

Based on this result, the existing licensing basis analyses for both control room and offsite doses following a steam line break were reviewed for the potential impact of a 10 gpm primary-to-secondary leak rate.

For the offsite consequence analysis documented in the UFSAR, the 10 gpm leak rate is bounded by the analysis assumption. Further, when the analysis is reperformed with the USNRC Standard Review Plan Methodology currently used in the industry and assuming a 10 gpm leak rate, the doses remain well within the SRP acceptance criteria.

For the control room dose analysis, the 10 gpm leakage is higher than the assumption used in the current licensing basis analysis. However, it has been demonstrated previously that the current licensing analysis is based on a physically unrealistic model for transport of radiosotopes from the steam release point in the turbine building to the control room. When a more realistic (but still conservative) transport model is used in conjunction with a 10 gpm primary-to-secondary leak rate assumption the doses remain bounded by those reported in the currently docketed analysis.

As a result, it is concluded that operation of North Anna Unit 1's steam generators through the end of the current refueling cycle will not create the potential for accident conditions more severe than already assessed in the currently docketed safety analyses.

SAFETY EVALUATION NUMBER 91-SE-OT-051

DESCRIPTION

While reviewing the code requirements for procurement of 8 new Recirculation Spray Heat Exchanger Service Water radiation monitoring pumps in Units 1 & 2 Quench Spray Pump House (QSPH) basements, QA questioned why the original purchase specification (NAS-184) did not call out ANSI B31.7 code compliance. This raised an operability concern for the existing pumps as noted in deviation report DR# N-91-1127. The intent of this evaluation is to answer the question of operability regarding the existing pumps (1(2)-SW-P-5,6,7, and 8).

SAFETY ANALYSIS SUMMARY

The pumps are considered Safety Related class III. Since they are not normally isolated they function as a system pressure boundary. These pumps are purchased commercial grade as allowed under purchase specification NAS-290 for pumps operating below 150 psi and 212°F. The pumps have a nominal flow rate of 6 GPM and a casing design pressure and temperature rating of 175 psig and 212°F. The Service Water system parameters expected during a LOCA would not exceed 150 psi or 212°F. Therefore, these pumps exceed the system design requirements for their application. Additionally, they meet the seismic requirements specified in NAS 184.

The associated radiation monitor are not specifically identified or discussed in the Tech Specs. Section 11.4 of the UFSAR does discuss the monitors and states they are only required during a LOCA event in order to detect Recirculation Spray Heat Exchanger tube leaks. The existing pump requirements were determined by Stone and Webster engineering analysis when designing the original system for expected conditions. From this data the original purchase specification NAS-184 was developed.

Performance testing to evaluate the function of the radiation monitors is performed at least every two years. This involves flowing Service Water through the pumps to the monitor. This testing is further evidence that these pumps perform their design function and so does not pose an operability concern as implied in the DR.

An extensive evaluation performed by EcoTech/RAM-Q (Report # 2132) for the Aurora replacement pump (1-SW-P-8) provides documentation of the suitability of the pump for the application. As this replacement pump is essentially identical to the original pumps, the EcoTech Report may be considered as further evidence to the adequacy of the original pump design.

SAFETY EVALJATION NUMBER 91-SE-OT-052

DESC'IPTION

1) The section on r. ssurizer Water Level currently states that no credit is taken for this trip in the accident analyses. This is not correct. In certain cases, the Pressurizer High Water Level trip is now assumed to operate. The section on Steam Generator Water Level currently states that a Turbine trip will cause a Reactor trip if above the P-7 setpoint. This is not correct. A turbine trip will initiate a Reactor trip if above the P-8 setpoint. 2) UFSAR is being changed to reflect that a Turbine Trip-Reactor Trip is interlocked with P-8.

1) In certain cases, which were analyzed for increasing the allowable Moderator Temperature Coefficient values and how the departure from nucleate boiling ratio (DNBR) is analyzed, the pressurizer filled prior to a High Flux or Overtemperature ΔT trip when no credit was taken for the Pressurizer High Water Level trip. Because of this, the Pressurizer High Water Level trip is now assumed to operate in these safety analyses. 1 & 2) DCP 88-03 & 88-04 changed Turbine Trip-Reactor trip permissive to P-8.

SAFETY ANALYS!S SUMMARY

No modifications are being made to the plant. This Safety Evaluation is being performed for a change to the Bases of Section 2.0 of the Technical Specifications and to Section 7.2 of the UFSAR.

The Safety Evaluation that was performed for License Amendments 112 and 100 for Units 1 and 2 respectively took credit for the Pressurizer High Level Trip in the Safety Analysis. Based upon this previous Safety Evaluation and the issuance of the License Amendments, the change to the Bases of Section 2.0, specifically the part concerning the Pressurizer Water Level trip, does not constitute an Unreviewed Safety Question.

Safety Evaluations were performed for DCP 88-03, DCP 88-04, License Amendments 119 and 103 for Unit 1 and 2 respectively, and Revision 14 of the UFSAR (which partially incorporated DCPs 88-03 and 88-04 into the UFSAR). Based upon these Safety Evaluations and the issuance of License Amendments 119 and 103, changing Section 7.2 of the UFSAR to reflect that the Turbine Trip-Reactor Trip is interlocked with P-8 and the Bases of Section 2.0 of the Technical Specifications, specifically the part concerning Steam Generator Water Level, does not constitute an Unreviewed Safety Question.

