Nebraska Public Power District
Nebraska's Energy Leader

NLS2002122
October 17, 2002

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: 10CFR50.59(d)(2) Summary Report
Cooper Nuclear Station
NRC Docket No. 50-298, DPR-46

The purpose of this letter is to provide the summary report of facility/procedure changes, tests, and experiments that have been completed (Attachments 1, 2, and 3), in accordance with the requirements of 10CFR50.59(d)(2). This report covers the time period from August 1, 2000, to July 31, 2002. On July 2, 2001, Cooper Nuclear Station (CNS) implemented the revised 10CFR50.59 rule. Accordingly, each 10CFR50.59 evaluation reported herein has been designated as either "OLD RULE" or "NEW RULE" for informational purposes. Also provided is a summary of commitment changes made to date since the last reporting period (Attachment 4), which have been revised per NEI 99-04, "Guidelines for Managing NRC Commitment Changes."

In accordance with 10CFR50.4, the original report is enclosed for your use, and copies are being transmitted to the Nuclear Regulatory Commission (NRC) Regional Office and the NRC Resident Inspector for CNS.

Should you have any questions concerning this matter, please contact Mr. Paul Fleming at (402) 825-2774.

Sincerely,

Michael T. Coyle
Site Vice President

/wrv
Attachments

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IE47

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Records, w/attachments

ATTACHMENT 1

FACILITY CHANGES

DC 91-121A
(USQE 2001-0013)- OLD RULE

TITLE: Installation of 69 kV Capacitor Bank

DESCRIPTION: This Unreviewed Safety Question Evaluation (USQE) revises a previous 10CFR50.59 evaluation to reflect transfer of the 69 kV line voltage monitoring responsibility for operation of the 69 kV capacitor bank from the Nebraska Public Power District (NPPD) Transmission Control Center to the Cooper Nuclear Station (CNS) Control Room. The revision also evaluates the ability of the 69 kV/4 kV Emergency Station Service Transformer (ESST) to fulfill its required function during anticipated accidents and operational transients.

10CFR50.59
EVALUATION: The 69 kV capacitor bank is by design, a highly reliable and passive component. Installation of the capacitor bank has improved the availability of the ESST, and thus decreased the probability of a total loss of offsite power. Review of accidents and transient events included in the CNS licensing basis determined that the probability of accidents and events where the ESST is assumed to be unavailable, including the Station Blackout (SBO) Event, remained unchanged by this design change. Normal operator actions to monitor grid voltage and communicate with the NPPD Transmission Control Center will assist in maintaining the continued operability of the ESST and associated offsite circuit. Therefore, the probability of an accident previously evaluated has not been increased. Failure of both offsite circuits and subsequent operation on the Diesel Generators is a previously evaluated event in the licensing basis. Failure of the capacitor bank to be available does not impact existing analyses as the low voltage permissive would not allow the ESST to be placed in service. Required operator actions post accident or events remain unchanged by this design. Therefore, there is no increase in the consequences of and accident previously evaluated. The possible functional failures have been evaluated and determined that the probability of equipment malfunction is not increased. Neither the installation of the capacitor bank, nor the operator monitoring of the 69 kV voltage can increase the consequences of an equipment malfunction, or create a new type of accident. Failure (shorting) of the capacitor band pre-accident is prevented from affecting the 69 kV line via fusing. Therefore, the possibility of a different type of malfunction is not created. The ESST continues to meet the Technical Specification bases requirements to maintain rated frequency and voltage. Therefore, there is no reduction in the margin of safety.

DC 95-033-1
OLD RULE

TITLE: Removal of Terminal Box YD125

DESCRIPTION: DC 95-033-1 raised the 120 VAC power supply cabling to the heat trace circuit for the Standby Gas Treatment System (SGTS) piping (North and South) and Elevated Release Point (ERP) drain piping (OG-104) above flood level. DC 95-033 relocated temperature switches RW-TS-Z1 and RW-TS-Z2 above flood level, but terminal box YD125 remained below flood level. This design change removed terminal box YD125 and installed a new cable/conduit using existing "B-line" supports, raising the entire circuit above flood level.

10CFR50.59
EVALUATION: The SGTS heat tracing and its power source are non-essential. No changes to the existing design were made except removal of terminal box YD125, re-routing the associated heat tracing power/cable above the flood level, and removal of an unnecessary receptacle. The heat trace

circuit does not provide any control functions and is not associated with any of the accident initiator discussed in Chapter XIV of the Updated Safety Analysis Report (USAR). The rerouting of cable/conduit reduced the possibility of the heat trace circuit from becoming inoperable due to flooding. The heat trace circuit is not used to mitigate the consequences of an accident or transient described in the SAR. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. Neither the SGTS nor power to the Z-Sump pump was impacted by this change. Therefore, this change did not increase the probability or consequences of a malfunction of equipment important to safety. For the same reasons, this change did not create the possibility of a different kind of accident or malfunction of equipment important to safety, nor result in a reduction of the margin of safety.

EE 96-007
OLD RULE

TITLE: Replacement of Turbine Equipment Cooling (TEC) Piping to Service Air Compressor With Heavier Walled Material

DESCRIPTION: The existing ¾" Schedule 40 TEC piping was experiencing repeated fatigue cracking where it connects to the Service Air System compressor intercoolers. This problem started shortly after the original schedule 80 piping was replaced with schedule 40 piping to all three compressors under DC 90-326 in an effort to decrease compressor outlet temperatures. The cause of these failures was determined to be the use of Schedule 40 pipe in lieu of Schedule 80 pipe as originally installed. This change replaced the ¾" schedule 40 pipe with ¾" Schedule 80 pipe, restoring this interface to its original configuration.

10CFR50.59
EVALUATION: This change was determined not to degrade the reliability, performance, or operation parameters of either the Instrument Air (IA) System or the TEC System. Neither system has been credited in the safety analysis with any accident mitigation functions. No changes were made to any plant procedures, protection features, or fire or flooding barriers. Therefore, this change did not increase the probability or consequences of an accident previously evaluated. The loss of instrument air and its effects upon interfacing equipment has been analyzed and is described in the USAR. This change did not impact the IA System capability, or create an increased potential for operator error. No new failure modes were created for either safety or non-safety related equipment. Accordingly, this change did not increase the probability or consequences of a malfunction of equipment important to safety. This change did not affect the design, function, and performance of the IA or TEC systems, change any operational procedures or alignments, or impact any accident parameters. Therefore, this change did not create the possibility for a new malfunction of equipment or accident. The change did not affect any parameters defined or used in support of those contained in the Technical Specifications; therefore, this change did not reduce the margin of safety defined in the basis for any Technical Specification.

MP 96-081 Change Notice 1
OLD RULE

TITLE: MS-MOV-MO74 and MS-MOV-MO77 Motor Operator Upgrade

DESCRIPTION: This modification upgraded the motor-operators on MS-MOV-MO74 and MS-MOV-MO77 (Main Steam Line Inboard and Outboard Drain Valves). The changes involved replacement of existing Limitorque SMB-000 motor-operators, fitted with 5 ft-lb, 460 VAC, 1800 rpm (nominal) motors with SMB-00 motor-operators, fitted with 10 ft-lb, 460 VAC, 1800 rpm (nominal) motors. The modification also replaced the supplied SMB-00 spring packs with those of higher torque range, replacement of the motor thermal overload heaters, installation of new valve yokes to accommodate the different actuator bolt patterns, and installation of new stem nuts compatible with

the SMB-00. Change Notice #1 removed work associated with RCIC-MOV-MO16 from this modification package.

10CFR50.59
EVALUATION: This modification consisted of replacing existing motor operated valve components with those of like or better performance qualities. The replacement components meet the same design requirements and safety classifications of the previous components. The operating conditions for the interfacing systems were not affected. The control logic was not changed. The modification increased the thrust/torque operating margin of the motor-operated valves (MOVs) thus, ensuring the capability to function as required under accident conditions. Therefore, this change did not increase the probability or consequences of an accident previously evaluated.

The valve components installed exceed the performance requirements of the previous valves. The calculated stroke times remain well below the IST and Technical Specification stroke time requirements. The spring packs have been set to ensure the MOVs produce adequate thrust and torque to perform their function under design basis conditions. The replacement parts have been seismically and environmentally qualified for their intended application. The ability of these MOVs to perform their mitigating functions has not been reduced. The change does not alter the function of any components, equipment, system or structure. Therefore, this change did not increase the probability or consequences of a malfunction of equipment important to safety.

This modification did not alter any operating conditions or introduce any new failure modes. No new demands were created for operations personnel, nor changes required to any operational procedure. The new actuators and the 10 ft-lb motors were evaluated in conjunction with the valve and determined to be acceptable. Therefore, this modification did not create the possibility for a new malfunction of equipment or accident. This modification increased the operating margin for these valves. The stroke times for these valves were increased slightly, but remained well below the Technical Specification limit of 30 seconds. The AC and DC distribution system loads increased slightly due to the increased motor size, but remain within the respective system capacity. Therefore, this change did not reduce the margin of safety defined in the basis for any Technical Specification.

<u>CED 1998-0005</u>
<u>(USQE 1999-0089)- OLD RULE</u>

TITLE: System Tank Volumes G & L Test/Drain Valves Acceptance

DESCRIPTION: CED 1998-0005 creates Component Identification Codes and updates documentation to reflect the existence of two test/drain valves for volumes G & L of the Post-Accident Sampling System (PASS).

10CFR50.59
EVALUATION: These PASS test/drain valves are normally closed, manually operated, and located downstream of primary containment isolation valves. The test/drain valves are used to test/calibrate volumes G & L tank level switches. As such, they do not impact the ability of PASS to perform its safety objective to provide increased accident assessment reliability and operations personnel control capabilities without exceeding individual radiation exposure limits. These valves do not directly or indirectly impact any safety-related equipment. They will not result in initiation or affect mitigation of accidents or transients and are not credited in the analyses thereof. Therefore, this change does not increase the probability or consequences of an accident or malfunction of equipment important to safety. No new failure modes are introduced since this change does not alter the design basis, function, or operation of the PASS, nor alter any protective equipment. Accordingly, an accident or malfunction of a different type has not been created. Finally, since the added test/drain valves do not alter or impact any equipment or operating parameters and do not inhibit the design function or operation of PASS, the margin of safety as defined in the basis of any Technical Specification has not been reduced.

CED 1998-0060
(USQE 1998-0019 Revision 2)- OLD RULE

TITLE: Nitrogen Cushion Installation Into Fire Protection System High Points

DESCRIPTION: As a result of preliminary analysis, it was determined that the Fire Protection system may be susceptible to rupture from a water hammer event, and that the postulated rupture could result in undesirable flooding of plant equipment. Pending the final evaluation results, temporary measures were initially taken under this CED to inject nitrogen into system high points at five locations in the Reactor Building. Injection of the nitrogen serves as a gas cushion that minimizes the transmittance of shock waves that could result in a water hammer event. The nitrogen injection was controlled such that no more than three gallons of water are displaced at any one location. Nitrogen was selected because piping corrosion would be negligible and the effectiveness of the Fire Protection system would not be reduced. Once the results of the evaluation are finalized, permanent corrective actions, if needed, will be implemented.

This temporary modification was reported in the NPPD's October 17, 2000 10CFR50.59 Summary Report. CED 1998-0060, Revision 2, revises the injections points from five to three, and removes the specific Temporary Configuration Change (TCC) completion date from the USQE as this date is subject to change based on monitoring and trending of the nitrogen cushions.

10CFR50.59
EVALUATION: This activity does not affect any accident precursors or initiators and, therefore, does not increase the probability of an accident previously evaluated in the SAR. The inputs and assumptions used in the Nuclear Safety Operational Analysis (NSOA) are unaffected by this activity and the backup water supply to the spent fuel pool is not degraded by the installation of the nitrogen cushions. The activity does not change, degrade, or prevent any equipment or operator actions. Therefore, the consequences of an accident previously evaluated in the SAR have not increased. The postulated damage to the Fire Protection system as a result of a water hammer event is minimized by the nitrogen cushions. Consequently, the probability of a malfunction of equipment important to safety previously evaluated in the SAR has not increased. Since the functionality and reliability of the Fire Protection system is unaffected by the nitrogen cushions, the assumptions made in evaluating the consequences of any malfunction of equipment important to safety have not changed. The inputs and assumptions of the NSOA are unaffected by this activity and no new accident or event initiators have been introduced. Therefore, the possibility for an accident of a different type has not been created. Likewise, the possibility for a different type of equipment malfunction has not been created because no new failure modes for the Fire Protection system have been introduced. Further, the issues of design pressure, corrosion, water hammer, fire brigade operations and system response have been addressed and are bounded by existing analyses. Finally, since the effectiveness of the system is unaffected, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0072 Change Notice 4
(USQE 2000-0007 Revision 1)- OLD RULE

TITLE: Optimum Water Chemistry (OWC) System

DESCRIPTION: This modification installed a system to improve the water chemistry in the Reactor Coolant System (RCS) and the Condensate and Feedwater Systems, to inhibit Intergranular Stress Corrosion Cracking (IGSCC) crack initiation and propagation. The OWC System is designed to generate and inject hydrogen into the reactor condensate/feedwater at the suction of the condensate Booster Pumps. Hydrogen injection suppresses the radiolytic formation of hydrogen and oxygen in the core, reducing the concentration of oxygen, hydrogen peroxide and other oxidizing species in the reactor coolant. Outside the core, the hydrogen recombines with residual hydrogen peroxide and oxygen to form water, changing the water chemistry from an oxidizing to a reducing environment.

Because the hydrogen injection results in a reduction in the radiolytic formation of oxygen, the OWC System also injects oxygen upstream of the Augmented Off Gas System hydrogen to restore the stoichiometrically correct ratio of oxygen to hydrogen to ensure recombination of the hydrogen contained in the offers effluent.

The OWC System consists of the Electrolytic Generation System, the Hydrogen Injection Module, the Oxygen/Air Injection Module, the Condensate Oxygen Injection Module, and the Main Control Panel. The Electrolytic Generation System consists of a Gas Generation Skid, Hydrogen Compression and Purification Skid, Oxygen Compression and Purification skid, Rectifier Bank, Analyzer Panel, Potassium Hydroxide Makeup Tank, the motor control center and the Gas Generation Control Panel. All the skids for the Electrolytic Generation System and the Main Control Panel were installed in the OWC Gas Generation Building, installed under CED 1999-0074, and located just north of the Turbine Building. The injection modules were installed in the north Turbine Building mezzanine.

Revision 1 to this USQE incorporated the following information resulting from Change Notice #4:

- Documented that on an OWC system shutdown, the system will automatically shutdown and that immediate Operator action is not required. Also, added discussion on classification of OWC System trip functions.
- Justified why a hydrogen leak in the Turbine Building will not create a potentially explosive environment.
- Added additional support references from GE on the OWC system design.
- Deleted calculation information that is no longer applicable due to a change in the actual installation.

10CFR50.59
EVALUATION: OWC is a proven technology for the prevention of IGSCC crack initiation and growth in the Reactor Coolant System. Prevention of IGSCC reduces the possibility of a large or small break Loss of Coolant Accident (LOCA). The injection of hydrogen into the Condensate System does not adversely affect the Condensate System, Feedwater System, Reactor Recirculation System or the vessel and vessel components. There is no increase in the probability of Reactor Recirculation System Pump seizure. The OWC System also injects oxygen into the condensate system, maintaining the dissolved oxygen concentration within the fuel warranty limits, thus ensuring the integrity of the fuel bundles are not adversely impacted. There are no adverse impacts on the Control Rod Drive Mechanism. Accordingly, the probability of a Control Rod Drop Accident (CRDA) is not increased. Oxygen concentration in the RCS will remain sufficiently high to prevent flow accelerated corrosion; therefore, the probability of a Main Steam Line Break Accident (MSLBA) is not increased. The OWC does not interface with any essential structure, system, or component with the exception of the control and indication circuits; therefore, the installation of this system does not impact accident initiators, and will not increase the possibility of an accident previously evaluated.

The OWC does not adversely affect any equipment or affect accident mitigation assumptions. The increase in main steam line radiation does not impact the main steam line radiation monitor alarm setpoints, which are based on the CRDA. The CNS containment is inerted during operation and operators have the capability to manually provide makeup nitrogen. Further, the hydrogen generation and injection and systems are designed to automatically trip on reactor low power, and loss of power. Evaluation has shown that hydrogen injection for up to 60 minutes following a reactor scram and determined to be acceptable. Therefore, the consequences of an accident are not increased.

Design provisions minimize the possibility for hydrogen leakage in the system, and generation equipment is located in a dedicated building. Hydrogen monitors are located in areas where leakage is possible to alert operators should leakage occur. Evaluation of these areas has

determined that any leakage would disperse and not concentrate. Therefore the risk of a fire or explosion has not been increased.

For the reasons stated above, neither the injection of hydrogen and oxygen into the Condensate System, nor the interfaces of the OWC System with plant equipment will increase the probability or consequences of equipment malfunction, nor create the possibility of a new accident or malfunction, or reduce the margin of safety.

CED 1999-0081 Change Notice 4
(USQE 1999-0062 Revision 1)- OLD RULE

TITLE: Mitigation Monitoring System (MMS) Installation, CED Change Notice Number 4

DESCRIPTION: CED 1999-0081, Change Notice #4 eliminates the 6 gpm limit for total flow to the MMS panel during "at power" operation when the Reactor Water Cleanup (RWCU) sub-cooling line is isolated. The limit was an arbitrary/estimated number. There are no safety concerns tied to this limit. Removing the limit adds operational flexibility with a negligible impact on RWCU performance. The revision also clarifies the MMS panel may remain in service during NobleChem applications (a process wherein noble metals are injected into the reactor coolant system to assist in cracking mitigation).

