Nebraska Public Power District
Nebraska's Energy Leader

NLS2000090
October 17, 2000

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

Gentlemen:

Subject: 10 CFR 50.59(b)(2) Summary Report
Cooper Nuclear Station
NRC Docket No. 50-298, DPR-46

In accordance with the provisions of 10 CFR 50.59(b)(2), the Nebraska Public Power District submits a summary report of facility changes, tests, and experiments completed in accordance with the requirements of 10 CFR 50.59. This report covers the time period from April 1, 1999, to July 31, 2000, and also includes those changes incorporated in the eighteenth revision of the Updated Safety Analysis Report. Also enclosed is a summary of commitment changes made during the same time period in accordance with Nuclear Energy Institute (NEI) 98-05 Revision 2, December 1995, "Guideline for Managing NRC Commitments."

Should you have any questions or comments regarding this report, please contact me.

Sincerely,

for
J. H. Swailes
Vice President of Nuclear Energy

/lrd
Enclosures

cc: Regional Administrator, w/enclosures
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The following table identifies those actions committed to by the District in this document. Any other actions discussed in the submittal represent intended or planned actions by the District. 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.	

ENCLOSURE 1

SUMMARY REPORT OF FACILITY CHANGES, TESTS, AND EXPERIMENTS COMPLETED IN ACCORDANCE WITH THE REQUIREMENTS OF 10CFR50.59

DESIGN CHANGES (DCs), MINOR MODIFICATIONS (MMs), MODIFICATION PACKAGES (MPs), CHANGE EVALUATION DOCUMENTS (CEDs)

DC 90-185

TITLE: Drywell Walkway Repair and Installation

DESCRIPTION: This design change provided for the repair of existing drywell walkways (the patching of holes cut into the existing grating), the replacement of temporary platforms/walkways with permanent platforms/walkways, the installation of new access ladders and platforms/walkways, and the addition of chain operators to certain manual isolation valves located in the drywell. The purpose of these changes was to improve accessibility to various valves and equipment within the drywell. The manual isolation valves are manipulated only during refueling outages, and the addition of the chain operators does not affect the safety function of the components. The structural changes made by this design change do not affect any safety function, system performance, or reliability of any component. Materials purchased for the modification are equal to or exceed the original material requirements. This modification was originally partially implemented during Refueling Outage 14, in 1993, with additional work performed during Refueling Outage 19, in 2000.

SAFETY EVALUATION: This change does not alter the redundancy, decrease the reliability nor degrade the structural integrity of any structure, system, or component. The new ladders, walkways, and walkway repairs are supported to seismic Class I requirements to ensure that Class II over Class I concerns are not created. The addition of the chain operators does not affect the pressure boundary or the safety function of the manual isolation valves. Accordingly, the probability of occurrence or consequences of an accident or malfunction of equipment important to safety has not been increased. Since the change merely augments current access ladders and walkways/platforms of the drywell, there is no degradation of the reactor building or drywell structural integrity. Thus, no new failure modes or effects are required to be postulated, and the possibility for an accident or malfunction of a different type has not been created. As stated previously, since this change does not alter the redundancy, decrease the reliability nor degrade the structural integrity of any structure, system, or component, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

DC 90-268

TITLE: Multi-Purpose Facility (MPF) Lighting Upgrade

DESCRIPTION: Under this design change, the following modifications were made to upgrade the lighting in the MPF:

- Portions of the existing 400 watt metal halide fixtures were replaced with 1000 watt metal halide fixtures.
- Portions of the remaining 400 watt metal halide fixtures were relocated to improve light dispersion.
- The existing 60 amp breaker providing power to the lights was spared out and a new 100 amp breaker, along with a larger feeder cable, was installed.
- A spare 100 amp breaker was installed for temporary power needs.
- Power hooks were installed on the remaining 400 watt metal halide fixtures to aid in repair and troubleshooting activities.

The MPF lighting circuits are supplied by nonessential power and the modifications meet National Fire Protection Association (NFPA) 70-1993, National Electric Code requirements.

SAFETY EVALUATION: This modification was implemented to address human factors concerns associated with the lighting in the MPF. The MPF is not a safety-related structure and the power for the lighting is supplied by the nonessential 12.5 kV distribution system. As the added load was confirmed to be within system

capacity and component ratings, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) have not increased. Since the modifications comply with National Electric Code requirements and no systems other than the Electrical Equipment system are affected, the possibility of an accident or malfunction of a different type has not been created. Neither the 12.5 kV distribution system nor the MPF lighting is addressed in the Technical Specifications. Therefore, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

MM 91-112

TITLE: Rod Position Information System (RPIS) Probe Connector Replacement

DESCRIPTION: Under this modification package, RPIS cable connectors manufactured by Deutsch were replaced with equivalent connectors manufactured by LEMO. The Deutsch connectors (which were provided with the original plant equipment) were found to have inherent design weaknesses that made them undesirable for the subject application over a long period of time. (Among other issues, the high number of coupling-uncoupling cycles for maintenance related activities caused the continuity through the pins to become unreliable.) The replacement LEMO connectors have design characteristics that will improve the long-term reliability and serviceability of the RPIS system.