SAFETY EVALUATION NUMBER 91-SE-OT-053

DESCRIPTION

The proposed Technical Specifications change affects Surveillance Requirement 4.4.5.4.a.9. The change deletes the schedular requirement which requires the preservice eddy current examinations of the tubes of the replacement steam generators be performed after the field hydrostatic pressure test. The proposed Technical Specifications change provides the benefit and flexibility of performing the required preservice inspections of the replacement steam generator tubes at the vendor's fabrication facility. This inspection schedule is a suitable alternative to performing steam generator tubing eddy current examinations in the field after installation.

SAFETY ANALYSIS SUMMARY

The purpose of this amendment request is to revise the Technical Specification acceptance criteria for preservice inspection of steam generator tubes by removing the unnecessary schedular restriction that the preservice inspection be performed after the field hydrostatic pressure test. North Anna Power Station's inservice inspection program for steam generator tubing conforms to the requirement of ASME Section XI, the North Anna Technical Specifications, and the guidance of NRC Regulatory Guide 1.83, Revision 1.

The proposed amendment would affect only the schedule for performing a preservice inspection of tubing in replacement steam generators by removing the restriction that the preservice inspection be performed after the field hydrostatic pressure test. This proposed change does not affect or change any limiting conditions for operation (LCO) or any other surveillance requirements in the Technical Specifications for North Anna Units 1 and 2. In addition, the proposed change continues to comply with the requirements of NRC Regulatory Guide 1.83, Revision 1, and ASME Section XI. Further, this proposed amendment is identical to the one issued for Surry Power Station Units 1 and 2.

The proposed amendment continues to ensure that preservice inspection of replacement steam generator tubes will be performed to establish the baseline condition of the tubing. Further, the inspection will continue to be performed prior to resumption of service following the replacement. Therefore, the change continues to ensure that subsequent inservice inspections will provide evidence of structural degradation of the steam generator tubes.

The proposed Technical Specifications change would provide the benefit and flexibility of performing required preservice inspections of the replacement steam generator tubes at the vendor's fabrication facility. In accordance with Regulatory Guide 1.83, Revision 1, this inspection schedule is suitable alternative to performing the tubing examinations in the field after the replacement steam generators have been installed. The current reactor coolant system reliability and operation are maintained in accordance with the descriptions found in the UFSAR. Further, the proposed change does not affect the assumptions, design parameters, or results of any UFSAR accident analysis.

The operability of each steam generator will continue to be verified by the augmented inservice inspections required by the Technical Specifications. This is not an operability concern for either the current steam generators or the replacement steam generators.

SAFETY EVALUATION NUMBER 91-SE-OT-054

DESCRIPTION

Mechanically block open Containment Isolation Valve 2-CC-TV-201A.

The purpose of manually blocking open 2-CC-TV-201A is to allow replacement of a leaking air line to the pressure regulator for the valve. 2-CC-TV-201A is the outside containment isolation valve on the common RCP thermal barrier return line. It is desired to maintain RCP thermal barrier flow during this maintenance evolution.

SAFETY ANALYSIS SUMMARY

2-CC-TV-201A has an air leak on the instrument air line going to the air pressure regulator. This leak is small and does not adversely affect valve performance. However, it is prudent to repair this leak before it becomes a problem. 2-CC-TV-201A is the outside containment isolation valve for the common RCP thermal barrier return line. As such, it is desired to maintain this valve in the open position during the maintenance effort. This can be accomplished by mechanically blocking open the valve prior to isolating the air for maintenance.

Blocking open 2-CC-TV-201A renders this valve inoperable with regard to its containment isolation function. This is acceptable as long as the associated containment isolation valve (2-CC-TV-201B) is operable and the work is completed within the Tech Spec LCO allowed 4 hour time period.

SAFETY EVALUATION NUMBER 91-SE-OT-056

DESCRIPTION

Operation without component cooling water flow to the RCP thermal barriers.

Perform maintenance on containment isolation valve 2-CC-TV-201A with the unit at power.

SAFETY ANALYSIS SUMMARY

Loss of thermal barrier flow to an RCP does not affect the ability of the RCP or its seal package to perform its design function. The thermal barrier acts only as a back-up source of seal cooling in case seal injection flow is lost. If seal injection is lost while no thermal barrier flow exists, the RCP(s) and the reactor will be shutdown in accordance with abnormal procedure 2-AP-33.2, Loss of Seal Cooling. It is unlikely, however, that seal injection will be lost due to the inherent redundancy of the system. Seal injection flow is provided by any one of three charging pumps and delivered through either the normal or alternate charging flow paths. In addition, the seal injection lineup includes only one air-operated valve (HCV-2186) which fails open on a loss of air. The additional redundancy built in to the seal injection subsystem ensures an extremely high degree of reliability and minimizes the potential for a loss of seal injection during the period that CC is isolation to the thermal barrier. Due to the high degree of reliability in the seal injection system, therefore, it is unlikely that the isolation of CC to the thermal barrier will have any effect on RCP or plant operation.

Reactor Coolant Pump vibration will be monitored to ensure that pump operability is not in jeopardy. Pump bearing temperature will be monitored using the P-250 process computer points on a five minute trend to ensure that degradation of the RCP radial bearings is not imminent. Any adverse trend in these parameters will lead to pump and reactor shutdown.

While work is in progress on 2-CC-TV-201A, 2-CC-TV-201B will be closed and deenergized in accordance with T.S.3.6.3.1, therefore maintaining acceptable containment integrity.

Operation of the RCPs without CC cooling to the thermal barrier does not affect any systems required to mitigate accidents or maintain safe shutdown. In the unlikely event that the RCPs are secured due to a loss of seal injection, natural circulation can be used to maintain RCS heat removal. Natural circulation heat removal has been successfully demonstrated on North Anna Unit 2.