10CFR50.59
EVALUATION: The Electrochemical Potential Mitigation Monitoring System (ECP/MMS) is a nonessential passive monitoring system. The installation is outboard of the containment isolation valves, in the nonessential portion of the RWCU System. Therefore, the probability of an accident previously evaluated in the SAR has not increased. The ECP/MMS has no effect on any equipment important to safety necessary for the mitigation of any accident, transient, or special event. Thus, the consequences of an accident or malfunction previously evaluated in the SAR have not increased. The ECP/MMS is used strictly for data acquisition and analysis. With the exception of the diversion of an insignificant RWCU system flow and a very minor increase in diesel load (which was evaluated as acceptable), there is no change in any system load or operating condition. As a result, the probability of a malfunction of equipment important to safety previously evaluated in the SAR has not increased. Other than the flow through the unit, the ECP/MMS has no impact on the plant or plant operations (i.e., no new safety-related equipment interaction, accident scenario, or sequence of events will be created by the installation of the system). Accordingly, the possibility of an accident of a different type than any previously evaluated in the SAR has not been created. The power supply to the electrical and electronic portion of the monitoring system is nonessential. Further, any failure of internal mechanical components of the monitoring system would be contained within the RWCU system, and leaks can be readily isolated by isolating MMS or the RWCU system. Consequently, the possibility of a different type of malfunction than any previously evaluated in the SAR has not been created. The system has no impact on any Technical Specification limiting condition for operation, surveillance or bases. Therefore, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0100
(USQE 2000-0046)- OLD RULE

TITLE: Configuration Evaluation (Electrical) for Various Items Associated with the Unauthorized Modification Follow-up Project

DESCRIPTION: CED 1999-0100 evaluated ten items identified during the CNS Unauthorized Modification Follow-up Project determined to require evaluations under 10 CFR 50.59. These included:

1. The identification of mispositioned 120 VAC lighting panel breakers in the Administration Building. Spare breakers on several 120 VAC panels were listed as SPARE per Procedure

2.2.90A, but identified as CLOSED/ON.

2. A list of breakers located outside the protected area identified as out-of position.
3. 240V/120V Mobile Skid D was not located as specified in Procedure 2.2.90A.
4. Incorrect breaker load designations and missing 12.5 kV disconnects located outside the protected area.
5. The identification of spare breakers on several 120 VAC panels listed as SPARE per Procedure 2.2.90A, but identified as CLOSED/ON.
6. Incorrect 12.5 kV disconnect identification outside the protected area.
7. Mispositioned breaker located in the Electric Shop in the Turbine Building.
8. Incorrect identification of 120 VAC lighting breakers located inside the Machine Shop/Cold Chemistry Lab Area/Mechanics Lunchroom.
9. The installation of a 10 kVA transformer, two 60 amp disconnect switches, and two power outlets in the Office Building 3rd Floor corridor under a Maintenance Work Request.

10CFR50.59
EVALUATION: Item 1 These breaker mispositionings were associated with several panels. One of these was determined to be an unauthorized modification requiring a USQE; this was the breaker mispositioning associated with Panel EE-PNL-LPTSC, breaker #23. Panel EE-PNL-LPTSC is powered by the non-essential 12.5 kV ring bus system and supplies nonessential loads, e.g., receptacle and lighting circuits. Emergency power is supplied by EE-MCC-CA. The emergency power loading was unaffected since associated calculations assume full transformer loading. Therefore, this unauthorized modification was determined not to involve an unreviewed safety question.

Item 2 One of these mispositioned breakers was determined to be an unauthorized modification. This was associated with the addition of a receptacle in the Learning Center hallway. The loads associated with these panels are nonessential, e.g., lighting and receptacles, and are not located within a seismic class IS structure. The affected panel is supplied by the nonessential 12.5 kV ring bus system, and does not affect Diesel Generator or Emergency Transformer loadings. Therefore, this unauthorized modification was determined not to involve an unreviewed safety question.

Item 3 240V/120V Mobile Skid D was located in the Training Center North Parking Lot providing temporary power for trailers. The loads associated with these panels were nonessential, and are not located within a seismic class 1S structure. The mobile skid was supplied by the nonessential 12.5 kV ring bus system, and did not affect Diesel Generator or Emergency Transformer loadings. Therefore, this unauthorized modification was determined not to involve an unreviewed safety question.

Item 4 One item identified in this group was determined to be an unauthorized modification. This was a discrepancy associated with the designation of loads on 120 VAC panel 345kV SWYD Control House Panel LC-1 Ckt #34. These circuits are located in the 345 kV switchyard and do not support any safety related functions. These circuits do not interface with any safety-related power sources and the panel is not located in a Class 1S structure. Therefore, this unauthorized modification was determined not to involve an unreviewed safety question.

Item 5 The mispositioned spare breakers identified were associated with panel OPA-PNL-LC5, located in the West Warehouse. The loads associated with these panels are nonessential, e.g., lighting and receptacles, and are not located within a seismic class IS structure. The affected panel is supplied by the nonessential 12.5 kV ring bus system, and does not affect Diesel Generator or Emergency Transformer loadings. Therefore, this unauthorized modification was determined not to involve an unreviewed safety question.

Item 6 Procedure 2.2.90A contained an entry for disconnect switch 58D for the "CLOSED" position. Disconnect 58D was supposed to be an external 12.5 kV disconnect used to isolate the Low Level Radwaste Area equipment, located southwest of the West Warehouse, from the 12.5 kV

distribution system. The transformer at the location of disconnect 58D was hard-wired to the 12.5 kV line with no visible disconnect. Review of the installed configuration determined that the installed transformer uses an internal disconnect to protect and isolate the transformer. Therefore the external disconnect is not required. The power supply does not interface with the normal or emergency AC standby power system. The 12.5 kV ring bus system is nonessential and not required to function under accident conditions. The affected loads in the Low Level Radwaste Storage Facility are not used to limit radiation exposure or control contamination. Malfunction of the affected equipment cannot impact an existing or create a new accident. Therefore, this unauthorized modification was determined not to involve an unreviewed safety question.

Item 7 The mispositioned breaker located in the Electric Shop, EE-PNL-SPTG2, Breaker 13, was listed as "SPARE" on the applicable drawing, but was found supplying a microwave oven. The receptacle was not shown on the applicable drawing, and did not match the associated procedure. Panel EE-PNL-SPTG2 is a non-essential 120/208 VAC panel located in the Turbine Building. The panel is supplied by a non-essential MCC, and does not supply any safety-related loads, and does not affect Diesel Generator or Emergency Transformer loadings. This modification was determined not to involve an unreviewed safety question.

Item 8 This item involved Panel EE-PNL-LPTG15 breaker load discrepancies between the applicable drawing and procedure. Panel EE-PNL-LPTG15 is located inside the Machine Shop/Cold Chemistry Lab Area/Mechanics Lunchroom. Review of these load discrepancies determined that they do not impact any accident or event described in the SAR. This unauthorized modification was determined not to involve an unreviewed safety question.

Item 9 This item involved the installation of a 10 kVA transformer, two 60-amp disconnect switches and two power outlets in the Office Building. The 120V receptacle was being used to power a copier, the 240V receptacle was not being used. The affected components are located in the Control Building, are not located near any safety-related equipment, and do not interface with the controls for any safety-related equipment. The local receptacles are protected by local disconnects, and the primary sided by a fused disconnect at MCC W. The addition of these loads to MCC W was determined not to impact the ability of MCC W to supply its loads. The fusing was determined to be appropriately sized to protect MCC W from postulated faults. Therefore, this unauthorized modification was determined not to involve an unreviewed safety question.

CED 2000-0128
(USQE 2000-0021)- OLD RULE

TITLE: Monitoring for Voltage Spikes on the 125 VDC System

DESCRIPTION: This Temporary Change Configuration (TCC) installed voltage monitoring equipment and a recorder at the 125 VDC High Pressure Coolant Injection (HPCI) transfer switch to collect data as part of investigations concerning the failure of associated indicating lights. The chart recorder was installed into EE-SW-125HPCI, classified as EQ. The scope of this TCC also included the Reactor Core Isolation Cooling (RCIC) vacuum pump, and the RCIC condensate pump. However, this part of the TCC was not implemented.

10CFR50.59
EVALUATION: The chart recorders are classified non-essential, non-seismically qualified components. The HPCI and RCIC systems are not accident initiators. This TCC was determined to have no impact on the operation of the HPCI or RCIC systems, and would not prevent them from performing their design basis functions. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. CED 2000-0128 cannot prevent HPCI or RCIC from operating when required, or impact any other equipment or components. Therefore, this change does not increase the probability or consequences of a malfunction of equipment important to safety. For

the same reasons, this change does not create the possibility for a new malfunction of equipment or accident, or reduce the margin of safety.

CED 2000-0162
(USQE 2000-0031)- OLD RULE

TITLE: Governor Valve Number 2 Close Limit Switch Jumper

DESCRIPTION: CED 2000-0162 installed a jumper to simulate a Main Turbine Governor Valve No. 2 close position, due to an inoperative position limit switch. The closed limit switch is interlocked to the Main Turbine Stop Valve No. 2 test solenoid valve. Installation of this jumper provides a necessary part of the logic makeup to enable energizing the test solenoid, which is required to close Main Turbine Stop Valve No. 2. This was performed as a temporary measure to support quarterly Main Turbine Stop Valve Closure and Steam Valve Functional testing required by the Technical Specifications, until the Main Turbine Governor Valve No. 2 limit switch could be repaired or replaced.

10CFR50.59
EVALUATION: The only function of the open or closed limit switch on the governor valve is to provide valve position indication in the control room and an interlock contact to the stop valve test solenoid circuit. The limit switches on the governor valves do not provide any safety-related function and are not discussed in the USAR. The jumper was installed in the same manner and is of the same material type and gage as existing circuit wiring. The possibility of the jumper failing and the odds of failure are the same as existing circuitry wiring. The activity did not impact or alter the function of the Digital Electro-Hydraulic (DEH) pressure control loop, turbine trip circuit or any safety-related system. Therefore, this change did not increase the probability or consequences of an accident previously evaluated, or increase the probability or consequences of a malfunction of equipment important to safety. This jumper interfaced with the main turbine stop valve test circuit only. The main turbine trip circuit and DEH system cannot initiate an accident that is not bounded by existing USAR analyses. Therefore, this change did not create the possibility for a new malfunction of equipment or accident. This change did not reduce the margin of safety defined in the basis for any Technical Specification.

CED 2000-0167
(USQE 2000-0041)- OLD RULE

TITLE: Service Water Pump Room Emergency Light Addition

DESCRIPTION: CED 2000-0167 installed an Appendix R Emergency Battery Light (EBL) in the Service Water Pump (SWP) Room vestibule. The EBL was installed to meet the requirements of 10CFR50, Appendix R, Section III.J.

10CFR50.59
EVALUATION: This design change was determined not to affect initiators of any accident. Loadings on the auxiliary electrical system were determined to be acceptable. The only event that the EBL is required to function is an Appendix R fire causing the loss of all plant lighting. This modification adds one EBL unit to the SWP Room vestibule, which will enhance operator access/egress during an Appendix R fire. The EBL unit is nonessential. Therefore, this change did not increase the probability or consequences of an accident previously evaluated. The EBL was mounted in accordance with Class I seismic requirements. The associated electrical load calculation was revised to reflect the negligible increase in electrical load. Accordingly, this change did not increase the probability or consequences of a malfunction of equipment important to safety. Failure of the EBL unit cannot initiate any new accident. Loss of non-essential lighting has previously been evaluated for CNS in accordance with 10CFR50, Appendix R requirements. This modification did not adversely affect existing circuitry or lighting levels. Therefore, this change

did not create the possibility for a new malfunction of equipment or accident, or reduce the margin of safety defined in the basis for any Technical Specification.

CED 2000-0178
(USQE 2000-0033)- OLD RULE

TITLE: Bypass Well Water Supply to Water Treatment

DESCRIPTION: This modification provided a temporary bypass from existing well water piping to the Water Treatment Building, via use of fire hoses and associated connections. This modification enabled re-establishing well water service until the leak could be isolated and repaired.

10CFR50.59
EVALUATION: The loss of well water to the Makeup Water Treatment System is not an accident initiator. The fire hose and connections met and exceeded the design requirements with respect to temperature, pressure and testing as specified in the original piping contract. The temporary bypass did not cause the well water piping system to operate outside of the design or testing limits for supplying well water to the Makeup Water Treatment System. Neither the Well Water System nor the Makeup Water Treatment System is required to mitigate the consequences of an accident. The Well Water System has no support function or interface with any radiological structure, system, or component (SSC). Therefore, this change did not increase the probability or consequences of an accident previously evaluated. CED 2000-0178 installation did not degrade the performance of or increase any challenges to a safety system because the Well Water System has no support function or interface with any safety related system. These components are passive components that do not perform any safety function. The failure of the connection flanges, fire protection valve, hose and/or adapters is no worse than the failure of the underground piping. Therefore, this change did not increase the probability or consequences of a malfunction of equipment important to safety, or create the possibility for a new malfunction of equipment or accident. There is no margin of safety applicable to this system.

CED 2000-0188
(USQE 2001-0002)- OLD RULE

TITLE: In-line Instrument Snubber for Gauges RWCU-PI-73A and RWCU-PI-73B

DESCRIPTION: RWCU-PI-73A, B, Reactor Water Cleanup (RWCU) Filter Demineralizer Effluent Pressure Indicators, have experienced repeated failures. These failures were caused by repeated pressure transients associated with cycling RWCU Filter Demineralizers in and out of service to perform washing and pre-coating operations. The high pressure differential (60 psig – 1000 psig) caused pressure transients resulting in repeated failure of RWCU-PI-73A, B. CED 2000-0188 installed in line pressure snubbers to dampen these pressure transients.

10CFR50.59
EVALUATION: RWCU-PI-73A, B are used for local indication, and are not relied upon to perform any safety function. The installation of these pressure snubbers will prolong the life of the pressure indicators without affecting their functions. The RWCU system does not perform any safety-related function. The installed snubbers exceed the pressure requirements of the RWCU System. Accident dose consequences are not affected by this modification. Therefore, this change did not increase the probability or consequences of an accident previously evaluated. RWCU-PI-73A, B are not relied upon to perform any Emergency Operating Procedure steps. The failure of these components will not affect the operation of any safety-related equipment. Therefore, this change did not increase the probability or consequences of a malfunction of equipment important to safety. The RWCU is evaluated for a pipe break is designed with sufficient instrumentation and controls to detect and initiate isolation of the RWCU line as necessary. RWCU-PI-73A, B do not provide any isolation signals, or create any new accident initiators. Failure of RWCU-PI-73A, B will prevent operators from accurately reading the filter demineralizer effluent pressure. The pressure indicators cannot

fail in a new manner to create a new malfunction that could affect safety. Therefore, this change did not create the possibility for a new malfunction of equipment or accident. The installation of these pressure indicators does not affect the margin of safety defined in the basis for any Technical Specification.

CED 2000-0192
(USQE 2000-0044)- OLD RULE

TITLE: Security System Upgrade

DESCRIPTION: CED 2000-0192 added a new, dedicated non-essential transformer, Uninterruptible Power Supply (UPS), and static switch, fed from the Division II essential bus to power the Security Alarm Station (SAS) loads. The static switch provides for automatic switching to the No Break Power Panel (NBPP) alternate AC power source on battery low voltage.

The SAS loads were previously fed from the Central Alarm Station (CAS) distribution panel via EE-MCC-K (Division I). CED 1999-0052 implemented a major upgrade to the CNS Site Security System. During development of CED 1999-0052, it was determined that a new power source was needed to avert exceeding the capacity of the existing CAS transformer and UPS, and prevent excessive voltage drops. Under CED 2000-0160, the SAS loads were temporarily powered from the Plant Management Information System (PMIS) UPS distribution panel. CED 2000-0192 and associated procedure changes provide a dedicated power source fed from EE-MCC-T (Division II), and automatic switching to the NBPP upon battery low voltage.

10CFR50.59
EVALUATION: CED 2000-0192 and associated procedure changes installed a dedicated power source for the SAS loads, consisting of a new inverter, UPS, and static transfer switch for automatic switching to an alternate AC power source on battery low voltage. The SAS is non-essential, but was evaluated and installed to comply with appropriate seismic criteria. Work control procedures used to install the modification ensured that the installation did not affect operable safety-related equipment. The Control Room envelope was restored following work requiring breech of this barrier. The additional loading on EE-MCC-T has been evaluated and determined to be acceptable. The cabling to the essential disconnect switch and to the load side of the second set of fuses is essential, ensuring isolation between essential and non-essential equipment. Therefore, this modification did not increase the probability or consequences of an accident or malfunction of equipment previously described in the SAR. The loss of SAS is not an accident addressed in the SAR. Electrical isolation and seismic mounting of the new equipment from the essential bus prevents the introduction of any new kind of accident or malfunction of equipment. Electrical, heat, and fire loadings were evaluated and determined to be acceptable. Therefore, the margin of safety as defined in the CNS Technical Specifications has not been reduced.

CED 2000-0201 and Change Notice 1
(USQE 2000-0043 Revision 1) – OLD RULE

TITLE: CD-FE-76 Leak Repair

DESCRIPTION: CED 2000-0201 was a Temporary Configuration Change to perform a leak repair around the flange/orifice associated with flow element CD-FE-76 located in the moisture separator drain line.

10CFR50.59
EVALUATION: CED 2000-0201 installed a temporary leak repair clamp around Moisture Separator "A" drain line flow orifice flanges and injected an approved sealing compound as needed. The sealant material and injection limit of 54 cubic inches of uncompressed volume were selected to ensure that the reactor chemistry requirements of Technical Requirements Manual T3.4.1 were not exceeded, even if all of the sealant entered the main condensate system. The limitation of sealant also ensured that excessive sealant was not injected causing a flow obstruction. Evaluation determined

that the installation of the leak repair clamp and sealant would not adversely affect the operation of the moisture separator drain line, flow orifice, or feedwater heaters. The most likely failure mode of the modification would be the failure to maintain a leak tight pressure boundary around the affected flange, allowing the leak to continue. Failure of this leak repair clamp would not aggravate or increase the leak, nor would it affect any other equipment important to safety in the area. Both of these failure modes would be fairly slow, and bounded by existing USAR analyses. Therefore, it was determined that the change did not increase the probability or consequences of an accident previously evaluated, increase the probability or consequences of a malfunction of equipment important to safety, or create the possibility for a new malfunction of equipment or accident. The change also did not reduce the margin of safety defined in the basis for any Technical Specification. Accordingly, this change did not involve an unreviewed safety question.