SAFETY EVALUATION NUMBER 91-SE-OT-057

DESCRIPTION

Three 6' sections of tygon hose in loop rooms below honeycomb supported with conduit clamps to unistrut members.

These sections of tygon hose could not be removed as they are not easily accessible and are securely fastened in place with clamps.

SAFETY ANALYSIS SUMMARY

The hoses are held securely with clamps. If they should become loose, they are too large to pass through the recirc sump screens, yet too small to cause significant flow blockage.

The hose is rated for 240 degrees assuming it is the same type as currently purchased. This temperature is below the maximum peak containment temperature reached during a LOCA or secondary pipe rupture inside containment. The hose is therefore assumed to reach its melting temperature. This melted material would disperse and is expected to harden sufficiently prior to its reaching the RS sump screens and therefore will either be trapped by the screens or, if small enough, would pass through the RS pumps and spray nozzles without damage due to the size of the fine mesh screens.

SAFETY EVALUATION NUMBER 91-SE-OT-058

DESCRIPTION

1. Correct the reactor coolant letdown high range rad monitor detector type to a Geiger-Mueller tube in Table 11.4-1.

2. Delete reference to the reactor coolant letdown low range radiation monitor from Table 11.4-2.

The purpose for this change:

1. A Geiger-Mueller tube has been in place since at least 1984 in the high range system.

2. The low range system was permanently disabled by EWR 85-492 but UFSAR Table 11.4-2 was not identified during the review process.

SAFETY ANALYSIS SUMMARY

In the high range portion of the letdown radiation monitor system, a Geiger-Mueller tube has been installed since at least 1984. This has been determined by a review of the maintenance work history. UFSAR Table 11.4-1 states that the detector type is a gamma scintillation tube. A scintillation detector is typically used in applications involving particulate or liquid sample streams, but due to the higher activity levels involved in the reactor coolant letdown, a GM tube was installed. A GM tube is appropriate for this application and provides all of the necessary indication and trending capability assumed in the UFSAR.

The low range portion of the letdown radiation monitor system was disabled by the Jumper process shortly after unit operation began as a result of higher than anticipated activity levels. The jumper was changed into a permanent feature by EWR 85-492 but UFSAR Table 11.4-1 was not identified during the EWR process. A Safety Evaluation was written and approved in accordance with that EWR and no unreviewed safety concerns were found.

SAFETY EVALUATION NUMBER 91-SE-OT-059

DESCRIPTION

This change will update the Technical Specifications to reflect the installation of four piezometers and the deletion of four inoperable piezometers that was done under DCP 90-01-3.

Four of piezometers (P-12, P-13, P-16 and P-17) that were used to monitor ground water levels became inoperable. DCP 90-01-3, "Service Water Reservoir Addition" installed four additional open-tube piezometers at selected locations around the service water reservoir and deleted the inoperable piezometers. Each new piezometer is located in a monitored zone of the service water reservoir and does not represent a replacement of the failed piezometers. This change will update the Technical Specifications to reflect this replacement.

SAFETY ANALYSIS SUMMARY

This change to the Technical Specification reflects work that was done previously under DCP 90-0103, as such it will not involve an unreviewed safety question.

SAFETY EVALUATION NUMBER 91-SE-OT-060

SAFETY EVALUATION NUMBER 91-SE-OT-060

DESCRIPTION

Change the range of the Triaxial Response-Spectrum Recorders from 1-30 Hz to 2-25.4 Hz to reflect the as built configuration of the plant.

SAFETY ANALYSIS SUMMARY

This Safety Evaluation is being done for a Technical Specification and UFSAR change. No modifications are being made to any plant equipment. The recorders, although inoperable pursuant to the current Technical Specification definition, are capable of performing their design functions and comply with Regulatory Guide 1.12, as modified by the exceptions stated in Section 3A.12 of the UFSAR. Therefore, there are no unreviewed safety question.

SAFETY EVALUATION NUMBER 91-SE-OT-062

DESCRIPTION

Incorporation of training program accrediation information in lieu of description of operations/STA programs to conform to regulatory guidance which allows deletion of training program descriptive material for accredited operator/STA training programs.

SAFETY ANALYSIS SUMMARY

Revision of the Technical Specifications to conform with regulatory guidance for accredited operator/STA training programs is strictly an administrative change and will not involve an unreviewed safety question.

Furthermore, NRC guidance in NUREG-1262 concluded that proposed changes to the Technical Specifications, consistent with NRC guidance on this issue, were considered to be administrative in nature. Because the proposed changes are consistent with NRC guidance, we conclude that the changes are administrative, and no unreviewed safety question exists.

SAFETY EVALUATION NUMBER 91-SE-OT-063

DESCRIPTION

Hydrogen Peroxide is added to the RCS prior to refueling to help control the release of radiocobalt. The addition of hydrogen peroxide advances the natural oxygenation of the coolant system which would normally occur when the reactor head is removed. Due to the potential for developing an explosive gas mixture of hydrogen and oxygen, the Westinghouse position is to perform the coolant oxygenation with the system water solid with no bubble in the pressurizer. North Anna would like to continue to oxygenate at reduced RCS inventories.

The purpose of North Anna performing oxygenation at reduced RCS inventories is to reduce the impact on outage critical path.

SAFETY ANALYSIS SUMMARY

Oxygenation of the RCS at reduced inventories should be allowed because it reduces the impact on the refueling outage critical path and its significantly reduces radiation levels near the vessel during refueling. The addition of hydrogen peroxide to the RCS at reduced inventories is acceptable as long as RCS samples show that the gas space is sufficiently degassed (<4% hydrogen) and the RCS liquid is sufficiently degassed (<5.0 cc/kg). Periodic sampling of the RCS should continue during the oxygenation process to ensure that an explosive mixture does not develop. This RCS sampling will ensure that no unreviewed safety question develops.

SAFETY EVALUATION NUMBER 91-SE-OT-064

DESCRIPTION

This will reword the description (UFSAR Sect. 9-5, Page 9.5-58) for valve indication for Emergency Diesel Generator exhaust bypass values.

This will reflect the actual plant configuration. The discrepancy was identified during Fall 1991 NRC EDSFI.

SAFETY ANALYSIS SUMMARY

The change is required to reflect the actual plant configuration. The method presently described in UFSAR Section 9.5 and the proposed change accomplish the same task (i.e. provide means to verify valve position). Therefore, the probability of an accident or equipment failure is not increased. The consequences of an accident or equipment failure are not increased

SAFETY EVALUATION NUMBER 91-SE-OT-066

DESCRIPTION

Operation with the circulating water pump/waterbox key switches in DEFEAT.

To prevent unnecessary tripping of the circulating water pumps due to waterbox MOV drift.

SAFETY ANALYSIS SUMMARY

The interlock trips a running pump if its corresponding waterbox inlet or outlet MOV is not full open. The units are currently run with the interlock in DEFEAT due to the occasional drifting of the waterbox MOVs off their full open seat. This prevents unnecessary circulating water pump trips and subsequent unit perturbations.

There will always be one more waterbox than number of circulating water pumps running with the keyswitches in DEFEAT, therefore there will not be a chance of intake tunnel overpressurization.

SAFETY EVALUATION NUMBER 91-SE-OT-067

DESCRIPTION

The steam dump valves will be manually isolated and the control system will be checked for proper operation by injection of a simulated test signal.

The purpose is to functionally verify that the steam dump control system will properly respond to a plant transient.

SAFETY ANALYSIS SUMMARY

During the performance of this test, the steam dumps will be manually isolated and unavailable in the event of a reactor trip or sudden loss of electrical load. The design basis of the steam dumps is to provide up to approximately 40 percent of full load steam flow as a means of preventing a reactor trip on loss of electrical load. This capability will be lost during the performance of the test. This loss of load feature is not used in any of the accident analyses. It is provided as a way to reduce unnecessary reactor and turbine trips.

All of the accident analyses assume that the steam dump system is unavailable and that the excess steam load following unit transient or trip will be relieved by the steam generator PORV's (if available) and the main steam safeties. UFSAR Section 15.2.7 states that the pressurizer and main steam safeties are sized such that the dumps are not required. All of the remaining scenarios are bounded by the assumption that there is a coincident loss of off-site power, in which case the steam dumps will be unavailable.

No unanalayzed accidents could be introduced by this test since steam line depressurization and main steam line rupture have been previously considered. Excessive leakaby of the steam dump manual isolations would be less severe than either of those accidents.

SAFETY EVALUATION NUMBER 91-SE-OT-068

DESCRIPTION

Jumper out Cell #19 of EDG 1H Battery

The purpose for this change is to remove a deficient cell from the battery.

SAFETY ANALYSIS SUMMARY

The batteries are described in the UFSAR as 60 cells. This change is to reduce the number of active cells to 59 cells until the battery can be replaced during the next Unit One refueling outage.

The battery's terminal voltage and capacity will be reduced, but will still be adequate to provide accident loads as analyzed by Engineering.

The 1H EDG will be declared inoperable during this activity and not available for safety functions. No other systems will be affected since the Battery is a dedicated power supply for the 1H EDG.

The 1H EDG Battery is not electrically connected to any control or protection circuitry. The battery charger is supplied from the 1H bus but will be disconnected during the activity. Following jumper installation, the battery charger will still function as designed.

The battery will retain its ability to flash the EDG field and provide power to control relays for the EDG. Logics and sequencing remain unchanged.

SAFETY EVALUATION NUMBER 91-SE-OT-069 Rev. 1

DESCRIPTION

Jumper out Cell #19 and Cell #42 of EDG 1H Battery.

The purpose for this change is to remove two deficient cells from the battery.

SAFETY ANALYSIS SUMMARY

This change is to reduce the number of active cells to 58 cells until the battery can be replaced during the next Unit One refueling outage or sooner if spare cells are available.

The battery's terminal voltage and capacity will be reduced, but will still be adequate to provide accident loads and will be adequate to meet the Tech Spec terminal voltage requirement of 129 Volts.

The 1H EDG will be declared inoperable during this activity and not available for safety functions. No other systems will be affected since the Battery is a dedicated power supply for the 1H EDG.

The 1H EDG Battery is not electrically connected to any process control or protection circuitry. The battery charger is supplied from the 1H bus but will be disconnected during the activity. Following jumper installation, the battery charger will still function as designed.

The battery will retain its ability to flash the EDG field and provide power to control relays for the EDG (see attached analysis). Logics and sequencing remain unchanged.

The battery terminal voltage will be maintained between 129 and 130.5 volts as per the manufacturer's recommendation. The upper limit prevents accelerated cell degradation due to overcharging.

SAFETY EVALUATION NUMBER 91-SE-OT-070

DESCRIPTION

The Technical Specifications are being changed to reflect updated pressure/temperature operating limits and low temperature overpressure protection system (LTOPS) setpoints. Revised heatup and cooldown curves, applicable to 12 EFPY and 17 EFPY for Units 1 & 2, respectively, have been developed. Existing Technical Specifications pressure/temperature limits expire at 10 EFPY (April, 1993 for Unit 1; September, 1993 for Unit 2) and must be replaced to permit continued operation.