CED 2001-0007 and Change Notice 2
(Evaluation 2001-0041 Revision 1) – NEW RULE

TITLE: Control Room Emergency Filtration System (CREFS) Upgrade

DESCRIPTION: This CED was developed to incorporate several design changes necessary to validate revised dose assessment calculations for the new fuel currently in use at CNS. Two of the six calculations submitted for NRC approval included assumptions based on CREFS design changes. The changes made by this CED are:

1. The automatic initiation signal for CREFS is changed from high radiation sensed from monitor RMV-RM-1 in the Control Room ventilation intake duct, to a signal based on a Group 2 or 6 isolation signal. Prior NRC approval was required for this change.

2. RMV-RM-1 will be removed and replaced with a two-channel Continuous Air Monitor (CAM) in the Control Room. NRC approval of Item 1 needed prior to implementation of this change. Change Notice 2 installs a temporary one-channel CAM to provide Control Room atmospheric monitoring between the time that RMV-RM-1 is removed and the permanent two-channel CAM is installed.

3. The stroke times for CREFS isolation valves HV-AOV-270AV and HV-AOV-272AV are being reduced by speeding up their operating time.

4. The control switch for the CREFS Emergency Booster Fan (1-BF-C-1A) has been changed from a two-position switch to a three-position switch, that will allow for an OFF position.

10CFR50.59
EVALUATION: This evaluation is predicated on NRC approval of Item 1. The changes associated with this CED are not associated with any of the accident initiators described in the USAR, and thus, do not increase the likelihood of an accident. The two-channel CAM unit will be installed utilizing Seismic II/I design requirements and will be powered via a safety-related circuit breaker. The AOV stroke times will continue to be monitored within the Inservice Testing Program. Therefore, there is no increased likelihood of a malfunction of equipment important to safety. This CED assures that CREFS will operate in conformance with assumptions of the Design Basis Accident (DBA) dose calculations, and the new two-channel CAM unit will monitor Control Room air the same as other CAM units required for Emergency Response Facilities. Therefore, there are no increased accident consequences or consequences of safety equipment malfunctions. The changes per this CED will not introduce new failure modes to CREFS or other types of malfunctions. Therefore, this CED will not create the possibility of new accidents or safety equipment malfunctions that have not previously been analyzed. This CED does not change any fission product barrier parameter, or result in a departure from USAR-described methodologies.

CED 2001-0011
(USQE 2001-0008)– OLD RULE

TITLE: Installation of GrayBoot Connectors or Raychem EQ Splices

DESCRIPTION: This CED has been implemented to replace Weidmuller SAK-6N terminal blocks with GrayBoot Connectors or Raychem splices. GrayBoot Connectors and Raychem splices are environmentally qualified for use in the drywell, as documented in CNS EQDPs. This modification will be performed for the following terminal box numbers: 103, 105, 107, 109, 111, 113, 115, 117, 124, 145, 147, 149, 151, 153, 155, 157, 159, 161, and 179.

10CFR50.59
EVALUATION: This modification activity replaced existing terminal blocks with an electrical splice or connector that meet EQ requirements. Both terminal blocks or connectors perform the same electrical function and therefore, this change will not adversely affect the safe operation of the plant. This evaluation concluded that there were no increased probabilities or consequences of previously evaluated accidents or safety equipment malfunctions, no increased likelihood of new accidents or equipment malfunctions, and no safety margin decreases defined in the bases of the Technical Specifications.

CED 2001-0017 and Change Notice 14
(Evaluation 2001-0043 Revision 2)– NEW RULE

TITLE: Reactor Recirculation Flow Control (RRFC) System Controllers Modification CED 2001-0017

DESCRIPTION: The following upgrades are made to the RRFC System: 1) replace the existing RRFC system GE/MAC analog equipment with Siemens Moore 353A Digital Controllers, 2) remove the Master Controller and Speed Limiter (the speed limiting function is now part of the new controller), 3) remove the Scoop Tube Lockout Reset Permissive light (lights will be relocated to the new controller), 4) new MV/I converters will be provided for the tachometer and scoop tube position feedback signal, 5) install new reactor recirculation pump speed and demand indicators, and 6) voltage regulating transformers are being added to the Siemens Moore controllers and MV/I converters. While this modification does make physical changes to the existing Reactor Recirculation Flow Control equipment, the current system functions and performance requirements are unaltered.

10CFR50.59
EVALUATION: Since the new RRFC system components are more reliable than the existing components and no new failure mode effects are introduced, the proposed activity does not result in an increase in the frequency of occurrence of an accident previously evaluated in the USAR. The new equipment being installed will not initiate any new malfunctions. The RRFC system is neither safety-related nor does this activity adversely affect systems important to safety. Therefore, there will be no increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the USAR. Performance requirements affecting the core coolant flow decrease and increase events are unaltered such heat fuel integrity will be maintained and the USAR analysis of radiological consequences remains bounding. The RRFC system is not used to mitigate the consequences of any accidents and the new equipment will not initiate any new accidents. This modification will not impair or prevent the ECCS from mitigating the consequences of any design basis accident. Therefore, this activity does not increase the consequences of occurrence of an accident previously evaluated in the USAR. Failure or malfunction of the new equipment will not prevent or affect the ability of safety-related systems or systems important to safety to respond to the accidents described in the USAR. Therefore, consequences of a malfunction of an SSC important to safety previously evaluated in the USAR will not be increased. Based on a Failure Modes and Effects Analysis, the potential RRFC malfunctions are already analyzed in the USAR; therefore, the possibility of an unanalyzed malfunction of an SSC important to safety or a new type of accident is not created. As described in the USAR Chapter XIV transient analysis, no

malfunction of the RRFC system can cause a transient sufficient to damage the fuel barrier or exceed the nuclear system pressure limits as required by the safety design basis. Therefore, the CED does not result in the design basis limits for fission product barriers being exceeded or altered. The modification does not result in a departure from a method of evaluation described in the USAR in establishing the design bases or in the safety analysis.

CED 2001-0019
(USQE 2001-0006)– OLD RULE

TITLE: Elimination of Pressure Locking on RHR-MOV-MO39A, RHR-MOV-MO39B, and HPCI-MOV-MO58

DESCRIPTION: This modification drills a small hole in a disc of RHR-MOV-MO39A, RHR-MOV-MO39B, and HPCI-MOV-MO58. This hole will eliminate the potential of pressure locking within the valves as described in Generic Letter 95-07. The hole is being drilled through the disc on the torus side of the valves.

10CFR50.59
EVALUATION: The subject valves are not accident initiators; therefore, this modification will not increase the probability of an accident previously evaluated in the SAR. The possible amount of Low Pressure Coolant Injection (LPCI) flow diversion resulting from this modification is minimal, as is the potential for increased leakage from these valves during a DBA. Therefore, the consequences of an accident are not increased. There are no increased probabilities or consequences of malfunctions of equipment important to safety resulting from this modification. The modification will not affect the configurations or characteristics of the Residual Heat Removal RHR or High Pressure Coolant Injection (HPCI) Systems and the subject valves will still perform their design basis functions. Accordingly, there is no new likelihood of a new malfunction or accident. This modification does not decrease any existing safety margins inherent with the Technical Specifications.

EE 01-023
OLCR 2002-015
UCR 2002-029
Procedure 2.2.9 (Revision 52)
(Evaluation 2002-0008)- NEW RULE

TITLE: Implementation of NLS2001064 (Amendment 192) via EE 01-023

DESCRIPTION: A License Amendment Request (LAR) was submitted in NLS2001064, "Net Positive Suction Head (NPSH) for ECCS Pumps." This LAR was approved by the NRC in Amendment 192 and now requires the proper incorporation into CNS licensing and design basis documentation. CNS procedures require a full 10CFR50.59 evaluation to accompany Technical Specification Bases changes. This activity incorporates a spatial evaluation that addresses the potential for steam ingestion during concurrent Safety Relief Valve and ECCS pump operations. The new methodology vacates the local suppression pool temperature limitation referenced in Section 3.6.2.1 of the Technical Specification Basis, resulting in deletion. This 10CFR50.59 Evaluation also covers a change to the USAR to update HPCI NPSH discussions relative to the current design calculations of record. Additionally, this evaluation covers a change to caution statement in Procedure 2.2.9 addressing the existing throttling of Core Spray pump flow for NPSH concerns rather than for diesel fuel oil consumption concerns.

10CFR50.59
EVALUATION: The proposed changes to the TS Bases as a result of EE 01-023 involve reference changes and clarifications which are supported by License Amendment 192. The changes to the USAR regarding HPCI continue to support the conclusion that adequate NPSH is available when HPCI is called upon to perform its safety function. The procedure change is only recharacterizing the basis

for the need for Core Spray flow throttling. Accordingly, the probability of an accident or transient previously evaluated in the USAR has not been increased. These activities do not change plant parameters, create new system interfaces, alter plant equipment, or result in new or different plant operational steps, sequences or modes; therefore, the probability of a previously evaluated occurrence of a malfunction of an SSC important to safety has not been increased. Likewise, the consequences of previously evaluated malfunctions of SSC's important to safety are not increased. The analyses performed in support of this activity conservatively demonstrate that the consequences of a previously evaluated accident or transient have not been increased, and that the validity of any existing assumptions used in the evaluations referenced in Chapter XIV of the USAR remain unchanged. Therefore, the consequences of previously evaluated accidents or transients have not been increased. The possibility of a new or different type of accident or transient than previously analyzed in the USAR is not created by this activity, nor has the possibility of an SSC malfunction of a different type from that previously analyzed in the USAR been created. The design basis limits for fission product barriers have not been affected by this activity. Since the NRC approved use of the methodologies to quantify the Containment overpressure being credited and to eliminate the local suppression pool temperature limit in License Amendment 192, this activity does not constitute a departure from accepted methodology.

CED 2001-0028 Change Notice 2
(Evaluation 2001-0036)– NEW RULE

TITLE: Revision of Programs to Include Residual Heat Removal (RHR) Containment Spray Mode Components

DESCRIPTION: The Containment Spray mode of RHR was approved by the Atomic Energy Commission in the CNS Safety Evaluation Report of 2/14/73. In 1989, it was determined that Containment Spray was not needed for a DBA LOCA. This led to certain portions of the Containment Spray subsystem being excluded from programs, such as the Motor-Operated Valve Program, which are required for essential equipment. Since this time, it has been determined that the Containment Spray mode has a safety-related function in mitigating a small steamline break inside Primary Containment. CED 2001-0028 upgrades plant programs for improved reliability of Containment Spray components. These program changes will assure the Containment Spray SSCs will perform their safety functions. This will allow crediting Containment Spray in a separate Engineering Evaluation and 10CFR50.59 Evaluation.

10CFR50.59
EVALUATION: The Containment Spray mode of RHR is not used during normal plant operation and is operated only after a LOCA inside the containment has occurred, and therefore, is not an accident initiator as identified in USAR Chapter XIV and Appendix G. There are no physical modifications required to the Containment Spray components or to any other components; therefore, there are no physical changes that can cause a change in adverse system interactions involved. Thus, this change does not result in more than a minimal increase in the likelihood of occurrence or the consequences of a malfunction of an SSC important to safety previously evaluated in the SAR. There are no changes to the manner in which the RHR system is tested or operated during normal plant operation as a result of this CED except to add in stroke time testing of the Containment Spray valves in the opening direction. Therefore, the incorporation of the Containment Spray mode of RHR into the programs required for safety related function does not result in more than a minimal increase in the frequency or consequences of an accident previously considered in the SAR. The change in program status by adding RHR Containment Spray components has minimal impact on the possibility for creating an accident or a malfunction of a different type than previously evaluated in the USAR. The change does not involve a physical alteration of the plant, or cause installed equipment to be operated in a new or different manner. Thus, the change does not result in a design basis limit for a fission product barrier as described in the USAR from being exceeded or altered. There are no analyses involved in this CED, so there can be no departures from a method of evaluation.

EE 01-035
OLCR 2001-029
TRMCR 2001-008
(Evaluation 2001-0044)- NEW RULE

TITLE: EQ Temperature Profile in Containment Based on Small Steam Line Break and DBA-LOCA

DESCRIPTION: EE 01-035 implements calculations which: a) define the Primary Containment long term response to a DBA LOCA and to a spectrum of Small Steamline Breaks (SSLB), b) provides a composite drywell temperature curve to be utilized for equipment qualification, and c) determines the RHR tube plugging limit based on the minimum Containment Spray flow. The long term Containment response to a DBA LOCA was reanalyzed for the Suppression Pool Cooling mode of RHR, and for the Containment Spray Cooling mode consistent with USAR Case E. The SSLB was utilized to determine the maximum drywell temperatures expected post-accident. Based on the results, the Containment Spray mode of RHR is considered a necessary safety-related function due to the need to mitigate drywell temperature effects during a SSLB. The results of this Engineering Evaluation also resulted in a change to the Technical Specification Bases and the Technical Requirements Manual.

10CFR50.59
EVALUATION: EE 01-035 does not result in any physical plant changes and the use of containment spray is already considered as an alternate means of containment cooling. Use of containment spray is not an accident initiator, and therefore, does not increase the frequency of occurrence of an accident previously evaluated in the USAR. Similarly, it does not create a possibility for an accident of a different type. Procedures and operator training are in place for utilizing containment spray for cooling the containment. The 10 minute operator response time for containment cooling is already contained in the USAR and sensitivity studies show that operator response time could be extended to 30 minutes or more without exceeding the containment design temperature. Containment Spray has always been considered an important operating mode and is designed for single failure. CED 2001-0028 ensured that the design requirements that will allow crediting the Containment Spray mode of containment cooling in the accident analyses are met. Therefore, use of Containment Spray for mitigation of a SSLB does not increase the likelihood of occurrence of a malfunction, nor does it create a possibility for a malfunction of an SSC important to safety with a different result. The use of SHEX and HXSIZ computer codes for analyzing the containment response for SSLB and DBA-LOCA has been benchmarked against an original Final Safety Analysis Report case and determined to yield similar or conservative results. These are both industry and NRC accepted computer codes. Plant parameter changes are conservative and do not deviate from methodology required values. Use of the SHEX code is already described in the USAR. The HXSIZ code is only utilized for long term EQ qualification profiles beyond the time frames utilized for containment response reflected in the USAR. HXSIZ was an industry accepted code long before SHEX and meets the requirements of NUREG-0588 for EQ analyses. Therefore, EE01-035 does not result in a departure from a method of evaluation described in the USAR. The use of the Containment Spray mode of RHR for mitigation of a SSLB accident ensures that the containment structural design temperature is not exceeded. Thus, ensuring that no design basis limits for fission product barriers are exceeded or altered.

CED 2001-0037
(USQE 2001-0014)- OLD RULE

TITLE: Replacement of the REC-SW System Piping

DESCRIPTION: This modification was prepared to remove two capped branch connections and one valved and capped instrument branch connection from each of the four 18-inch Service Water (SW) System headers. The purpose for removal of these unused branch connections was to eliminate low flow sites as potential Microbiological Induced Corrosion (MIC) breeding grounds. Two of these

capped branch connections were removed during CNS RE-20. These branch connections were removed via replacement of the associated piping spool pieces with new piping spool pieces without the branch connections. Removal of the remaining branch connections is currently planned for a future outage.

10CFR50.59
EVALUATION: This modification restores the affected SW System piping header sections to the original design requirements, while removing stagnant or low flow dead-ended branch connections considered potential MIC breeding grounds. The system design, operating modes, interfaces and safety design functions, and the operational function and parameters of the SW System remained unchanged, and thus did not increase the probability or consequences of an accident previously evaluated. The piping replacement does not change or alter the interface with any piece of equipment, and therefore does not increase the probability or consequences of a malfunction of equipment important to safety. The replacement piping was selected and purchased in accordance with the original piping specifications, and therefore cannot create any new malfunction of equipment important to safety. For the same reasons, the replacement piping cannot create the possibility of a different kind of accident. The modification did not change the function, operation, basis, safety limits for systems subsystems, or components described in the CNS Technical Specifications, and therefore did not reduce the margin of safety assumed in any CNS Technical Specification.

CED 2001-0045
(USQE 2001-0012)-OLD RULE

TITLE: Jumper Installation for 161 kV Switch, 1604-D2

DESCRIPTION: CED 2001-0045 installed a temporary jumper across a damaged 161 kV disconnect switch, 1604-D2. The jumper was installed to ensure that the current-carrying capability of the 161 kV line across the damaged disconnect switch was sufficient to meet station safety loads.

10CFR50.59
EVALUATION: The disconnect switch provides the passive function to conduct sufficient electrical current from the offsite transmission network to the Startup Station Service Transformer to support operation of station safety loads. The disconnect switch is provided only to support maintenance activities, and has no active safety function. The installation of the jumper across the damaged disconnect switch ensures that the load-carrying capability of the 161 kV line across the disconnect switch is sufficient to meet required safety loads. Therefore, the consequences of any accident previously evaluated are not increased. The jumper was sized consistent with the existing circuit and connections made using standard industry practices. The conductor length was limited to prevent interference of the line with other circuit phases or ground. Use of standard industry practices for installation of the jumper ensures that the probability of the loss of offsite power is not increased. Accordingly, installation of the jumper did not involve an unreviewed safety question.

EE 01-054
(Evaluation 2001-0042)- NEW RULE

TITLE: Evaluation of Non-Conforming Condition Associated With OD/OE 4-13806 – Torus Analysis Methodology

DESCRIPTION: A non-conforming condition was discovered wherein it was determined that a calculation methodology used in 1996/1997 to evaluate torus pitting was not identical to that used to establish the design basis requirements as delineated in the Plant Unique Analysis Report (PUAR). A 1996/1997 reanalysis of the torus was performed to reestablish a general corrosion allowance for the torus shell which was subsequently used to disposition instances of torus corrosion and pitting identified during outage inspections of the torus shell. Notably, the original design basis for the torus did include a 1/16" corrosion allowance, but during the original Mark I program analysis, credit for the full thickness was required to accommodate the hydrodynamic loads associated with

that program, therefore the corrosion allowance was disallowed. This reanalysis used different methodologies, analysis techniques, and computer software to remove unnecessary conservatism from the existing analysis.