SAFETY ANALYSIS SUMMARY

The heatup and cooldown curves required by Appendix G of 10 CFR 50 have been extrapolated to 12 EFPY and 17 EFPY for North Anna Units 1 and 2, respectively, by including the effects of the incremental radiation exposure on the reactor vessel beltline region. The results are referenced to the analyses of the North Anna Units 1 and 2 Capsule U results. The revised Appendix G curves were prepared using standard B&W and Westinghouse methodologies including Regulatory Guide 1.99 Rev. 2. PORV setpoints were developed to provide bounding heatup and cooldown curve protection for the worst case mass and heat addition low temperature overpressure transients. The next Unit 1 reactor vessel surveillance capsule (Capsule X) is scheduled to be removed after the tenth fuel cycle (10 EFPY) which allows sufficient time for analysis prior to exceeding 12 EFPY. The next Unit 2 reactor vessel surveillance capsule (Capsule W) is scheduled to be removed after the thirteenth fuel cycle (15 EFPY) which allows sufficient time for analysis prior to exceeding 17 EFPY. The heatup and cooldown curves prepared by B&W and Westinghouse were determined in a conventional manner according to Section III of the ASME code as required by 10 CFR 50 Appendix G. Both steady-state and transient thermal conditions were considered in order to bound the possible combinations of pressure (i.e. membrane) and thermal stresses. The new North Anna Unit 1 low temperature overpressure protection system PORV lift settings should be less than or equal to 450 psig whenever any RCS cold leg temperature is less than or equal to 270°F, and less than or equal to 390 psig whenever any RCS cold leg temperature is less than 150°F. The new North Anna Unit 2 low temperature overpressure protection system PORV lift settings should be less than or equal to 510 psig whenever any RCS cold leg temperature is less than or equal to 321°F, and less than or equal to 360 psig whenever any RCS cold leg temperature is less than 210°F.

PTS evaluations were made for the limiting beltline locations. It was demonstrated that (a) predicted end-of-license fluences do not result in RT_{PTS} values in excess of the screening criteria when calculated using the methodology of Regulatory Guide 1.99, Revision 2; (b) there is an excellent comparison between experimentally determined and calculated vessel fluences; and (c) the extrapolated fluences at the burnup limit to which the revised heatup and cooldown curves are applicable for each unit are significantly less than the extrapolated end-of-license fluences (which have been demonstrated to not result in a violation of PTS screening criteria). On this basis it may be concluded that there is neither a significant change in predicted RT_{PTS} values; nor is there a PTS concern for either unit up to the burnup limit to which the revised heatup and cooldown curves are valid.

SAFETY EVALUATION NUMBER 91-SE-OT-071

DESCRIPTION

Remove the resistance temperature detector (RTD) bypass system and install thermowells that extend into the main reactor coolant system (RCS) piping.

This modification will remove a source of radiation exposure and will reduce the potential for RCS leaks.

SAFETY ANALYSIS SUMMARY

The probability and consequences of thermowell leakage or missile generation is bounded by the existing analysis for the RTD bypass system. The response time of the temperature measurement is important for accident analysis. The new system will reduce the response time due to water transport by 1.75 seconds and increase the response time due to thermal conduction by 1.75 seconds for a net change of zero. Because the Technical Specification response times do not include water transport, these figures will have to be increased by 1.75 seconds.

SAFETY EVALUATION NUMBER 91-SE-OT-072

DESCRIPTION

UFSAR Section 11.3..2 - Safety Considerations for the Waste Gas Disposal System, will be revised to reflect a change in Technical Specification (TS) requirements for the Waste Gas Decay Tank (WGDT). The TS change removes hydrogen monitoring from the explosive gas monitoring requirements.

To make the UFSAR consistent with current practice.

SAFETY ANALYSIS SUMMARY

The change is a clarification change which will make the UFSAR consistent with the Technical Specifications. Explosive mixtures in the WGDT are limited by maintaining the oxygen concentration less than 2%. Controlling oxygen concentration alone is sufficient to prevent an explosive oxygen/hydrogen mixture.

SAFETY EVALUATION NUMBER 91-SE-OT-073

DESCRIPTION

This is an update of UFSAR Table 6.2-72, Mechanical Penetrations, Containment Leak Rate Test Status" to be consistent with Technical Specifications. This changes system status (valve lineups) for Type A Testing and application of Type C penalties to Type A test results.

Update UFSAR table (revised 6/85) to reflect the current Technical Specifications (Table 3.6.1), Units 1 & 2, which has had several changes since then.

SAFETY ANALYSIS SUMMARY

The changes to the UFSAR Table 6.7-72 (Mechanical Penetrations, Containment Leak Rate Test Status) are to update it to be consistent with Table 3.6-1 (Containment Isolation Valves) in the Unit 1 & 2 Technical Specifications.

The piping penetration status (Vented, Flooded, Isolated, etc.) for the Type A test is changed for some of the penetrations. Also the requirements for adding a penalty from the Type C test results to the Type A test results is changed because of some of these status changes and other Technical Specification changes which have been made.

The appropriate verification of containment integrity is still made directly by exposure to test pressure or indirectly by adding Type C penalties. The change should be allowed to update the UFSAR to the current Technical Specifications.

An unreviewed safety question does not exist because containment integrity is still verified to meet the same requirements and individual penetrations are still tested/verified as before.

SAFETY EVALUATION NUMBER 91-SE-OT-074

DESCRIPTION

This test will deenergize the process cabinet's main power supply and verify that each cabinet's backup power supply energizes the cabinet.

The purpose is to functionally verify that the process cabinet backup power supplies will provide power in the event of a failure of the main power supply.