10CFR50.59
EVALUATION: Torus failure is not an initiating event or single failure for any accident described in the USAR; therefore there is no associated increase in accident frequency. Periodic desludging activities and the ECCS suction strainers prevent adverse effects on SSCs important to safety resulting from acceptance of the torus corrosion allowance. Analyses demonstrate that the Primary Containment will remain within Code allowables as a result of this change and ECCS equipment will be unaffected; therefore, the accident and SSC failure consequences will be unchanged. Since the Primary Containment remains within Code allowables, there is no possibility for an accident of a different type. Since the function of ECCS equipment is not adversely affected, no new SSC malfunction possibilities are created with different results than previously evaluated. The torus remains within Code allowables; therefore, the associated Design Basis Limit for Fission Product Barriers (DBLFPB) is not exceeded. The new methodology used to calculate the torus stresses utilize either: a) methodology element changes that yield results that are conservative or essentially the same, or b) use new or different methods of analyses previously approved by the NRC for the intended application.

EE 01-071
FHACR 2001-001
(Evaluation 2001-0045)- NEW RULE

TITLE: Evaluation of the Impact of Rubatex Pipe Insulation on the Combustible Loading in Various Plant Areas

DESCRIPTION: It has been determined that Rubatex pipe insulation should no longer be considered non-combustible. Rubatex pipe insulation is an anti-sweat material and is currently utilized in relatively small amounts on cold water piping in various areas throughout the plant. As a result, NEDC 93-161 "CNS Fire Hazards Analysis Combustible Loading Calc" was revised to document the quantities of Rubatex and their associated contribution to combustible loading throughout the plant. Engineering Evaluation EE 01-071 was prepared to evaluate the impact of these additional combustible loads (due to the Rubatex pipe insulation) on the various plant areas.

10CFR50.59
EVALUATION: Rubatex has a relatively high fire ignition temperature (896°F) and based on the change to NEDC 93-161 the contribution to the combustible loading due to Rubatex throughout the plant is very small. The largest contribution to any fire zone within the plant is 0.4 minutes (24 seconds). The Special Event of a plant shutdown from outside the Control Room (due to a fire) is the only event that may be credibly affected as a result of the associated combustible loading increases due to Rubatex. Based on the low ignition temperature and negligible contribution to combustible loading these proposed changes do not change the frequency of occurrence of an accident nor do they change the likelihood of occurrence of a malfunction of an SSC important to safety as previously evaluated in the USAR. The small increases in combustible loading are bounded by the current Appendix R safe shutdown analysis and do not increase the consequences of malfunctions assumed as part of that analysis. There are no new failure modes that could initiate an accident of a different type or create the possibility for a malfunction of an SSC with a different result. There are no system parameter changes; therefore, no DBLFPBs are exceeded or altered. Assuming that Rubatex is combustible is a conservative change to a design basis methodology, and therefore does not constitute a departure from a method of evaluation described in the USAR.

EE 01-134
(Evaluation 2001-0050)- NEW RULE

TITLE: Revision of Calculation NEDC 96-039 for Issue as Status 1 Document

DESCRIPTION: This activity determines the maximum stroke times for selected DC powered MOVs under design basis differential pressure and degraded voltage conditions. The following valve stroke times become less restrictive as a result of implementation of this design calculation:

- HPCI-MOV-MO16 (HPCI Steam line isolation valve) design basis required closing stroke time limit is being increased from 50 seconds to 57 seconds for consistency with the limiting stroke times currently assumed in the High Energy Line Break (HELB) analysis.

- HPCI-MOV-MO25 (HPCI minimum flow line isolation valve) design basis required closing stroke time limit is being eliminated from the design basis. This is consistent with CNS Technical Specification Bases Section B3.3.5.1.3.f.

- RCIC-MOV-MO16 (RCIC Steam line isolation valve) design basis required closing stroke time limit is being increased from 19 seconds to 27 seconds for consistency with the limiting stroke times assumed in the HELB analysis.

- RCIC-MOV-MO27 (RCIC minimum flow line isolation valve design basis required closing stroke time limit is being changed from its Pre-Op Test Program acceptance value of 7.3 seconds to 15 seconds. Revision of the design basis closing stroke time limit is acceptable since the revised stroke time still ensures that RCIC will deliver its design flow within 30 seconds, as required by the Pre-Op Test program.

- RHR-MOV-MO67 (RHR suppression pool to Radwaste isolation valve) design basis required closing stroke time limit is being increased from the 20 seconds to 40 seconds.

- RWCU-MOV-MO18 (RWCU suction isolation valve) design basis closing stroke time limit is being increased from 30 seconds to 44 seconds for consistency with the limiting stroke times assumed in the HELB analysis.

10CFR50.59
EVALUATION: Most of the changes to the valve stroke times in NEDC 96-039 and EE 01-134 either do not change existing stroke time criteria or establish conservative stroke time criteria where no previous criteria had existed. Several of the proposed changes to the allowable DC valve closure stroke times did, however, result in slower stroke times than previously contained in the USAR. The valves having slower revised stroke times are not initiators of any accident, or equipment malfunction and do not introduce new failure modes or modes of operation. The revised stroke times were made to take into account design basis differential pressure and degraded voltage conditions which had not previously been included in the station design basis calculations, thus ensuring proposed failure modes or degraded conditions have been considered. The valves with slower closing stroke times are responding to accident initiation signals and thus provide mitigative features to minimize the consequence of the accident or protect equipment important to safety, rather than being initiators of the accident. The proposed changes do not affect the valve's limiting component analyses or its ability to function under degraded voltage conditions with design basis differential pressure. The changes do not introduce any new mode of valve operation, plant operation, system interfaces, create/initiate new types of equipment failure modes, or system lineups. The proposed changes do not result in a change that would cause any fission product barrier design basis limit to change (because these valve stroke time changes do not affect analyzed mitigation or containment isolation system performance), or create new system interactions and interfaces that would create new challenges to fission product barrier integrity.

No methods of evaluation used to establish the design basis or safety analysis of the facility are being affected by the changes being made to the valve stroke times.

EE 01-135
(Evaluation 2001-0049)- NEW RULE

TITLE: Jet Pump 6 Upper Bracket Sensing Line Weld Crack

DESCRIPTION: The upper support bracket for the sensing line of Jet Pump 6 has a crack-like indication on one side of the sensing line to bracket weld. The sensing line provides a d/p signal for the determination of jet pump loop flow and core flow. The weld is not going to be repaired and is being accepted "use-as-is." Reliance on the one remaining weld on the upper support bracket to support and restrain the sensing line does not alter the function of the sensing line. It has been concluded that one weld has and will continue to support and restrain the sensing line at the upper bracket.

10CFR50.59
EVALUATION: The sensing line inside the reactor vessel does not form part of the reactor coolant pressure boundary, nor does it provide input to protective equipment or interlocks. Therefore, the jet pump sensing line cannot be an initiator of any accident, transient, or special event, and accordingly, cannot increase the frequency of a previously evaluated accident. The remaining weld will adequately support the sensing line. Therefore, there is no increase in the likelihood of failure of an SSC important to safety. Since the sensing line is not part of any fission product boundary and performs no function important to safety, the consequences of an accident are not increased by reliance on a single weld on the upper support bracket. There is no effect on the consequences of previously evaluated SSC failures. The sensing line does not create the possibility of either an accident of a different type, or a malfunction of an SSC with a different result than previously evaluated. The sensing line has not interface with any fission product barrier, and therefore can have no effect on an associated design basis limit. The method of sensing line support is not an input to any method of evaluation described in the USAR.

EE 02-036
(Evaluation 2002-0006)- NEW RULE

TITLE: Station Blackout (SBO) Control Room Temperature Analysis

DESCRIPTION: The purpose of EE 02-036 is to document the acceptability of using a FORTRAN heatup analysis software code to evaluate the Control Room (CR) temperature following the onset of an SBO by demonstrating that the new software produces results that are essentially the same or conservative to the original CNS SBO CR temperature analysis, document resolution of the SBO CR temperature analysis non-conservatisms discussed in the CNS SBO Safety Evaluation, and recommend the maximum CR ambient air temperature during normal operation using the SBO CR temperature profiles. EE 02-036 establishes the configuration control basis for the CNS SBO CR temperature analysis design inputs; and provides the basis document for closing the SBO related OD/OE 10095767, revising USAR Section X-10.4.6.1 to reflect the design input parameter revisions made to address the non-conservatisms noted in the CNS SBO Safety Evaluation, and revising the operator actions taken in Procedure 2.4HVAC to address CR temperature equipment and personnel habitability requirements from an SBO standpoint.

10CFR50.59
EVALUATION: EE 02-036 and its associated recommendations involve the determination of Control Room temperature after an SBO has occurred. None of the accident or SSC malfunction (loss of onsite and offsite power) initiators or failure modes assumed in the SBO analysis are affected by this EE. Additionally, no other design basis accidents or other events are assumed to occur immediately prior to or during the SBO and no other independent failure events, other than those causing the SBO, are assumed to occur during the SBO transient. The proposed action to perform a plant

shutdown using existing station procedures, when certain Control Room temperatures are reached, preserves the SBO conclusion that the Control Room is not a Dominant Area of Concern and ensures Control Room personnel and equipment will be able to perform their credited actions or mitigative functions. Although the EE involves a method of evaluation and affects its elements of methodology the method of evaluation continues to be an analytical analysis, vice some other method of evaluation such as using actual plant data, test cases or generic heatup studies. The elements of methodology were revised but benchmarking and conservative selection of input parameters provides Control Room heatup results that are essentially the same or as conservative as the original Control Room SBO analysis. Therefore, EE 02-036 and its associated recommendations do not require prior NRC approval.

CED 4163326
(USQE 2001-0031)– OLD RULE

TITLE: Replacement of ASCO Series 8342 4-Way Solenoid Operated Valve (SOV) With Automatic Valve Company Model U0403AABR-AAS 4-Way SOV

DESCRIPTION: The SOVs serving RW-AOV-AO82, RW-AOV-AO83, RW-AOV-AO94, and RW-AOV-AO95 have been relocated, and are being and replaced with SOVs from another manufacturer. The associated 5/16" air tubing will be replaced with ½" tubing. This will address repeated failures to close within the required IST operability time.

10CFR50.59
EVALUATION: The proposed modification does not constitute an unreviewed safety question as it does not introduce any new modes of plant operation, create new systems or new system interfaces. Additionally, the proposed modification does not affect overall system design, existing system interfaces, operating parameters, surveillance test requirements, or methods for performing safety functions by plant SSCs credited in the SAR. Therefore, the probability of an equipment malfunction is not increased and the possibility of an equipment malfunction or an accident of a different type is not created. Similarly, the consequences of an accident or equipment malfunction are not increased by the changes. The modification does not reduce margins of safety assumed in any safety analysis, affect or create accident/transient initiators, affect fission product barriers, or affect accident/consequence mitigation systems or assumptions. Therefore, the CED does not increase the probability of an accident or create an accident of a different type, nor does it reduce the margin of safety as defined in the basis for any Technical Specification.

CED 6005480
FHACR 2002-001
TRMCR 2002-001
(Evaluation 2001-0047)-NEW RULE

TITLE: Control Room Fire Detection Modification

DESCRIPTION: Automatic fire detection that alarms in the control room is not provided for in the instrument maintenance area located behind the control room panels, or in the control room kitchenette; however, this thermal detector provides only a local bell alarm and is not connected to the station's automatic fire detection panel. Zone 17 of the Pyrotronics Panel currently provides detection coverage for the control room and the computer room area. This modification extends Zone 17 to provide coverage for the instrument maintenance area and the kitchenette via the addition of the new fire detection instruments. This CED upgrades the kitchenette (upgrade detector to a rate compensated thermal detector) and the instrument maintenance area (upgrade detector to an ionization smoke detector) detectors that are supervised by the Pyrotronics system. Conforming changes are being made to the Fire Hazards Analysis and the Technical Requirements Manual.

10CFR50.59
EVALUATION: The instrumentation installed by this CED does not act as a precursor or initiator to any design basis events described in the USAR; therefore, there are no associated changes to accident frequencies. Since there is no adverse effect on the reliability of the existing detectors, the likelihood of a malfunction of equipment important to safety is not increased. This installation has no impact on the radiological consequences of design basis events, or in achieving safe shutdown. Since the span of coverage is increased, fire-induced failures are more likely to be identified and suppressed. Thus, there are no increased radiological or safe shutdown consequences resulting from equipment malfunctions as a result of this CED. This CED does not create the possibility of new accidents or malfunction types. Since Safe Shutdown can still be achieved during an Appendix R fire in the control room, there is no threat to the DBLFPBs described in the USAR. This CED does not involve a method of evaluation as defined in the USAR.

CED 6006758
(Evaluation 2001-0048)-NEW RULE

TITLE: Lock Open Device for Mechanical Overspeed Butterfly Valve on Right Bank of DG-2

DESCRIPTION: CED 6006758 implements a Temporary Configuration Change (TCC) to install a gag (locking) device in order to maintain the Diesel Generator (DG) 2 right bank air inlet butterfly valve in the open position. This TCC is required as a result of damage to the valve control cable. The cable was returned to the manufacturer and its repair is estimated to take four weeks. The right bank air inlet butterfly valve must be gagged open to support testing and declaring DG 2 operable until the cable can be repaired and reinstalled.

10CFR50.59
EVALUATION: Since the DGs are not an initiator of any of the abnormal operating transients or postulated accidents described in the USAR, this temporary configuration change does not increase the possibility of a change in the frequency of an accident previously evaluated in the USAR. DG 2 retains the safety shutdown features and emergency operation/functions as specified in the USAR and does not more than minimally increase the possibility of the likelihood of a malfunction of an SSC important to safety previously evaluated in the SAR. This modification will not adversely affect the ability of DG 2 to provide emergency power to Engineered Safety Feature systems used for accident mitigation and therefore, does not increase the consequences of occurrence of an accident previously evaluated in the USAR. This temporary configuration change will not change the consequences of a failure of a DG on any other safety system as stated in the USAR and does not increase the consequences of a malfunction of an SSC important to safety. Changes to the DG cannot create the possibility of an accident of a different type (i.e., the DGs are a mitigation system) and as such, there is no increase in the possibility of an accident of a different type than previously evaluated in the USAR. The USAR assumes that only one onsite DG is available during the entire DBA LOCA, therefore, the possibility of a malfunction of an SSC important to safety with a different result than any previously evaluated in the USAR will not be created. The change incorporated by this TCC does not affect the accident analysis for the release of radioactive material and does not affect any of the radioactive material. The activity does not result in design basis limit for a fission product barrier as described in the USAR being exceeded or altered. This TCC does not result in a departure from a method of evaluation described in the USAR in establishing the design bases or in the safety analysis.

ATTACHMENT 2

PROCEDURE CHANGES

Procedure 0.24 (Revision 10)
Procedure 2.2.84 (Revision 32)
Procedure 10.25 (Revision 30)
Procedure 10.25.1 (Revision 11)
(USQE 2000-0039)- OLD RULE

TITLE: Working Over or In Reactor Vessel of Fuel Pool Requirements (0.24)
HVAC Main Control Room and Cable Spreading Room (2.2.84)
Refueling- Core Unload, Reload, and Shuffle (10.25)
Refueling- Core Refueling Support Operations (10.25.1)

DESCRIPTION: A nonconformance was identified in the design of the Reactor Building ventilation exhaust isolation valve configuration that could result in a limited unfiltered release during a Fuel Handling Accident (FHA), given a particular single active failure. This was determined to be an Unreviewed Safety Question. Operability Evaluation 2-24337 was generated which established that with certain compensatory measures (incorporated into the above procedures), CNS could continue to operate with an adequate margin to 10CFR100 and 10CFR50 Appendix A General Design Criteria 19 dose limits. The compensatory measures evaluated under this USQE were:

1) For movement of loads (which could potentially damage irradiated fuel) over fuel which has been irradiated within the last 293 hours, or movement of fuel which has been irradiated within the last 293 hours, the Control Room must be isolated with CREFS in service, and Reactor Building HVAC exhaust flow limited to a maximum of 51,000 cfm, as read from Plant Management Information System (PMIS).

2) For movement of loads (which could potentially damage irradiated fuel) over fuel which has been irradiated within the last 293 to 399 hours, or movement of fuel which has been irradiated within the last 293 to 399 hours, the Control Room must be isolated with CREFS in service if Reactor Building HVAC exhaust flow exceeds 51,000 cfm, as read from PMIS.

10CFR50.59
EVALUATION: The compensatory actions to manually place the CREFS in service and limit Reactor Building flow under specific refueling activity scenarios (as described in OE 2-24337, Revision 6), and the associated temporary procedure changes to implement these compensatory actions do not alter the physical configuration of the plant, and do not require any system to be operated in a fashion for which it is not designed. As such, they do not prevent the Secondary Containment from meeting its safety objective or safety design basis. The compensatory actions and associated procedure changes do not result in the initiation, or adversely affect mitigation assumptions, associated with accidents or transients described in the USAR. No new failure modes or accident initiators are created by this change since this change does not alter the design basis function, design operating parameters or system interfaces of plant systems. Accident mitigation features and functions are not altered by the compensatory actions. Accordingly, an accident or malfunction of a different type has not been created, and the probability or consequences of an accident or malfunction of equipment important to safety is not increased. Finally, since the compensatory actions and associated procedure changes do not impact the function or design operating parameters of plant equipment, and do not inhibit the mitigation features of these components, the margin of safety as defined in the basis of any Technical Specification is unaffected. In summary, the Operability Evaluation compensatory actions do not constitute an Unreviewed Safety Question.

Procedure 2.2.24.2 (Revision 2)
(USQE 2001-0026)- OLD RULE

TITLE: 250 VDC Electrical System (DIV 2)

DESCRIPTION: Procedure 2.2.24.2 is being revised to stipulate additional plant areas in which an hourly fire watch will be required when the "C" Battery Charger is used in lieu of the "B" Battery Charger to power the "B" 250 VDC bus. These compensatory measures are required since operating in this configuration may result in the inability to fulfill the safe shutdown and cold shutdown repair requirements of 10CFR50 Appendix R Sections III.G.1 and III.L.1.