SAFETY ANALYSIS SUMMARY

These new procedures test the instrument process cabinet power supplies. The test is performed by de-energizing the primary power supply in each cabinet and verifying that the backup supply continues to provide power to the racks. In the event that a backup supply is found not to perform its required function, the instrument technician performing the test will restore cabinet power by re-energizing the primary power supply.

In the unlikely event that a test results in a card failure, a replacement card may be rapidly installed to restore instrument/control function.

Test conditions have been specified to ensure that potential automatic actuations, resulting from a loss of instrument/control power, are minimized. Where actuations can occur, these have been verified as conservative. When required, alternate channels will be selected to ensure that there will be no loss of control.

Only one cabinet will be tested at a time.

Every attempt will be made to perform this test with the unit defueled. In this configuration, the potential affects on RHR, NDT Protection, and RCS inventory control become less of a concern for decay heat removal and RCS integrity. The Safety Evaluation, however, was written to evaluate the test's impact during any cold shutdown condition, and therefore is conservative during a defueled condition.

SAFETY EVALUATION NUMBER 91-SE-PTU-001

DESCRIPTION

Change to procedure for removing RCS loops from service.

The procedure change allows for draining the RCS loops using a temporarily installed vacuum breaker arrangement vice the VA header.

SAFETY ANALYSIS SUMMARY

Draining the RCS loops by use of a vacuum breaker arrangement versus the VA header provides for a more efficient method. No unreviewed safety question exists because the evolution is allowed by T.S., the evolution is allowed by the UFSAR, and the PDTT subsystem will be operated within design conditions.

SAFETY EVALUATION NUMBER 91-SE-PTU-002

DESCRIPTION

Addition of a caution to procedure MOP-49-08 removing need to comply with requirements to throttle the CC Hxs and to delete those steps from procedure 1-CP-49.1.

The installation of a temporary (non-seismic) pump for removing Service Water from the header that is out of service.

The purpose for this change is to facilitate operation with one Service Water header out of service.

SAFETY ANALYSIS SUMMARY

These changes are acceptable because they ensure the ability of the Service Water system to respond to a CDA actuation on Unit 2 while one Service Water header is out of service. This capability is ensured even in the event of the additional failure of another Service Water pump or EDG. The probability of these failures is not increased because the end result of operation in this configuration is the same (i.e. all heat loads will still be serviced). The modified line-up of the system provides sufficient redundancy to protect against pump runout and to supply adequate Service Water flows to all necessary components in the limiting case of a Unit 2 CDA occurrence. Technical Specifications will be complied with at all times.

SAFETY EVALUATION NUMBER 91-SE-PTU-003

DESCRIPTION

To collect the baseline data of the Unit 1's Control Room Chillers for performance evaluation.

The purpose for this change is to satisfy Generic Letter 89-13, Service Water System Problems Affecting Safety Related Equipment.

SAFETY ANALYSIS SUMMARY

The Periodic Test is to obtain baseline data of the Unit 1 Control Room Chillers for performance evaluation. All activities related to the Periodic Test are considered routine in nature. Hence, the possibilities for errors or operational problems are small. The chiller being tested will be declared "inoperable." The other chillers will be secured but will remain "operable" as required. The secured chillers will be under the control of an operator during the testing.

SAFETY EVALUATION NUMBER 91-SE-PTU-004

DESCRIPTION

Change the position of 1-RC-141 and 1-RC-142, Pressurizer Spray Bypass Valves, from "Open" to "Closed" on Operating Procedure 1-OP-5A Valve Checkoff-Reactor Coolant.

The purpose for this change is to anhance pressurizer pressure control with leakby of spray valves.

SAFETY ANALYSIS SUMMARY

The Pressurizer Spray Valve leakby is large enough to provide the same function as manual bypass valves being open. The pressurizer spray line low temperature alarm provides sufficient warning of loss of spray flow. Pressurizer volume is small compared to total RCS volume, therefore, boron mixing effects are minimal.

SAFETY EVALUATION NUMBER: 91-SE-PTU-005

DESCRIPTION

a. Permanently changes acceptance criteria of the PM concerning Main Generator Exciter Rectifier fuses from: "No more than three (3) fuses with raised flags per phase (red, white, blue) per polarity. If more, generator must be shutdown and repaired" to "No more than three...If more, supervision has been notified and work order is submitted."

b. Also permanently changes acceptance criteria of PM from "The core monitor is operating properly." to "The core monitor is operating properly, or a work order has been submitted."

This permanent change allows supervision to make the decision of whether or not to shut the Main Generator down as the procedure previously required and as the Westinghouse technical manuals and letter dated 4/29/91 recommend.

SAFETY ANALYSIS SUMMARY

The limiting condition assumed in this safety analysis is not nuclear safety. The nuclear steam supply system is designed for a 100% loss of turbine load resulting from a main generator trip.

The Main Generator and Exciter are not in a Radiological Controlled Area.

The operation of the Main Generator is not safety-related, nor is it described in the UFSAR.

The main generator has no direct affect on reactivity during steady state operation. A Main Generator trip would result in a Reactor Trip.

The main generator has no connection with protective circuitry, emergency buses, or instrument buses. Reactor Protection, Emergency Diesel Generators, Vital Instrumentation, and Safety-Related loads are not affected.

The principle concern in this analysis is equipment and personnel safety.

Continuing to operate the Main Generator with 4 fuses blown on the red phase and 3 blown on the white phase limits the ability of the exciter to control main generator output voltage in a transient. In addition, one fuse is blown on the blue phase. In the previous 48 hours, another fuse blew on the inboard wheel. Normal operating limit curves on the main generator have been reduced to reflect the degraded state of the exciter before the last fuse blew. The Voltage Regulator is required to be in Manual

Base Adjust instead of Automatic, thus imposing an additional burden on the operators. Main Generator MVARs are now restricted to 0 MVARs, which would require the operators to manually adjust MVARs during system voltage changes or turbine load changes.

Continued operation of the main generator with 4 fuses blown on one phase subjects the remaining fuses to higher exciter current.

The current that was carried by the 4 blown fuses is now being carried by the remaining 16 fuses, which are not carrying approximately 125% of their normal current. As additional fuses are blown, the result may be a cascade effect, with the current load from each additional blown fuse being added to the remaining fuses. This load can result in a destructive failure of the last fuse which could spread contaminants throughout the rectifier wheel, causing arcing and burning which would result in a forced outage.

The potential also exists for an explosion or fire in the exciter since hydrogen leakage into the exciter is possible. A recent inspection of the exciter housing air revealed a 1% hydrogen concentration, well below the 4% level necessary to sustain a fire. A fire in the exciter may lead to fire and/or damage to the main generator.

Main Generator Exciter fuses will be monitored every six hours for further degradation. A generator outage to inspect/replace the fuses is scheduled for 5/10/91. This reduces the time that the unit remains in a degraded condition and therefore reduced the chances of an exciter failure and/or generator trip.

SAFETY EVALUATION NUMBER 91-SE-PTU-006

DESCRIPTION

The temporary procedure provides guidance for establishing, securing, and adjusting feedwater flow around the First Point Feedwater Heaters.

The purpose for this change is to provide instructions for directing operation of the First Point Feedwater Heater Bypass Valves to obtain 100 percent reactor power.

SAFETY ANALYSIS SUMMARY

The major issue considered was the long term affect of operating with feedwater partially bypassed around the first point feedwater heaters. Although operation in this alignment is different than described in the UFSAR, all potential accidents and malfunctions are bounded by UFSAR accident analysis.

This temporary procedure should be allowed because it is simple and provides increased electrical generation without sacrificing significant unit thermal efficiency or creating a safety concern. This Safety Evaluation is valid until the next refueling cycle. At that time inspection of the piping/tees associated with the first point feedwater bypass valves will be performed to verify adequacy for continued operation.

SAFETY EVALUATION NUMBER 91-SE-PTU-010

DESCRIPTION

Procedure to evaluate the following potential reactor coolant pump 1-RC-P-1C #1 seal flow paths to determine the total #1 seal leakoff flow:

a. 1-RC-P-1C standpipe flow (#2 seal leak-off)
b. 1-RC-P-1C #1 seal bypass line flow

The purpose for this change is to evaluate the cause of the low #1 seal leak-off flow associated with 1-RC-P-1C.

SAFETY ANALYSIS SUMMARY

Performance of this Procedure Action Request (PAR) will not impact pump operation or RCS integrity. RCP seal integrity will not be adversely affected by this evolution.

The potential exists for personnel injury when opening 1-RC-168 due to reactor coolant flashing. A caution has been placed in the PAR to warn the operator of this possibility.

An RWP will be prepared to properly outfit the operator that will be handling reactor coolant so that his dose will be minimized.

SAFETY EVALUATION NUMBER 91-SE-PTU-011

DESCRIPTION

This Procedure Action Request (PAR) changes the equations used to calculate Lithium Concentration Bands. This results in a change in Lithium Concentration Bands from what is included in the UFSAR.

Procedure revision to implement new boron-lithium bands for Reactor Coolant System chemistry in accordance with memorandum to G. Kane from W. Wigley, dated June 19, 1991.

SAFETY ANALYSIS SUMMARY

This change in the operating band for the Lithium-Boron ratio does not constitute an unreviewed safety question. The effects of the change have been evaluated by both Westinghouse and Virginia Power, and the change does not adversely affect the fuel cladding, RCS piping, or S/G tubes. Westinghouse has performed extensive research in the area of fuel performance (specifically cladding performance) at the proposed Lithium concentrations and has found that the cladding integrity is not compromised. Westinghouse has also performed extensive testing and analysis of Primary Water Stress Corrosion Cracking on the S/G tubes and has found that the proposed change can limit the Primary Water Stress Corrosion Cracking phenomena, and thus extend the lifetime of the S/G tubes. Finally, the Containment sump pH has been evaluated, and the higher Lithium concentrations were necessitated by the higher boron concentrations necessary at normal Beginning Of Cycle operation.

SAFETY EVALUATION NUMBER 91-SE-PTU-012

DESCRIPTION

Application of the Westinghouse Owners' Group (WOG) methodology (WCAP-11394-P-A) for performing the dropped rod(s) evaluation and potential removal of the administrative control rod insertion restrictions for operation in the automatic mode above 90% power.

The current analysis of record for the dropped rod event was performed by Westinghouse using the methodology of WCAP-10297-P-A. Use of the WOG methodology will provide consistency between the analysis of record and the verification on a reload basis of meeting DNB limits for the dropped rod event as they are both based on the same assumption bases, and correlations approved by the NRC. Use of this methodology also allow removal of the administrative control rod insertion restrictions for automatic control operation above 90% power.

SAFETY ANALYSIS SUMMARY

When operating at power, a dropped (withdrawn) control rod, single or multiple, may result in a transient leading to reduced margins to fuel design limits and in particular to DNB limits. This would be a result of increased power distribution peaking factors with the inserted (dropped) rods and a possible "return to power" transient, produced by feedback or automatic control, which might, depending on the control system, including a power level exceeding the initial level. A dropped rod(s) transient resulting from a single failure and exceeding DNB limits is considered unacceptable.