10CFR50.59
EVALUATION: The proposed compensatory measures are not initiators of any accidents or transients. Therefore, the probability of an accident or a malfunction of equipment important to safety previously evaluated in the SAR are not increased. The consequences of an accident or a malfunction of equipment important to safety are not increased since these changes ensure appropriate compensatory measures are taken when the "C" charger is being used as a replacement for the "B" charger. Because no changes are being made to the equipment itself nor the manner in which it is operated, these changes will not create the possibility of an accident of a different type or possibility of a different type of malfunction. In addition, these changes have no effect on the acceptance limits for any system important to safety. Thus, no margins of safety as defined in the basis for any technical specification are reduced.

Procedure 2.2.25.2 (Revision 1)
(USQE 2001-0027)- OLD RULE

TITLE: 125 VDC Electrical System (DIV 2)

DESCRIPTION: Procedure 2.2.25.2 is being revised to stipulate additional plant areas in which an hourly fire watch will be required when the "C" Battery Charger is used in lieu of the "B" Battery Charger to power the "B" 125 VDC bus. These compensatory measures are required since operating in this configuration may result in the inability to fulfill the safe shutdown and cold shutdown repair requirements of 10CFR50 Appendix R Sections III.G.1 and III.L.1.

10CFR50.59
EVALUATION: The proposed compensatory measures are not initiators of any accidents or transients. Therefore, the probability of an accident or a malfunction of equipment important to safety previously evaluated in the SAR is not increased. The consequences of an accident or a malfunction of equipment important to safety are not increased since these changes ensure appropriate compensatory measures are taken when the "C" charger is being used as a replacement for the "B" charger. Because no changes are being made to the equipment itself nor the manner in which it is operated, these changes will not create the possibility of an accident of a different type or possibility of a different type of malfunction. In addition, these changes have no effect on the acceptance limits for any system important to safety. Thus, no margins of safety as defined in the basis for any technical specification are reduced.

Procedure 2.2.47 (Revision 23)
(USQE 2000-0045)- OLD RULE

TITLE: HVAC Reactor Building

DESCRIPTION: This procedure change revises the normal operating point of the Reactor Building differential pressure controlled from –0.25 inches of water differential pressure to a band that will keep the building air pressure between –0.30 to –0.33 inches of water subatmospheric. The purpose of this change is to prevent inoperability of Secondary Containment during routine Reactor Building ventilation changes.

10CFR50.59
EVALUATION: Secondary Containment and the reactor building ventilation system are not capable of being precursors to failures that would lead to abnormal transients or other events for which the plant is designed to cope and are described in the SAR. There is no increase in previously evaluated accident consequences since increasing the differential pressure improves the response of the system to LOCA and Fuel Handling Accidents by reducing the chance of uncontrolled outflow from the reactor building. Increasing the normal reactor building differential pressure will not cause a failure of Secondary Containment or reactor building system, structures, or any other components important to safety. The consequences of failures of equipment important to safety are similarly not increased. Operating at a higher reactor building differential pressure does not reduce the margins of safety for Technical Specification-controlled equipment for which this is an initial condition (e.g., SGTS and Secondary Containment).

Procedure 5.3SBO (Revision 1)
(USQE 2001-0025 Revision 1)- OLD RULE

TITLE: Station Blackout

DESCRIPTION: The proposed activity is a procedure change request (PCR) made necessary by the fact that the security system has added a second UPS under CED 2000-0192 which provides a power source for the Security Alarm Station (SAS) Distribution Panel UPP-C-1A separate from the CAS Distribution Panel UPP-GH-1A. This results in a second circuit in the NBPP designated as an alternate supply for the security system. Circuit No. 21 will remain as the alternate source for the original security system UPS1 which continues to provide the power supply to UPP-GH-1A. Previously spare Circuit No. 22 was designated by CED 2000-0192 as the alternate source for the new security system UPS2. This PCR reflects the revised configuration which will now required switching off two circuits at NBPP instead of one to strip the security system load in the event of a Station Blackout.

10CFR50.59
EVALUATION: This PCR does not constitute an unreviewed safety question as it does not delete or affect the overall functional aspects of existing programs, processes, procedures or activities at CNS; nor does it affect the design, installation, function, method of performing a function, system interfaces or operating parameters/margins of any plant systems, structures, and components important to safety. The PCR does not: a) introduce any new modes of plant operation, b) reduce margins of safety assumed in safety analyses, c) affect accident/transient initiators, d) affect fission product barriers, nor e) affect SAR accident/consequence mitigation analysis assumptions or input parameters.

Procedure 6.EE.607 (Revision 8)
Procedure 6.EE.608 (Revision 8)
UCR 2001-047
(USQE 2001-0018 and 2001-0019)- OLD RULE

TITLE: 250V and 125V Station Battery Performance Discharge Test

DESCRIPTION: The changes to 6.EE.607, 6.EE.608 and associated USAR change allows use of a new laptop computer with new software. The new computer/software make testing and reporting the results of the Station Batteries, easier and more controlled. The new software is easier to setup and has greater customization than the old computer/software. Steps have been removed to take pre-test readings within this procedure, they are performed in their own Surveillance procedure. Fire impairment information has been added, and information not required in certain tables have been removed. The requirement to use insulated tools has been added. Because this procedure will now be using a new computer and software, the steps that related to the old computer have been removed and replaced with the necessary steps to use the new computer. The new test allows one

test to effectively take the place of two tests. This practice is allowed by IEEE 450-1995. The new test is physically set up the same way as the old tests. The difference is in the duration and discharge rate, both of which are controlled by the computer software.

10CFR50.59
EVALUATION: The Station Batteries are accident mitigators, not initiators, and this change does not introduce a different type of accident or malfunction previously evaluated in the SAR. This change does result in a change that needs to be made to the USAR to reflect the proper version of IEEE 450 Standard to be used. Since neither the 125 VDC nor the 250 VDC systems contribute to any analyzed accident initiator, the probability of an accident previously evaluated in the SAR is not increased by this activity. As this activity is only applied to one 125 or 250 VDC battery at a time, any resulting loss of the affected battery is bounded by existing analysis. Therefore, the consequences of an accident previously evaluated in the SAR are not increased by this activity. The potential for an equipment malfunction is limited to the battery being tested under this activity. As noted above, the failure of a 125 VDC or 250 VDC battery has been analyzed in the SAR. Since loss of the affected battery will result in the same set of consequences regardless of the cause of this failure, the consequences of an equipment malfunction previously evaluated in the SAR are not increased by this activity. Further, each 125 VDC and 250 VDC battery is electrically independent of the other AC and DC power supplies, and no fault within the 125 VDC or 250 VDC battery can impact the remaining plant electrical distribution systems differently than analyzed. For these reasons, the possibility of an accident or malfunction of a different type cannot be created by this activity.

<u>Procedure 6.PC.503 (Revision 12)</u>
(USQE 2001-0009)- OLD RULE

TITLE: Drywell-to-Suppression Chamber Leakage Test

DESCRIPTION: The proposed change relaxes the acceptance criteria (operability limits) for the drywell-to-torus leakage test. The relaxed acceptance criteria remains bounded by CNS the licensing and design basis requirement of equivalent leakage through a "1-inch diameter orifice" as identified in Final Safety Analysis Report Amendment 15, Question 5.17. Additionally, the change places the Containment H_2 /O_2 monitors in standby, ensures the RCIC Gland Seal Vacuum Pump is not operating, and pumps down the drywell floor and equipment drain sumps prior to the leak test to ensure these systems do not contribute to the drywell-to-torus leak rate during testing.

10CFR50.59
EVALUATION: The changes do not entail new accident initiators or affect any accident mitigation assumptions related to the safety function or objective of Primary Containment. The relaxed acceptance criteria is within the parameters bounded by the 1-inch diameter orifice size, therefore the consequences of an accident previously evaluated in the SAR is not increased or changed. This change does not alter, modify or change the vacuum breakers; therefore, there is no increase in the probability of a malfunction of equipment important to safety, or increase in the radiological consequences of a malfunction of equipment important to safety. No new initiators of accidents are being implemented in the plant; therefore, an accident of a different type is not being created because existing conditions in the plant are not being changed or modified. The SSCs identified by this surveillance procedure are not altered or changed; therefore, there is no possibility of creating a malfunction of a different type than previously analyzed in the SAR. The design/licensing basis maximum bypass leakage continues to be based on a 1-inch diameter orifice; therefore, the proposed changes do not constitute a reduction in the margin of safety in any Technical Specification.

Procedure 6.SUMP.101 (Revision 12)
(USQE 2001-0028)- OLD RULE

TITLE: Z Sump and Air Ejector Holdup Line Drain Operability Test (IST)

DESCRIPTION: This procedure change replaces the requirement to declare a Standby Gas Treatment Subsystem or System inoperable when acceptance criteria are not met with the requirement to declare the associated Z Sump pump inoperable when the acceptance criteria is not met. It remains acceptable to enter the Standby Gas Treatment System Technical Specification Action Statement when the acceptance criteria is not satisfied, but it is not required due to the fact that the sump may still be capable of performing its design function with the degraded component.

10CFR50.59
EVALUATION: The procedure change revises the actions to be taken of the surveillance acceptance criteria are not met. This change does not affect either the initiators or initiation assumptions for the accidents for which the Standby Gas Treatment System is credited (LOCA and FHA), and therefore does no increase the probability of an accident previously evaluated in the SAR. This change has no adverse effect on the systems or assumptions used in mitigating the LOCA or FHA; therefore, the consequences of these accidents are not increased. The change has no impact on component malfunction, only what is required when the equipment is not working correctly. Accordingly, there is no effect on the probability or consequences of previously evaluated equipment failures in the SAR. Since the testing of the component is not being changed, the possibility of an accident of another type is not being created. The change satisfies existing Technical Specification requirements and processes; therefore, the margins of safety are unchanged.

Procedure 14.7.9 (Revision 7)
(USQE 2000-0032 Revision 1)- OLD RULE

TITLE: Main Turbine Governor Valve LVDT Replacement During Plant Operation

DESCRIPTION: This activity changes the maintenance procedure for replacement of a failed linear variable differential transformer (LVDT) on-line by manually controlling Governor Valve #1 by closing the hydraulic supply to the valve or by disconnecting the Moog servo valve amphenol connector (electronic signals to the valve). To accommodate this, Reactor Power is reduced to approximately 560 MWe. Each governor valve has its own hydraulic fluid supply valve and Moog servo valve connector. Therefore, this activity does not affect the function of the other governor valves. The other governor valves will control reactor pressure as Governor Valve #1 closes. Other turbine valves supplied by turbine high pressure fluid have their own hydraulic fluid supply isolation valve. None of these isolation valves are being manipulated by this activity.

10CFR50.59
EVALUATION: This activity isolates the oil supply to one governor valve in a slow controlled manner, which does not increase the likelihood of a governor valve fast closure event, or a turbine generator trip or generator load rejection without bypass event. Neither the governor valves nor the Digital Electro-Hydraulic (DEH) system are relied upon to mitigate the consequences of any of the events in the SAR, and this activity will not affect the response time of the bypass valves or change the stroke time of the governor valves. Therefore, this activity does not increase the consequences of an accident previously evaluated in the SAR. The process of placing the plant in a condition to repair the failed LVDT will not increase the probability of a malfunction of equipment as the process uses the operating equipment in the same manner in which it is currently operating. Closing one governor valve during plant operation at this reduced Reactor Power and operating in the single valve mode is an analyzed condition. A 0.03 value will be added to existing Minimum Critical Power Ratio limits for operating in single valve mode. Therefore, this activity does not increase the probability of a malfunction of equipment important to safety. This activity does not affect any of the equipment that is relied upon to mitigate the consequences of an event described in the SAR. Accordingly, the consequences of a previously evaluated malfunction of equipment important to

safety are not increased. This activity is bounded by the events described in the USAR; therefore, the possibility of an accident or malfunction of a different type than any previously evaluated in the SAR is not created. At the power level at which single valve operation is being conducted, only 3 governor valves are required to control reactor pressure; therefore, isolating Governor Valve #1 does not reduce the margin of safety as defined in the basis for any technical specification.

Special Procedure 99-006
Special Procedure 00-008
(USQE 2000-0036)- OLD RULE

TITLE: Emergency Feed of PCI In the 345 KV Substation

DESCRIPTION: This special procedure involves a change to the plant as described in USAR Sections VIII-2.2.5 and VII-4.5.1. The control and protection power to the 345 kV main station breakers and the 161 kV breakers is provided by 125 VDC circuits. Under normal conditions, 4160 station service buses 1A and 1B feed results in the control and protection power being supplied from the batteries. The batteries have a capacity sufficient to provide 8 to 10 hours of control and protection power to the breakers. The subject special procedure provides for the connection of a mobile generator to provide AD power to the battery chargers and subsequently the batteries in the event of a loss of 4160 station service buses 1A and 1B. This contingency action will eliminate the time constraint associated with the batteries.

10CFR50.59
EVALUATION: The station is designed to operate without the mobile transformer under all operating conditions, and a complete loss of offsite power is an analyzed event. Therefore, the probability of occurrence or consequences of an accident or malfunction previously evaluated in the SAR were not increased. The plant configurations associated with implementation of this special procedure are bounded by the previously analyzed loss of offsite power event. Technical Specification limiting conditions of operation will be complied with during implementation of this Special Procedure. Therefore, the margin of safety as defined in the basis for any Technical Specification will not be reduced.

ATTACHMENT 3

OTHER CHANGES[1]

USQE 1998-0073 (Revision 2)
OLD RULE

TITLE: Rerouting of the 161kV Line from 345/161kV Auto-Transformer to the Startup Station Service Transformer Via a New 161kV Switchyard.

DESCRIPTION: This safety evaluation was performed to analyze the addition of the 161 kV switchyard and the 161 kV Auburn line, completed in 1981 under MDC 81-53. The original design consisted of the T-2 transformer (345 kV/161 kV) supplying the Station Startup Service Transformer (SSST) (161 kV/4 kV) via a direct feed with no 161 kV breakers between them. In 1981, a 161 kV switchyard was added with a 161 kV transmission line to the city of Auburn. This switchyard and transmission line were designed and installed in accordance with standard industry practice for a 161 kV system. The original 10CFR50.59 screens/evaluations did not discuss the independence between the 161kV feed to the startup station service transformer and the 69kV feed to the emergency station service transformer, nor the impact of the 161 kV Auburn line on the operability of the SSST.

Revision 0 of this evaluation was reported in NPPD's 2000 10CFR50.59(b)(2) report. Subsequent to this report, two revisions were made to the evaluation. Revision 1 addressed concerns raised relative to the design requirements of 10 CFR 50, Appendix A, General Design Criterion 17, and CNS licensing basis requirements. Revision 2 was made to correct an error identified in the evaluation.

10CFR50.59
EVALUATION: Revision 2 of the Safety Evaluation states that the approved design continues to credit the T-2 transformer (with or without the Auburn Line concurrently connected) to be supplying power to the SSST. If supplied solely from the 161 kV Auburn Line, the SSST is not considered operable. Additionally, it was concluded the existence of the 161 kV Auburn Line does not adversely impact the reliability or function of the existing 345 kV or 69 kV transmission networks or the SSST or Emergency Station Service Transformer (ESST). The addition of the 161 kV Auburn line enhances the stability of the 345 kV and 69 kV transmission systems. The addition of the 161 kV Auburn Line has no adverse impact on the grid stability analysis, i.e., loss of CNS generation does not result in loss of the 345 kV or 161 kV transmission systems. Evaluation of the 161 kV Auburn Line failures determined that the existing breakers, coordination, and relaying adequately protected to the transmission network, the T-2, and the SSST transformers. The added load of the 161 kV line on the T-2 and the 345 kV transmission system is within the design rating of the offsite circuits and transmission network, and is adequately controlled by transmission network operators. Plant protective design features remain unaffected by this change. The routing of the 161 kV feeder line to the SSST maintains adequate physical separation and remains electrically separated. Accordingly, it was concluded that this change did not involve an unreviewed safety question.

1. This attachment includes the following types of change activities: USAR Change Requests (UCRs), Technical Requirements Manual Change Requests (TRMCRs), Operating License Change Requests (OLCRs), Drawing Change Notices (DCNs), Resolve Condition Reports (RCRs) corrective action program documents, and stand-alone USQEs.

RCR 1998-1222
(USQE 2001-0022)- OLD RULE

TITLE: Installation of Gage and Tubing On Bypass Valve Exhaust Header

DESCRIPTION: RCR 98-1222 was to evaluate the finding of a gage in the lubricating oil reservoir room that was not identifiable on any prints. Subsequent investigation and evaluation identified that the gage and associated tubing were connected to the exhaust side of the bypass valves. The connection was made at a test valve installed during original construction and shown on Jelco Drawing X2841-221. Corrective actions in the RCR were to close the test valve, disconnect the tubing from the bypass valves exhaust header, and evaluate the installation of the tubing and gage as an unauthorized modification. The tubing and gage were disconnected during RE19. The piping that this tubing is connected to is normally at condenser vacuum.

10CFR50.59
EVALUATION: The potential for air leakage via the tubing is within the capabilities of the air removal system, and the configuration would not affect the operation of the bypass valves. Therefore, there is no increase in the probability of an accident previously evaluated in the USAR, nor are there increased consequences to previously analyzed accidents. The tubing and fittings were installed in a manner that assured adequate pressure boundary integrity. The failure of the tubing or associated components would not result in a previously evaluated malfunction of equipment important to safety, or result in increased consequences from previously evaluated failures. Failures of the tubing are bounded by the turbine trip without bypass event, the main steam line break outside secondary containment, and the loss of condenser vacuum event. Accidents of a different type were not created by installation of this tubing. No new equipment was added that could fail in a way different than a previously evaluated pipe break or bypass valve malfunction. The margins of safety as defined in the basis for any technical specification remain unchanged as a result of this activity.

DCN 98-1984
OLD RULE

TITLE: Design Change Notice 98-1984 to Add HPCI Valves to B&R Drawing 2044

DESCRIPTION: DCN 98-1984 adds two valves to Burns & Roe Drawing 2044, which is incorporated into the USAR by reference. These valves were established by DC 84-114. This design change provided for the installation of a permanent vibration monitoring system for the High Pressure Coolant Injection) HPCI turbine. The design change included installation of eight vibration pickups with two machinery monitors installed on the 903' level of the Reactor Building.

10CFR50.59
EVALUATION: The installation of the vibration equipment will enhance the possibility of detecting HPCI mechanical problems due to abnormal vibration readings, and the installed equipment will not degrade the performance characteristics of the HPCI System. Accordingly, neither the probability of occurrence nor the consequences of an accident or equipment important to safety previously evaluated in the SAR will be increased. The design change will not affect normal functioning of the HPCI System. Therefore, there is no possibility of an accident or malfunction of a different type introduced. The margin of safety as defined in the bases to the Technical Specifications is not reduced.

TRMCR 1999-003
(USQE 1999-0020 Revision 1)- OLD RULE

TITLE: TRM 3.0 & "Enter Problem in CAP" Actions

DESCRIPTION: This change added Section 3.1.1 to he TRM. This new section provided the scram timing criteria for control rod scram testing at < 800 psig. The format, wording, and content are in accordance with NUREG-1433, Standard Technical Specifications, General Electric Plants BWR/4. September 97, rev. 0. The scram time limit for low reactor pressure came from CNS Startup Test #5, which provided test data for CNS single rod scrams at 0, 600, 800, 930, and 1000 psig. TRMCR 1999-003 initiated and promulgated these changes.

10CFR50.59
EVALUATION: The change to allow scram time testing < 800 psig will not cause systems to be operated outside of their design condition, and system and system interfaces will not be changed; therefore, the probability of accidents is not increased. The accident and transient analyses assume that all of the control rods scram at a specified insertion rate; therefore, establishing surveillance criteria that support this function does not increase the consequences of previously evaluated events. The plant SSC's are not being changed or operated in a different manner; therefore, the likelihood of a malfunction of equipment important to safety is not increased. Since this change helps demonstrate that scram times are within those assumed in the accident ant transient analyses, the change does not increase the consequences or a previously evaluated malfunction. Adding surveillance criteria for a depressurized scram does not involve, initiate, cause, or introduce an accident or a malfunction of a different type. The margins of safety defined in the bases to the Technical Specifications are not reduced by this change.

TRMCR 1999-006 (Revision 1)
(USQE 2001-0010)- OLD RULE

TITLE: Technical Requirements Manual Chloride Limits/Action Statements During NobleChem

DESCRIPTION: Chloride concentration limits are reestablished in the TRM that were erroneously removed in a previous TRM change. TRM Action Statements were added which correspond with the new chemistry limits of new Condition 3 (during NobleChem Application).

10CFR50.59
EVALUATION: The chemistry limits and associated Action Statements help to assure reactor coolant pressure boundary integrity, and other equipment affected by chlorides. Accordingly, there is no increase in the probability of an accident. The chemistry limits and Action Statements do not adversely affect equipment used to mitigate accidents or affect radiological source terms. Therefore, there is no impact on the consequences of an accident. The new chemistry limits and Action Statements are consistent with previously established limits and time frames. Thus, there is no increase in the probability or consequences of a malfunction of equipment important to safety. No new equipment or accident initiators are being added to the facility; therefore, there is no new possibility of accident or malfunction of a different type. The Technical Specification operating limits and the associated bases are not affected by this change; thus, there is no reduction in Technical Specification margins of safety.

OLCR 99-025
(USQE 2000-0053)- OLD RULE

TITLE: Technical Specification Bases 3.3.1.1 for SDV Level, Function 7

DESCRIPTION: For proper application of TS 3.3.1.1, Action and Surveillances, the RPS trip "Function" associated with a single SDV (north or south SDV) is required to be considered separately. While the

"Required Channels Per Trip System" in the Technical Specification Table 3.3.1.1-1 correctly specify the appropriate number of channels, the Bases description did not adequately detail which channels are associated with which "Function." This change provides the appropriate description of the design and clarification of associated "Function." Clarification is also made to define when "trip function" is maintained.

10CFR50.59
EVALUATION: Bases clarification consistent with the Technical Specification requirements and existing intent will not increase the probability or consequences of a previously analyzed accident, create a new or different type of accident, or decrease the margin of safety defined in the Technical Specification Bases.

TRMCR 2000-002
CED 2000-0114
(USQE 2000-0019 Revision 1)- OLD RULE

TITLE: PC-PS-119A, B, C, D Upgrade to EQ Including TRMCR

DESCRIPTION: CED 2000-0114 replaced the PC-PS-119A – D Containment Spray High Drywell Pressure Interlock Pressure Switches and associated terminal boards to establish environmental qualification (EQ). The change was necessary to upgrade the switches to EQ status following discrepancies identified with accident profiles, resulting in higher temperatures than those previously assumed which required crediting the Containment Spray function. In addition, the setpoints for PC-PS-119A – D were changed from ≤ 2 psig to ≥ 2 psig to be consistent with the licensing and design basis. This change required a corresponding change to the Technical Requirements Manual, Table 3.3.2-1, Function 3.a allowable value for these instruments. A change in the calibration frequency from once/92 days to once/18 months was also made.

10CFR50.59
EVALUATION: The function of the pressure switches is to provide a permissive interlock function for operator initiation of Containment Spray. During accident mitigation, the loss of this permissive on decreasing pressure will provide greater margin to the –2 psig containment design pressure. The Containment Spray function will continue to be available for operator use in mitigating accidents. This change does not introduce any new accident initiators. Therefore, this change did not increase the probability or consequences of an accident previously evaluated. The pressure switches are being replaced on a one-for-one basis with switches that are EQ qualified. The setpoint change will reduce the probability of inadvertent Containment Spray initiation by increasing the margin between the normal operating pressure and the Containment Spray initiation permissive. The calibration frequency is supported by a calculation using GE Setpoint Methodology, which ensures that the switches will continue to meet their performance requirements. Accordingly, this change did not increase the probability or consequences of a malfunction of equipment important to safety, nor create the possibility for a new malfunction of equipment or accident. The setpoint change will improve margin between normal operating pressure and the Containment Spray permissive, providing a corresponding increased margin for inadvertent Containment Spray initiation. The setpoint change also provides greater margin to the –2 psig containment design pressure, by initiating Containment Spray header valve closure sooner upon decreasing containment pressure. Therefore, this change did not reduce the margin of safety defined in the basis for any Technical Specification.

OLCR 2000-007
(USQE 2000-0012 Revision 1)- OLD RULE

TITLE: Change of Low Pressure Requirement for Testing HPCI and RCIC

DESCRIPTION: Technical Specification Bases 3.5 was revised to change the defined adequate reactor steam pressure from 150 psig to 145 psig to take credit when running HPCI and RCIC below 150 psig. It is conservative to test HPCI and RCIC at a reactor steam pressure less than 150 psig because it demonstrates that the required flow can be obtained at a lower pressure. Accordingly, if the systems can meet all acceptance criteria before 150 psig, then there is no concern that they can meet the criteria above 150 psig.

10CFR50.59
EVALUATION: This surveillance requirement demonstrates the operability of HPCI and RCIC and is not a precursor to an accident. The change does not alter the design function of HPCI or RCIC, and testing at lower RPV pressure does not adversely affect any interfacing SSCs; therefore, the consequences or previously evaluated accidents are not increase. The revised pressure range remains within the design basis of HPCI and RCIC and the change does not cause any equipment to malfunction. Accordingly, neither the probability of a malfunction of equipment important to safety, nor the consequences thereof, is increased. Surveillance testing that remains in accordance with the Technical Specifications, and that are performed prior to the time HPCI and RCIC are required to be operable, does not create the possibility of an unevaluated malfunction or accident. This change does not alter the intent of the Technical Specification for testing HPCI and RCIC to assure operability; therefore, there is no reduction in the margin of safety defined in the bases for any technical specification.

TRMCR 2000-007
(USQE 2000-0034)- OLD RULE

TITLE: Delete TRM Specification T 3.6.1 (Liquid N_2) and associated Bases

DESCRIPTION: This change deletes TRM Specification T 3.6.1 and associated Bases related to normal liquid nitrogen storage quantities. This TRM Specification, and its requirement for a plant shutdown is not necessary for plant safety. Plant shutdown would not be required until actual drywell oxygen concentrations exceeded the limits of Technical Specification 3.6.3.1.

10CFR50.59
EVALUATION: Failure to maintain a minimum quantity of liquid nitrogen is not an initiator of any plant event. The consequences of previously analyzed events that credit an inerted drywell atmosphere are dependent on the initial oxygen concentration in the drywell, and not on any quantity of liquid nitrogen in the storage tank. The analyzed limits on drywell oxygen concentration are still maintained, and the credited post-accident nitrogen makeup is provided by the Standby Nitrogen Injection System; therefore, there is no increase in the probability or consequences of previously evaluated equipment important to safety. No new accidents or malfunctions of equipment important to safety are introduced by this change. The Technical Specification requirement for drywell oxygen concentration limits assures all related margins of safety are being met.

TRMCR 2000-009
(USQE 2000-0038)- OLD RULE

TITLE: Revise TRM Specification & 3.8.1 (Battery Room Ventilation) and associated BASES Section

DESCRIPTION: This TRM change revises TLCO 3.8.1 by: 1) redesignating a portion of CONDITION B as a new CONDITION C to require that a PIR be initiated for evaluation of the degraded condition if Action A is not met, 2) replacing REQUIRED ACTION B.1, "Be in MODE 4" and associated

COMPLETION TIME, "24 hours" with "initiate action to establish battery room ventilation" and "Immediately," and 3) revising the associated BASES to support that initiating actions to restore Battery Room ventilation is governed by approved procedures.

10CFR50.59
EVALUATION: This TRM change does not involve a physical alteration of the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. This change will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No existing equipment or system failure mechanism has been increased, nor has any new failure mechanism been introduced. The specific changes are consistent with approved CNS Procedures and Station Service Battery OPERABILITY continues to be governed by CNS Technical Specification requirements. Therefore, no unreviewed safety question exists and the proposed change may be implemented without prior NRC approval.

TRMCR 2000-010
(USQE 2000-0050)- OLD RULE

TITLE: Entering and Exiting Technical Specification LCO Conditions; Safety Function Determination Program

DESCRIPTION: This change provides a new process for an administrative program. The intent of the Safety Function Determination (SFD) Program is to detect a loss of safety function should it be necessary to declare an SSC inoperable. When a loss of safety function is detected, it is required to enter the LCO Conditions and Required Actions for the supported system. This change enhances nuclear safety by improving the implementation of the SFD process. The general guidance in the TRM was deleted and a new procedure was created with step-by-step instructions.

10CFR50.59
EVALUATION: This change to the TRM is administrative in nature. Accordingly, there is no effect on the probability or consequences of previously evaluated accidents or malfunctions of equipment important to safety. There is no effect on the possibility of new accidents or malfunctions. This change does not affect any safety margins described in the Technical Specifications.

TRMCR 2000-011
(USQE 2001-0001)- OLD RULE

TITLE: Control Rod Block Instrumentation "Scrub"

DESCRIPTION: The following changes have been made to TRM 3.3.1, "Control Rod Block Instrumentation":

- The "Required Channels per Trip System" column heading is revised to be "Required Channels" in recognition that the Control Rod Block logic is a 1-out-of-"n" trip logic for each function. Appropriate changes are also made to the specified number of channels for consistency.
- The Source Range Monitor (SRM) "not-full-in" Function "Allowable Value" is revised to "N/A," as well as consistent with the CNS Intermediate Range Monitor (IRM) Not Full-in Function of this same table. The Applicability is also revised to reflect new Footnote (b) indicating that the range of MODE 2 Applicability for this Function is "with SRM count rate < 100 cps."
- Footnote (a) (applied to Functions 1.a, b, c, d for the SRMs and 2.a, b, c, d for IRMs) is therefore eliminated.
- As a MODE 2 modifier, the Applicability for Function 1.d, SRM Downscale, and Function 2.d, IRM Downscale, is more clearly stated for Function 1.d in proposed Footnote (a) as "with IRMs on Range 2 or below"; and more clearly stated for Function 2.d in proposed Footnote (d) as "with IRMs on Range 2 or above."

- The Bases detail for the SRM and IRM Detector-Not-Full-In Function calibration have clarifying information added reflecting the appropriate applicability of the requirement to TSR 3.3.1.7; the 18 month Channel Calibration.
- The SRM Downscale Function for Mode 5 is revised to eliminate the Footnote to the Allowable Value. This imposes the requirement for an operable downscale rod block Allowable Value of 3 cps during all MODE 5 operation.
- Created a new Technical Requirements Manual (TRM) surveillance TSR 3.3.1.2 applicable to Function 3.a and 3.e (Average Power Range Monitor (APRM) Control Rod Block MODE 1 Upscale Functions). Wording is identical to Technical Specification SR 3.3.1.1.2. This provides consistency in APRM Control Rod Block calibrations with the Technical Specifications for APRM RPS calibrations for these functions.
- IRM & APRM TRM Rd Block Required Channels, Footnote (h), limits the minimum total required channels to a specific subset; i.e., those associated with each RPS Trip System. This total number of channels is 4 APRMs (with a 1-out-of-8 logic for initiating a rod block). This footnote restriction is eliminated, allowing any 4 APRMs and 6 IRMs. This provides sufficient protection without restricting the channels on a per-RPS-Trip-System basis.

10CFR50.59
EVALUATION: The operability for the credited control rod block function is maintained by the requirements of Technical Specification 3.3.2.1, which is unaffected by this change. These TRM requirements are not credited in any design basis event. Therefore, these changes do not involve an increase in the probability or consequences, or decrease in the margin of safety of any accident previously evaluated. The design and operation of the control rod block instrumentation is unchanged. As such, no new failure modes are being introduced, and the change does not create the possibility of a new or different kind of accident than previously evaluated in the SAR. The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The TRM control rod block instrumentation is not credited in any design basis event and does not reflect a margin of safety.

OLCR 2000-018
(USQE 2000-0035)- OLD RULE

TITLE: Delete Technical Specification 3.6.4.2 Bases Reference to Location of SCIV Stroke Times

DESCRIPTION: Currently, the LCO Bases for Technical Specification 3.6.4.2 contains a statement that Secondary Containment Isolation Valve (SCIV) stroke times are located in the TRM, which is inconsistent with the TRM. The TRM does not list SCIV stroke times. Adding references to the required stroke times into the TRM is not necessary. This change will delete that portion of the sentence that indicates SCIV associated stroke times are listed in the TRM.

10CFR50.59
EVALUATION: With this deletion of an erroneous Bases reference to the TRM as containing the required stroke time for SCIVs, the USAR (and Bases) continues to appropriately reflect the plant design. No change in the design or method of performing the function of secondary containment isolation is made. The Bases correction will not increase the probability or consequences of a previously analyzed accident, create a new or different type of accident, or decrease the margin of safety defined in the Technical Specification Bases.

OLCR 2000-027
(USQE 2000-0051)- OLD RULE

TITLE: TS 3.8.1 Bases

DESCRIPTION: There were no details in the Technical Specification Bases regarding the boundaries of the portions of the electrical offsite and onsite electrical distribution system that are intended to constitute the portion required by Technical Specification 3.8.1. This administrative change adds clarification providing the details of the components that encompass the Technical Specification offsite circuits.

10CFR50.59
EVALUATION: This change enhances the Bases presentation by providing more detailed information. The requirement for offsite circuit operability remains unchanged. Therefore, there is no associated increase in either the probability or consequences of previously evaluated accidents. There are no physical alterations to the plant, or changes to the procedures, setpoints, or functional demands on plant equipment. Accordingly, there are no increases in the probability or consequences of malfunctions of equipment important to safety. This added informational detail is not affecting any parameter or equipment credited in any design basis event and does not reflect a margin of safety.

UCR 2000-038 Change 101
(USQE 2000-0026)- OLD RULE

TITLE: Change to USAR Chapter X, Section 14.6.2.1

DESCRIPTION: This USAR revision incorporates the latest revisions of the internal flooding calculations into the USAR. They indicate that the 16" and 24" RHR Moderate Energy Line Break (MELB) is no longer a threat to environmentally qualified equipment in the NW Quad. Additionally, for the 18" FW line break, the resulting flooding of the quad areas does not adversely affect any essential components that are needed to mitigate that event.

10CFR50.59
EVALUATION: Since the revised internal flooding calculations deal only with events following a Feedwater (FW) High Energy Line Break (HELB), they cannot affect the probability of an accident occurring. In the most limiting of the revised analyses, the dose consequences of the FW HELB remain bounded by the Main Steamline Break accident. The threat to Environmentally Qualified devices in the NW Quad from internal flooding is eliminated, and other the devices that will become submerged are not necessary to mitigate the FW HELB. Therefore, there is no increased likelihood of malfunction of equipment important to safety. There are no increased dose or safe shutdown consequences from assumed single failures or event initiators due to equipment malfunctions currently described in the USAR. Eliminating the threat to environmentally qualified devices in the NW Quad due to an RHR MELB does not cause a new plant event. The revised FW break is bounded for existing USAR accident analyses; therefore, no new accident possibilities are introduced. Since the calculations only deal with events following an accident and do not in any way affect normal plant operations prior to an accident, they cannot cause the possibility of a malfunction of equipment important to safety of a different kind than previously considered in the Safety Analysis. Since compliance is maintained with the requirements of 10CFR50.49, no reduction in safety margin results.

UCR 2000-045
Procedure 0.29.1 (Revision 12)
Procedure 0.8 (Revision 7)
(USQE 2000-0022)- OLD RULE

TITLE: USAR Change to Remove 50.59 Requirement for ODAM Changes

DESCRIPTION: This activity is a revision of the USAR and implementing procedures to eliminate the requirement that Offsite Dose Assessment Manual (ODAM) changes be reviewed under 10CFR50.59, as change control requirements are outlined by Technical Specifications section 5.5.1.c.

10CFR50.59
EVALUATION: This change does not constitute an unreviewed safety question as it is administrative in nature and does not affect the overall functional aspects of existing programs, processes, procedures, or activities at CNS; nor does it affect the design, installation, function, method of performing a function, system interfaces, or operating parameters/margins of any plant SSCs important to safety. The change does not introduce any new modes of plant operation, install new equipment, reduce margins of safety assumed in safety analyses, affect accident/transient initiators, affects assumptions or input parameters. Therefore, NRC approval was not required prior to implementation of this change.