Normally the plant is protected from exceeding DNB limits through a negative flux rate trip system. The system will sense the initial rapidly decreasing neutron flux (as a negative rate) and trip the scram system thus ending the event for many of the dropped rod occurrences. For some events, however, the flux decrease rate may be insufficient for a trip. For these, it was originally thought the peaking factors and transient powers would not result in exceeding limits as very limited overshoot above the initial power level was assumed to result.

It was subsequently recognized that when operating with the Rod Control System in automatic, the control system could be reading a lower than average nuclear power level signal. This could cause a larger transient power overshoot than had been considered and result in lower DNB ratios than had been previously reported. The lower control signal and transient overshoot occurs if the dropped rod (which lowers the flux in its vicinity) is near the excore neutron detector used for control.

The problem is most limiting when the Rod Control System is in automatic and there is sufficient control rod bank reactivity worth inserted to allow the Rod Control System to withdraw the bank and raise the power into an overshoot condition. An "interim operational" solution for the problem was proposed by Westinghouse in 1979 which required either manual control or minimal control rod insertion (D bank control rods above 215 steps) above 90% power. The NRC found this proposal acceptable and in late 1979 Virginia Power notified the NRC by letter of the administrative imposition of these rod insertion restrictions at the North Anna Power Station.

The process used by Virginia Power to evaluate the dropped rod event now consisted of calculating the cycle specific radial power tilt (to determine the impact on the control signal) as well as the radial peaking factors and dropped rod worths for single and paired dropped rods for evaluation against a set of analysis limits to ensure the transient analysis results will remain acceptable. Virginia Power further modified this evaluation to include all single and paired dropped rods regardless of worth when it was questioned whether the negative flux rate trip would be actuated on worths greater than 300 pcm. Virginia Power has continued to employ this evaluation procedure described in the Virginia Power Nuclear Reload Design Methodology Topical and administratively apply the rod insertion limits for operation above 90% power while in automatic rod control mode.

Westinghouse, after the implementation of this interim operational solution, developed analytical methods which allowed removal of these restrictions on control rod insertion for operation above 90% power in automatic rod control mode. This methodology described in WCAP-10297-P-A, "Dropped Rod Methodology for Negative Flux Rate Trip Plants", retains the negative flux rate trip protection (for dropped rod worths of greater than 400 pcm for three loop plants) and evaluates the consequences of the transient for dropped rods which may not cause reactor trip on negative flux rate (dropped rod worths less than 400 pcm). Westinghouse submitted this methodology for NRC review in 1982. In 1983 the NRC found the methodology acceptable and indicated use of this methodology for the dropped rod analysis would allow removal of the rod insertion restrictions currently being administratively applied provided a nominal setpoint of 5% RTP with a time constant of 2 seconds was used for the negative flux rate trip.

In 1986 a core uprate program was implemented for North Anna which required reanalyses of most of the UFSAR Chapter 15 accidents. These reanalyses which were performed by Westinghouse, included a reanalysis of the dropped rod event using the methodology of WCAP-10297-P-A. Virginia Power at this time was performing the reload safety evaluations in-house for both the North Anna and Surry units. Without access to the WCAP-10297-P-A dropped rod methodology, Virginia Power continued to use the conservative method for reload dropped rod evaluations which required the restrictions on control rod insertion for operation above 90% power while in auto rod control mode for North Anna.

WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event", describes a dropped rod analysis methodology developed by Westinghouse and funded by the Westinghouse Owners' Group (WOG). This methodology approved by the NRC in 1989 is an extension of the methodology of WCAP-10297-P-A and eliminates the need to take credit for the negative flux rate trip for dropped rod worths greater than 400 pcm. The negative flux rate trip then becomes a backup trip and could be eliminated completely, if desired, via Tech Spec change.

In 1990 Virginia Power purchased from Westinghouse the transient database and methodology information necessary to perform the dropped rod analyses of either WCAP-10297 or WCAP-11394 in-house. Since the purchase of this information, Virginia Power has performed evaluations which show the applicability of the methodology, the correlations, and the transient database for analysis of the dropped rod event for North Anna Units 1 and 2. Having verified the applicability of the methodology, reload calculations were performed for North Anna Unit 2 Cycle 8 and North Anna Unit 1 Cycle 9 which provide assurance DNBR limits are met in the event of a dropped rod(s) for these cycles.

The verification of the applicability of this dropped rod evaluation methodology and its correlations and database for North Anna, along with the results of the evaluation for these cycles demonstrate that use of the methodology does not constitute an unreviewed safety question and the administrative rod insertion limits for automatic rod control operation above 90% power are no longer required when this methodology is used.

SAFETY EVALUATION NUMBER 91-SE-PTU-013

DESCRIPTION

Temporarily remove the upper limit on feedwater pH and cation conductivity, if not adding morpholine, while performing a special corrosion product transport study.

The purpose for this change is to test corrosion product transport rates at a different pH.

SAFETY ANALYSIS SUMMARY

The NAPS Nuclear Plant Chemistry Manual limits for feedwater pH under ammonia/boric acid chemistry are 8.8 to 9.2 at 25 degrees C. This limit was originally developed to address corrosion concerns in copper alloy tubes in the feedwater heaters with optimum POWDEX system operation. The copper tubing in the feedwater heaters has been replaced and this phase of the corrosion product transport study does not call for POWDEX operation.

The Westinghouse pH limit is 9.3 to 9.6 for all ferrous feedwater systems. In addition, the EPRI Secondary Water Chemistry Guidelines state that in a boric acid-treated system the pH should be maintained above 8.5. Under these guidelines, operation of the feedwater system with the pH above 9.2 may be permitted for the duration of the test period.

The secondary systems will continue to operate within vendor guidelines as established specifically to address the secondary chemistry design criteria.