UCR 2000-054
UCR 2001-009
(USQE 2000-0037, 2001-0003)- OLD RULE

TITLE: Organizational Changes

DESCRIPTION: UCR 2000-054 revised the CNS organizational structure and responsibilities as described in the USAR. Specifically, a transfer of reporting assignments for the Manager of Projects to the Senior Manager of Technical Services along with the Nuclear Licensing & Safety Manager; a transfer of reporting assignments for the Facilities & Construction Manager to the Plant Manager; and the transfer of the Business Services manager to the Fuels & Reactor Engineering Manager.

UCR 2001-009 revised the CNS organizational structure and responsibilities as described in the USAR. Specifically, the Work Control Department and the Outage Department were combined into one organizational group reporting to the Work Control Manager who shall report to the Plant Manager.

10CFR50.59
EVALUATION: These changes do not constitute an unreviewed safety question as they are administrative in nature and do not delete or affect the overall functional aspects of existing programs, processes, procedures, or activities at CNS; nor do they affect the design, installation, function, method of performing a function, system interfaces, or operating parameters/margins of any plant SSCs important to safety. The changes do not: introduce any new modes of plant operation, install new equipment, reduce margins of safety assumed in safety analyses, affect accident/transient initiators, affect fission product barriers, nor affect SAR accident/consequence mitigation analysis assumptions or input parameters.

UCR 2000-055
(USQE 2000-0042)- OLD RULE

TITLE: Update USAR Table III-3-1

DESCRIPTION: This USAR Change updates Table III-3-1, "Reactor Vessel Thermal Cycles." The allowable cycles for the current analysis have been added. The latest analysis takes into consideration repairs

and replacements that have affected the Reactor Coolant Pressure Boundary and more rigorous analysis techniques than were originally used.

10CFR50.59
EVALUATION: The analysis is not a USQ because the design basis, a cumulative usage factor less than 1.0 over the life of the plant, has not been changed. The vessel thermal cycles are not a precursor to any design basis accident, are not used in accident mitigation, and the possibility of a new or different kind of accident or malfunction has not been created. Operational events and transients are monitored to ensure that the accumulated fatigue usage factor remains less than the limit.

UCR 2000-057 Change 19a
OLD RULE

TITLE: USAR Rebaseline Project- Chapter IX, Change 19a

DESCRIPTION: This USAR change incorporates the effects to the Liquid Radwaste System by MDC 84-265. The MDC modified the system so that water from the Distillate or Chemical Waste Sample Tank in Augmented Liquid Radwaste could be transferred directly to the Waste Collector Tank.

10CFR50.59
EVALUATION: The Radwaste System is not designed as a safety system to mitigate accident conditions and is not a precursor to any design basis accident or transient. Therefore, neither the possibility nor consequences of accidents previously evaluated are affected. There is no interface with safety equipment; therefore, there is no effect on the likelihood of previously evaluated safety equipment malfunctions or their consequences. This change does not create the possibility of new malfunctions or accidents. There are no effects on the safety margins of any technical specification bases.

UCR 2000-057 Change 19c
(USQE 2000-0023)- OLD RULE

TITLE: USAR Change, Centrifuge Effluent

DESCRIPTION: UCR 2000-057 Change 19c relocates the Centrifuge Effluent as one of the sources of the Waste Collector Tank for High Purity Waste to the floor Collector Drain Tank as a source of Low Purity Waste. This reflects the effects of MDC 74-105 on the Updated Safety Analysis Report.

10CFR50.59
EVALUATION: Since the Radwaste System is not considered the initiator of an event or accident, the probability of accidents or of malfunctions of equipment important to safety previously evaluated is not changed. Similarly, the consequences of accidents and malfunctions previously evaluated are not changed. Since changes within the Radwaste System did not affect the parameters or limitations on releasing waste product to the environment, there are no new possibilities of accidents or equipment malfunctions not previously evaluated. Since the change does not affect equipment important to safety, there are no changes to Technical Specification margins.

UCR 2000-057 Change 98
OLD RULE

TITLE: USAR Rebaseline Project- Chapter IX, Section 4.5.1, Change 98

DESCRIPTION: This USAR change implements the effects of MDC 76-9 which provided for venting the Z sump to the Elevated Release Point (ERP). This change was in response to a 1976 hydrogen explosion in the Offgas Building.

10CFR50.59
EVALUATION: The Z Sump is not part of a system previously evaluated as an event initiator; therefore, the change to the vent can not increase the probability of an accident. Although it is unlikely that fission product gasses would migrate through the Standby Gas Treatment System (SGTS) to the Z Sump and out to the ERP, this potential leakage path has been considered in the offsite dose calculation; thus, he change in the Z Sump vent path will not increase the consequences of an accident. There is no credible malfunction of the Z Sump vent that would initiate any design basis accident. Furthermore, this modification reduces the possibility of a hydrogen gas explosion in the offgas line, which could damage equipment important to safety. The Offgas System and Augmented Offgas System isolate in the event of a design basis accident and the Z Sump has no active components. Thus, no credible malfunctions of the Z Sump vent would increase the consequences of any design basis accident. Rerouting the vent to the ERP does not change the event initiators or create the possibility of a different type of accident than previously evaluated. The change to the Z Sump vent path does not affect the bases of the SGTS technical specifications.

UCR 2000-058 Changes 34 and 41
OLD RULE

TITLE: USAR Rebaseline Project, Chapter VII, Binder 23, Change 34 and 41

DESCRIPTION: This USAR change reflects the effects of DC 86-147. This design change removed a differential pressure transmitter and indicator as a result of information from O&M Reminder 299 that indicated oscillations in a sensing line could be transmitted to another sensor and cause an inadvertent trip. References to a differential pressure transmitter and indicator that were removed as a result of DC 86-147 are being removed from USAR text.

10CFR50.59
EVALUATION: The function of the differential pressure transmitter was to provide indication on Panel 9-3. This indication had no alarm or control function. No safety functions were affected since this transmitter was for indication only. The stability and reliability of the plant during operation was improved in that any pressure oscillations that could have occurred would not be able to be coupled across this transmitter and cause an inadvertent scram. Accordingly, neither the probability of occurrence or consequences of an accident, nor the malfunction of equipment important to safety were increased. The transmitter and indicator were not identified in the Technical Specifications as required instruments, and the isolation of the transmitter did not reduce the margin of safety as defined in any Technical Specification basis.

UCR 2000-059
(USQE 2000-0048)- OLD RULE

TITLE: USAR Change to Allow Local Actuation of Stop Valve Limit Switches

DESCRIPTION: This activity changes the USAR to indicate that local actuation of the Main Turbine Stop Valve Reactor Protection System limit switches is an acceptable testing method to perform the Technical Specification surveillance requirement for a channel functional test. The previous USAR text only described performing testing by remote manipulation.

10CFR50.59
EVALUATION: Local actuation of the limit switch does not increase the probability or consequences of the plant events that rely on the Main Turbine Stop Valve as the primary scram signal to the reactor. Verification methods by local actuation of the limit switches ensure the entire channel will perform its intended function when called upon; therefore, neither probability nor the consequences of malfunctions of equipment important to safety are increased. Local actuation of the limit switches will not increase the potential for equipment damage more than the existing testing procedures, nor will it create a different failure mode. The change will also not cause the equipment to malfunction

or leave it in a degraded state that would result in a different malfunction. Therefore, the possibility of a malfunction of a different type or a new accident is not created. The new testing methods do not alter equipment configuration or performance, and at no time during the test is the stop valve closure scram inhibited; therefore, the margin of safety as defined in the basis for any technical specification is not reduced.

UCR 2000-060 Change 8
(USQE 2000-0040)- OLD RULE

TITLE: USAR Rebaseline Project, Chapter VII, Binder 29, Change 8

DESCRIPTION: This USAR change revises USAR Section VII-11.5.2, paragraph 2.d. The paragraph discusses the means of testing the backup overspeed trip device of the Main Turbine. The change corrects a previous USAR change that inappropriately designated the electrical overspeed device as the backup trip mechanisms. The mechanical overspeed trip is set to trip at a higher speed than the electrical speed trip, and is therefore the backup trip.

10CFR50.59
EVALUATION: The order of preference of the electrical and mechanical trips does not affect accident probability, and the consequences of previously evaluated accidents are unchanged. The primary purpose of the overspeed trip devices is to protect the Main Turbine from excessive rotational forces. The overspeed trip devices do not change the operation of interfacing SSCs that are important to safety; therefore, there is no increase in the probability or consequences of a malfunction of safety equipment. An overspeed of the Main Turbine does not provide any new pathways for radiological release or increase the radiological release of existing pathways. The order of trip preference does not change their function or the function of components and equipment dependent on their actuation; therefore, the possibility of a different type of malfunction is not created. Since the setpoints remain constant regardless of the initiating device, there is no change to the margin of safety.

UCR 2000-060 Changes 69 and 104
(USQE 2000-0047)- OLD RULE

TITLE: USAR Rebaseline Project, Chapter VII, Binder 23, Change 69 and 104

DESCRIPTION: These USAR changes revise the discussions of specific types of radiological protection equipment to a discussion of the function and capabilities of the equipment. The change rewrites existing information for the purpose of clarity. The type of devices necessary to monitor personnel contamination, measure radiation, analyze radioactive gases, liquids, or solids does not pertain to the type or level of detection needed depending on the circumstances involved. The meaning is modified to develop a better understanding of the purposes of radiological protection and radiochemistry instruments.

10CFR50.59
EVALUATION: Radiological instrumentation is not considered the initiator of an event, accident or transient. Radiological instrumentation being discussed does not mitigate an accident, but performs detection and data analysis of samples taken under various plant conditions. The instrumentation does not interface with equipment important to safety; therefore, there are no effects on the probability of malfunction or consequences of safety equipment that have previously evaluated. There is no interface of radiological equipment with SSCs involved in abnormal transients, accidents, or events. Therefore, the possibility of a different malfunction or accident is not created. There are no effects on the margins of safety defined in the bases of the Technical Specifications.

OLCR 2001-001
(Evaluation 2000-0052)- NEW RULE

TITLE: TS Bases Correction Regarding ECCS Pump Minimum Flow Valve Setpoint

DESCRIPTION: The Technical Specification Bases 3.3.5.1 statement regarding Low Pressure Coolant Injection (LPCI), Core Spray (CS), and High Pressure Coolant Injection (HPCI), pump discharge low flow instrumentation allowable value ("yet low enough to ensure that the closure of the minimum flow valve is initiated to allow full flow into the core") was deleted. No correlating Technical Specification exists to this Bases statement. This is considered to be an administrative change.

10CFR50.59
EVALUATION: This changes revises the Technical Specification Bases to ensure consistency with the Technical Specifications. Accordingly, there is no change to the frequency of occurrence or consequences of previously evaluated accidents. Similarly, no changes result in the likelihood or consequences resulting from previously evaluated malfunctions of equipment important to safety. Since there are no physical alterations to the plant or changed operating modes of existing equipment, no new failure modes are being introduced. Therefore, no possibility is created for a new malfunction of equipment important to safety, or of an accident of a different type. Since the change is administrative in nature, there is no effect on any design basis limit for a fission product barrier. There is no effect on existing methodologies described in the USAR for establishing the design bases or used in the safety analysis.

TRMCR 2001-002
OLCR 2001-013
(USQE 2001-0015)- OLD RULE

TITLE: EBS Phase II Project Workbook/EBS Phase II Integration Test Plan

DESCRIPTION: As a result of the Enterprise Business Solution Project, and adoption of the SAP R/3 software for use in CNS work management, the term "Notification" was used to replace "Problem Identification Report" to document and resolve issues associated with the Corrective Action Program. This resulted in administrative changes to the TRM and ODAM to incorporate this new term.

10CFR50.59
EVALUATION: The adoption of the SAP R/3 software results in administrative changes that do not affect the probability or consequences of previously evaluated accidents. Data transfers to the new system were verified and validated, such that the possibility or consequences of previously evaluated safety equipment malfunctions are not affected. Implementation of the SAP R/3 software is a business application for integrating CNS work and data; therefore, no possibility is introduced for new accident or safety equipment malfunctions. The SAP R/3 software has no effect on software associated with control of the reactor or any safety systems. Accordingly, there is no impact on the margin of safety in the basis for any technical specification.

TRMCR 2001-003
(USQE 2001-0023)- OLD RULE

TITLE: TRM 3.0 & "Enter Problem in CAP" Actions

DESCRIPTION: Several changes are made, the most significant of which is to explicitly provide the applicable Section 3.0 requirements within the TRM rather than the existing approach of incorporation of the Technical Specification Section 3.0 by reference. This TRM change eliminates the majority of TRM required plant shutdown actions by revising TRM LCO 3.0.3. The revised LCO provides: a) an Action to initiate efforts to restore compliance with the TLCO or associated Actions, and b) an Action that requires entering the circumstances into the Corrective Action Program.

10CFR50.59
EVALUATION: The operability and function of mitigating systems and the operation of the facility within assumed initial conditions are unaffected by this change. In each case, the changes were evaluated to not impact any accident initiators, or features that perform a mitigative function. Thus, the changes do not result in an increase in the probability or consequences of a previously analyzed accident. Furthermore, they do not introduce the possibility for an accident or malfunction of a different type than evaluated previously. Additionally, the margin of safety defined in the basis for any Technical Specification is not reduced.

UCR 2001-003 Change 53
OLD RULE

TITLE: Revise USAR Sections VII-1.7.3.3.E and F to Reflect Sprinkler Coverage

DESCRIPTION: This USAR change reflects the effects of MDC 79-12 and MDC 79-32 which modified the sprinkler system for areas in the Reactor Building, Motor-Generator (MG) Set Oil Heat Exchanger area, Cable Expansion Room, Laundry Room, and the Instrument Storage area. These modifications increased the area protection factor for these areas.

10CFR50.59
EVALUATION: The design changes increase plant reliability by increasing the area protection factor, and increase the reliability of the Fire Protection System. Therefore, neither the probability an occurrence, the consequences of an accident, nor the malfunction of equipment important to safety is increased. The integrity of not only cable trays, but all equipment in the area covered by the addition of this sprinkler system will be reinforced. Compliance with applicable codes is ensured both during and following implementation. Accordingly, the possibility of an accident or malfunction of a different type than previously evaluated is not increased. The margins of safety as defined in the bases for the Technical Specifications are not reduced.

TRMCR 2001-006
(Evaluation 2001-0040)- NEW RULE

TITLE: Change to Technical Requirements Manual Basis Section B 3.3.3

DESCRIPTION: The change to the TRM Bases deletes reference to drywell temperature recorder PC-TR-503 and adds reference to drywell temperature indicators PC-TI-510 A, B, C, D and E for instruments that are required to meet TSR 3.3.3.1 and 3.3.3.6. This change reconciles the TRM with the Regulatory Guide 1.97 Instrument List.

10CFR50.59
EVALUATION: The TRM bases change has no impact on the frequency of occurrence of an accident previously evaluated in the USAR because the proposed TRM change does not involve or affect any possible accident initiator and no new failure modes are introduced due to the TRM change. Since the TRM change does not involve or affect any possible initiator of a malfunction of an SSC and no new failure modes are introduced, the change has no adverse impact on the likelihood or consequences of a malfunction of any SSC's important to safety. The operators continue to have means of assuring drywell temperature remains within initial conditions assumed in the accident analysis; therefore, the previously evaluated dose consequences remain bounding. The TRM change does not create the possibility for an accident of a different type than previously evaluated in the USAR because the change does not involve or affect any possible accident initiator and no new failure modes are introduced. The change does not introduce the possibility of a malfunction of another SSC with a different result because the activity does not introduce a failure result. This TRM change does not cause any system parameter to change; therefore, it does not result in a DBLFPB as described in the USAR being exceeded or altered. This TRM change does not involve a method of evaluation used in establishing the design bases or in the safety analysis.

OLCR 2001-012
(USQE 2001-0017)- OLD RULE

TITLE: Technical Specification 3.6.1.3 Action Bases for Remote Manual Valve

DESCRIPTION: This Technical Specification Bases change added clarification to the Bases for TS 3.6.1.3 Actions stating that remote manual valves are considered "manual valves" (and thus, can be used to satisfy the Actions to isolate a penetration) once deactivated. Otherwise, a remote manual valve is not considered a manual valve. This clarification is based on a review of various regulatory references that discuss various valve types and classes. The clarification is being provided in the Action Bases for Specification 3.6.1.3, Primary Containment Isolation Valves.

10CFR50.59
EVALUATION: Bases clarification consistent with the Technical Specification requirements and existing intent will not increase the probability or consequences or a previously analyzed accident, create a new or different type of accident or malfunction, or decrease the margin of safety defined in the Technical Specification Bases.

OLCR 2001-014
(USQE 2001-0020)- OLD RULE

TITLE: "Critical Bus(es)" References Replaced With Class 1E Bus or Division(s)

DESCRIPTION: The NUREG-1433 Standard wording, for TS 3.8.1 Required Actions A.2 and B.2 Bases, which discusses the "redundant required feature" as being associated with "a division," was modified (during original conversion to the Improved Technical Specification format) to the CNS terminology "critical bus." These changes were called "editorial" and nomenclature" only. There is no documentation of intent to change the purpose or meaning. However, the CNS "critical buses" do not include all emergency powered Class 1E AC buses. This could lead to a potential unintended interpretation as to which redundant features are to be considered for compliance with Required Actions A.2 and B.2. As such, the Bases description associated with "redundant required feature(s)" for Required Actions A.2 and B.2 is being revised to replace "critical bus(es)" with the more encompassing terminology "Class 1E bus" and "division(s)" (consistent with NUREG-1433 terminology).

10CFR50.59
EVALUATION: This change provides that explicit description to eliminate possible confusion and provide uniform understanding. The Technical Specification requirement for the AC electrical power sources is unchanged and the required operability of the supported and associated redundant equipment continues to be required in accordance with the intent of the Technical Specification actions. The Bases level of detail is not identified as the initiator or mitigator of any event, nor does it impact any margin of safety. Therefore, these changes do not involve: (a) an increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR; (b) a possibility for an accident or malfunction of a different type than any evaluated previously in the USAR; or (c) a reduction in the margin of safety as defined in the basis for any technical specification.

UCR 2001-015 Change 20a and 22
(USQE 1998-0095)- OLD RULE

TITLE: Rebaseline of USAR Appendix A

DESCRIPTION: Change 20a provided clarifying information that original construction nondestructive examinations in excess of code requirements were performed to provide additional assurance of quality, but that these additional examinations were not part of the plant's design basis and are not required for the

current operation or maintenance of the plant. Deleted information on the installation, inspection, and testing of piping systems that exceeded was what necessary for compliance with the design code.

10CFR50.59
EVALUATION: The additional fabrication criteria described in Appendix A are not precursors to any design basis accident; therefore, there is no increase in the probability of an accident previously evaluated in the USAR. Neither the operation nor the structural integrity of the piping systems is affected by the proposed change; thus, the consequences of an accident previously evaluated in the USAR are not affected. No credit for the fabrication techniques in excess of code requirements described in Appendix A was assumed in any accident analysis. Furthermore, piping systems are passive components. Therefore, removal of this detail does not increase the probability of a malfunction previously evaluated in the USAR. Neither the operation nor the structural integrity of the piping systems is affected by the proposed change; thus, the consequences of a malfunction previously evaluated in the USAR are not affected. The removal of this level of detail does not create the possibility of a new malfunction or accident. The additional fabrication requirements described in Appendix A do not define any operational limitations or margins of safety and therefore, do not form the bases for any Technical Specifications.

OLCR 2001-018
(USQE 2001-0029)- OLD RULE

TITLE: CFT Relay Testing (TSTF-205)

DESCRIPTION: This Technical Specification Bases change adds clarifying detail relating to the acceptability of testing a single contact of a relay during a Channel Functional Test (CFT) for verification of relay function. This clarification is consistent with past and current testing practices at Cooper Nuclear Station, and is further supported by NRC-approved changes to the Improved Standard Technical Specifications (ISTS). These approved changes to ISTS are documented in Technical Specification Task Force (TSTF) change documents. This specific change is associated with TSTF-205, Revision 3. (It is noted that TSTF-205 involved additional, independent, changes to the Technical Specification definitions. This portion of TSTF-205 was not addressed with this change, as it would require a License Amendment to adopt.)

10CFR50.59
EVALUATION: Bases clarification consistent with the Technical Specification requirements and existing intent will not increase the probability or consequences of a previously analyzed accident, create a new or different type of accident, or decrease the margin of safety defined in the Technical Specification Bases.

UCR 2001-034
(USQE 2001-0016)- OLD RULE

TITLE: Isolation Bypass Indication USAR Change

DESCRIPTION: This activity revises USAR Section VII-3 to clarify that the existing design of the Primary Containment Isolation System (PCIS) does conform to safety design bases 9b as described in the USAR and that no additional action is necessary as currently implied in the USAR. When considering the guidance provided in Regulatory Guide 1.47, those valves with the capability to have their isolation function manually bypassed are not required to have control room indication of the bypass condition since it is not expected to occur more frequently than once per year and then only when the system is not required to be operable. No additional actions are required to comply with Safety Design Basis 9b. The compensatory actions previously implemented will remain.

10CFR50.59
EVALUATION: Clarification of how the existing design of the PCIS conforms to safety design bases 9b does not increase the probability or consequences of a previously evaluated accident. Since this activity addresses only how the existing design of PCIS conforms to safety design basis 9b (without additional changes in design), the probability and consequences of malfunction of equipment important to safety are not increased. Since there is no change in the plant design or in the method of operation, the possibility of a different type of accident is not created. The only credible malfunction that could result from this change is an operator error based on misunderstanding of the bypass condition. However, single operator errors are already considered; therefore, there is no possibility of a different type of malfunction. This change does not affect any technical specification or the bases for any technical specification.

UCR 2001-045
(USQE 2001-0011)- OLD RULE

TITLE: Below Grade Areas

DESCRIPTION: This USAR change deleted the criteria for having full Reactor Equipment Cooling flow to the HPCI room fan coil unit when the unit is operating. An NPPD calculation has established that full flow is overly restrictive, and unnecessary for fulfillment of this safety support function. Surveillance procedures assure that the requisite amount of cooling flow is provided.

10CFR50.59
EVALUATION: Since adequate flow will be available and the HPCI fan coil unit will continue to be capable of performing its required function, this USAR deletion will not increase either the probabilities or consequences of previously evaluated accidents. Additionally, no new malfunctions are being introduced; therefore, the likelihood of safety equipment malfunctions, and the consequences thereof, are not increased. The possibility of a new accident or safety equipment malfunction is not introduced by this change. The HPCI fan coil unit will continue to perform its intended function; therefore, the margins of safety defined in the bases for the technical specifications are not reduced.

UCR 2001-053
Procedure 2.4PC (Revision 2)
(USQE 2001-0030)- OLD RULE

TITLE: USAR/Procedure Change – Drywell Humidity Indication for RCS Leakage

DESCRIPTION: The change involves revising USAR Section IV-10, Reactor System Leakage Rate Limits, and deleting an entry condition for Procedure 2.4PC. Specifically, the change results in the use of drywell relative humidity as a secondary indicator of potential Reactor Coolant System (RCS) leakage into the drywell, rather than as an additional primary indicator. Use of relative humidity as a secondary indicator results in it being monitored when one or more of the other direct instruments indicate the possibility of a leak.

10CFR50.59
EVALUATION: The CNS licensing basis already acknowledges that drywell humidity is not a parameter that is directly measurable, and that other more direct means are available. Since more diverse and direct indicators of excessive RCS leakage into the drywell are available, a scenario in which drywell relative humidity is the first or only means of indication is not credible. Therefore, the probability of an event involving a loss of Reactor Coolant Pressure Boundary (RCPB) integrity is not increased. The instrumentation does not play a role in post-event mitigation of an event involving the integrity of the RCPB and does not affect systems that perform event mitigation functions. Therefore, the consequences of previously evaluated accidents are not increased. The methodology of accomplishing the early detection of RCS leakage into the drywell has no direct

bearing on RCPB integrity; therefore, the probability of a malfunction of equipment important to safety is not affected. Because the early detection of potential RCPB leakage into the drywell is not compromised by this change, the consequences of a malfunction of this equipment important to safety are not increased. The possibility of a different type of accident or malfunction of equipment important to safety is not created. The margin of safety is the margin between the unidentified leakage rate level limits, to that level determined to be amenable to critical crack propagation in the RCPB. This safety margin is not decreased by this change.

UCR 2001-055
(USQE 2000-0049)- OLD RULE

TITLE: USAR Change to Adopt GIP-3 as Alternative Seismic Qualification Method for Mechanical and Electrical Equipment

DESCRIPTION: Sections VII, XII and Appendix C of the USAR have been revised to allow the use of Revision 3 of the Generic Implementation Procedure (GIP-3) as supplemented by NRC SSER No. 2 and NRC SSER No. 3 for the seismic design and verification of modified, new and replacement equipment as an alternative to the methods currently specified in the USAR for the seismic design of Seismic Category I equipment.

10CFR50.59
EVALUATION: The GIP methodology is an NRC-approved alternative method for demonstrating seismic adequacy of equipment. Relative to the existing CNS seismic licensing basis, the GIP method provides an equivalent or superior level of assurance that equipment will perform required safety functions during and after a seismic event. Accordingly, there is no increase in the probability or consequences of previously evaluated plant events. Similarly, there is no increase in the probability or consequences of previously evaluated malfunctions of equipment important to safety. No new possibility is created for a different type of malfunction or accident. Use of the GIP methodology will not reduce the plant margin of safety as defined in the basis for any technical specification.

OLCR 2002-002
(Evaluation 2002-0003)- NEW RULE

TITLE: Change to Technical Specification Bases Section 3.8.1

DESCRIPTION: Changed the definition of a qualified offsite circuit so that an offsite circuit must be capable of automatically supplying both critical buses to be considered operable. The current definition allows an offsite circuit to be considered operable if it can automatically supply one critical bus and the other offsite circuit is capable of supplying the other critical bus. This change will return the definition to what was in the Technical Specifications prior to implementation of ITS with Amendment 178 and will then agree with existing USAR statements.

10CFR50.59
EVALUATION: This change results in the Technical Specification Bases being consistent with the USAR description and no physical change is being made to any SSC. Since the Technical Specification Bases and the USAR will be consistent and no SSC is being changed, there is no change to the frequency of occurrence, likelihood of occurrence, or consequences of an accident or SSC malfunction previously evaluated in the USAR and does not impact any fission product barrier. Being consistent with the USAR does not create the possibility of an accident of a different type or for an SSC malfunction with a different result than any previously evaluated in the USAR. This activity does no change any evaluation method descried in the USAR.

ATTACHMENT 4

SUMMARY OF REGULATORY COMMITMENT CHANGES

The following regulatory commitments were revised based upon evaluations performed in accordance with NEI 99-04, "Guidelines for Managing NRC Commitment Changes." This summary does not include those commitment changes where separate NRC notification was provided.

Source of Commitment: Letter from G. Trevors (NPPD) to NRC, dated June 8, 1988, "NPPD Response to NRC Inspection Report No. 50-298/88-13" (CNSS887289).

Revision of Commitment: In CNSS887289, NPPD stated that Procedures 0.27 and 1.8 will be revised to clearly indicate that the performance of a documented operability review will be required prior to the issuance of a non-essential part for an essential application. Since this time CNS Procedure 0.27 was deleted and requirements concerning operability evaluations were incorporated into CNS Procedure 0.5.OPS. The requirements that were incorporated into Procedure 1.8 concerning the issuance of non-essential parts to an essential application are being removed and incorporated into CNS Procedure 0.40.

Source of Commitment: Letter from G. Trevors (NPPD) to NRC, dated August 23, 1989, "NPPD Response to NRC Inspection Report No. 50-298/89-22."

Revision of Commitment: In this Notice of Violation response, NPPD stated:

"The existing methods of ensuring that procedure changes and training instructions for on-the-spot changes (OSC) to design changes are performed prior to required system operability will be reviewed and enhancements will be made as required, by December 1, 1989."

This was revised to:

"The existing methods of ensuring that procedure changes and training instructions for revisions to plant modifications are performed prior to required system operability will be reviewed and enhancements will be made as required."

The revised wording reflects the evolving Configuration Control Process. Specifically, OSCs are not allowed to change affected documents (i.e., procedures, training, etc.). These types of changes require a revision to the CED. Therefore, the existing commitment language is confusing and requires revision. Furthermore, "design change" is now generically referred to as "plant modification."

Source of Commitment: Letter from M. Peckham (NPPD) to NRC, dated November 27, 1996, "Inspection of Core Spray Spargers and Piping" (NLS960198).

Revision of Commitment: This letter committed to submit flaw evaluations of the Core Spray piping and spargers to the NRC for their review and approval. In an NRC Safety Evaluation, dated March 3, 2000, the NRC stated that future Core Spray piping and sparger inspections should be performed in accordance with the NRC Safety Evaluation dated December 2, 1999 for BWRVIP-18, "BWR Core Spray Internals and Flaw Evaluation Guidelines." Since that time BWRVIP-94 was issued which supersedes previous NRC reporting guidance contained in the BWRVIP Inspection and Evaluation Guidelines. Accordingly, the commitment was revised as follows:

"NRC reporting of Core Spray piping and sparger flaw evaluations shall be made in accordance with BWRVIP-94, "BWR Vessel and Internals Project Program Implementation Guide."

Source of Commitment: Letter from J. Swailes (NPPD) to NRC, dated September 30, 1998, "Response to the Systematic Assessment of Licensee Performance (SALP) Report, NRC Inspection Report No. 50-298/98-99" (NLS980156).

Revision of Commitment: The original commitment was to revise Abnormal and Emergency Procedures in accordance with the "1999/2000 Operations Procedures Betterment Project Plan," dated October 19, 1999. This Project Plan was revised and changed in scope. Accordingly, the commitment is revised to upgrade Abnormal and Emergency Procedures in accordance with the associated project plan and schedule. Considering the available resources and revised scope, the Project completion date was extended from December 31, 2000 to December 31, 2001.

Source of Commitment: Letter from J. McDonald (NPPD) to NRC, dated April 4, 2000, "Licensee Event Report No. 2000-006" (NLS2000030).

Revision of Commitment: NPPD committed to perform a multi-disciplinary assessment to determine where additional component-specific guidance/criteria is needed. If a task is identified that requires a procedure and it currently does not exist, a procedure will be generated to perform the identified task within 120 days following completion of the assessment. The assessment completion date was extended from August 3, 2000 to March 31, 2001 due to the Technical Training Programs being placed on INPO probation and the unanticipated Forced Outage due to Environmental Qualification issues, which diverted resources. Additionally, the commitment scope was revised for Maintenance to review tasks/Preventive Maintenance (PM) activities, routine or done only during specified plant conditions, to determine where additional guidance/criteria is needed. If a task is identified that requires a procedure and it currently does not exist, generate a procedure to perform the task within 120 days following the completion of the assessment. The intent of the original commitment was to ensure that Maintenance has proper and sufficient guidance for jobs which may be outside the "skill of the craft," and that the scope of the review include PM's, infrequently performed tasks, and routine complex tasks. The scope of review performed achieved these results. In lieu of a multi-disciplinary approach, Maintenance employed Craft, Specialists, Crew Leads, and Supervisors in this review. Equivalent results were obtained using this Maintenance hierarchical approach.

Source of Commitment: Letter from J. McDonald (NPPD) to NRC, dated April 4, 2000, "Licensee Event Report No. 2000-006" (NLS2000030).

Revision of Commitment: NPPD committed to develop a detailed Maintenance Procedure which includes guidance and criteria on how to properly inspect, repair, and align the torus-to-drywell vacuum breakers. The commitment due date was extended from August 3, 2000 to October 25, 2000 due to the unanticipated forced outage involving Environmental Qualification issues and conflicting resource requirements associated with Training Programs re-accreditation.

Source of Commitment: Letter from J. Swailes (NPPD) to NRC, dated November 14, 2000, "Proposed License Amendment Service Water Backup to the Reactor Equipment Cooling Post LOCA Response to Request for Additional Information" (NLS2000020).

Revision of Commitment: In Question 9 of the Request for Additional Information (RAI), the NRC asked that given the alarms indicated in the Amendment Request are not essential, what Control Room indications will be used to alert the operator of the need to initiate the SW/REC back-up. In the RAI response, NPPD provided a list of 42 sensors that would be available to alert the operators of the need to initiate the SW back-up cooling function.

CED 6008243 removes 4 of these sensors (ECCS pump area Fan Coil Unit flow switches REC-FS-478, 479, 480, and 481). The removal of the flow switches does not adversely affect the ability to monitor the REC System and identify the need to switch to SW backup of REC if needed. Also, removal of the flow switches does not significantly decrease the diversity of indication and annunciation available for monitoring the status of the REC System.

Source of Commitment: Letter from J. McDonald (NPPD) to NRC, dated November 15, 2000, "Licensee Event Report No. 2000-010-01" (NLS2000102).

Revision of Commitment: The original commitment was to replace the gaseous channel ratemeter on Control Building radiation monitor RMV-RM-1C with a new one by December 7, 2000. The completion date of this commitment has been extended to May 31, 2001. The spare gaseous channel ratemeter was found to require repair by the vendor. The commitment due date requires extension to allow return of the spare ratemeter to the vendor for repair. The ratemeter is to be returned by the vendor by May 15, 2001, thus allowing installation by May 25, 2001.

Source of Commitment: Letter from J. McDonald (NPPD) to NRC dated November 29, 2000, "Safeguards Licensee Event Report No. 2000-S01" (NLS2000107).

Revision of Commitment: NPPD committed to implement a requirement for the use of 2-part repeat back communication for Security operational crew personnel during the course of normal work by December 15, 2000. Full implementation of this commitment was changed to March 15, 2001. Training and use of 2-part communication was in place by December 15, 2000. However, full implementation requires capturing this commitment procedurally as well. This extension allows time for proceduralizing this commitment.

Source of Commitment: Letter from J. McDonald (NPPD) to NRC, dated March 12, 2001, "Licensee Event Report No. 2001-001" (NLS2001023).

Revision of Commitment: The commitment was to determine if NBI-CV-48BCV has a safety function or not; and if so, to determine an alternate method for verifying NBI-CV-48BCV operability and complying with Technical Specification SR 3.6.1.3.8 by May 31, 2001. The commitment due date was changed to November 3, 2001. It was determined that the valve has a safety function. However, additional time is required to complete a method of testing by inspecting this system in the drywell during the next refueling outage (RE 20). The extension is to provide time to determine and develop the optimum test methodology based upon drywell entry and inspection. Procedure(s) will be developed by November 3, 2001 for testing this valve.

Source of Commitment: Letter from J. Swailes (NPPD) to NRC, dated September 13, 2001, "Reply to a Notice of Violation NRC Letter No. EA-01-154" (NLS2001081).

Revision of Commitment: In this Notice of Violation response, NPPD committed to ensure the training associated with the corrective actions outlined in the letter is incorporated into the Emergency Preparedness (EP) and Emergency Response Organization (ERO) initial training programs by 11/29/01. This commitment was extended until December 14, 2001. This was justified since no new qualifications were scheduled until 2002. The revised schedule was to allow for incorporation of the new training requirements into the training program prior to the administration of the training to any new ERO candidates.

Source of Commitment: Letter from J. Swailes (NPPD) to NRC, dated September 13, 2001, "Reply to a Notice of Violation NRC Letter No. EA-01-154" (NLS2001081).

Revision of Commitment: In this Notice of Violation response, NPPD committed to establish and complete minimum training requirements for EP personnel by October 15, 2001. This commitment was extended to November 30, 2001. The training needs for EP personnel have been benchmarked against other utilities, resulting in the establishment of new training requirements. Only one of the five EP personnel does not meet the new requirements and a schedule for completion has been established.

Correspondence No: NLS2002122

The following table identifies those actions committed to by the Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the NL&S Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
None	N/A