SAFETY EVALUATION: The modification was performed during cold shutdown conditions. Although the refueling interlock for rod position was bypassed during implementation to permit the performance of other surveillance procedures, no core alterations were done concurrently with the connector replacement activities. Therefore, the probability of an accident previously evaluated in the SAR was not increased during implementation. Additionally, the control rods affected during the implementation process were verified to be fully inserted and the rod movement switch tagged out to prevent unmonitored control rod movement. These administrative controls ensured that the consequences of an accident were not increased during the connector replacement activities. Since the replacement connectors will improve long term reliability, the probability of an equipment malfunction previously evaluated in the SAR has not increased. Since the replacement connectors are equivalent in function, the consequences of an equipment malfunction would be unchanged. Likewise, since the function of the RPIS and the connectors is unchanged, the possibility of an accident or malfunction of a different type has not been created, and the margin of safety as defined in the basis for any Technical Specification has not been reduced.

MM 95-107

TITLE: Gaitronics Upgrade

DESCRIPTION: Under this modification package, permanent fused connection points were installed in the Gaitronics system to allow for temporary hookups on an as needed basis. Prior to this modification, any temporary structure (e.g., an outage support trailer) resulted in a formal modification package to install and subsequently remove Gaitronics coverage, which was an inefficient use of engineering resources. The zones affected by this modification were evaluated to ensure that potential loads added at the fused connection points will be within the capacity of the system.

SAFETY EVALUATION: The Gaitronics system is not an initiator of any accident, transient, or event. Therefore, the probability of an accident previously evaluated in the SAR has not increased as a result of this modification. Proper fuse coordination ensures that any localized fault will not adversely affect the remainder of the system, including the ability to notify site personnel via the public address/evacuation warning signal. Accordingly, the consequences of an accident previously evaluated in the SAR have not increased. Proper fuse coordination also ensures that the probability of occurrence and the consequences of an equipment malfunction previously evaluated in the SAR have not increased. In addition, proper fuse

coordination prevents any Gaitronics failure from affecting any other equipment, and a complete Gaitronics failure cannot create an accident. Therefore, the possibility for an accident or malfunction of a different type has not been created. Since the Gaitronics system is not addressed by the Technical Specifications, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

MP 92-021
(Unreviewed Safety Question Evaluation (USQE) 1998-0051)

TITLE: Kaman Flow Control Modifications

DESCRIPTION: The Kaman Radiation Monitors have had a history of unreliable operation. A significant number of failures were attributed to problems with the flow control systems for the monitors. Under Temporary Design Change (TDC) 92-021, the original flow control motor, valve, and flow transducer were replaced with an integral flow transducer and flow control valve on RMV-RM-30A (Radwaste/Augmented Radwaste monitor). This temporary modification provided favorable results. Therefore, under this modification package, the TDC was closed and permanent modifications were made to RMV-RM-3A and B (Elevated Release Point monitors), RMV-RM-30A and B (Radwaste/Augmented Radwaste monitors), RMV-RM-10 (Multi-Purpose Facility monitor), and RMV-RM-40 (Reactor Building monitor). The Kaman radiation monitors perform a monitoring function only and are classified as nonessential.

SAFETY EVALUATION: Since Kaman radiation monitors do not interface with any equipment or systems that could initiate an accident or transient, the probability of an occurrence of an accident previously evaluated in the SAR has not increased. The monitors are relied upon to provide information for assessing off-site dose. As such, improving system performance and reliability provides assurance that this function will be available when needed to mitigate the effects of an accident. Therefore, the consequences of an accident previously evaluated in the SAR have not increased. Improving system performance and reliability also provides assurance that the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR has not increased. As noted above, the monitors perform a monitoring function only and do not interface with any safety-related equipment. Since the function was unchanged by the modification, the consequence of a malfunction of equipment important to safety previously evaluated in the SAR has not increased. For the same reasons, the possibility for an accident or malfunction of a different type has not been created. In addition to improved performance and reliability, the process sampling method was changed from isokinetic to anisokinetic. Since the existing process flows are fixed, the fixed sampling flow provides more representative particulate, iodine, and gaseous sample results. Therefore, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

MP 95-049

TITLE: Turbine Equipment Cooling (TEC) Pumps Mechanical Seals Upgrade

DESCRIPTION: This modification package was generated to authorize replacement of the mechanical seals in TEC Pumps A, B, and C. The rotating face material is being changed from carbon to reaction bonded silicon carbide (RSC) and the stationary face material is being changed from ceramic to RSC. The seal O-ring is being changed from Viton to Ethylene Propylene. Work will be performed when any one of the TEC pumps mechanical seals is declared by the system engineer to be leaking excessively. This modification was previously reported (reference NLS970194, dated November 20, 1997) indicating that only the TEC Pump C had been upgraded. Since that time, the remaining TEC pumps (A and B) have been upgraded.

SAFETY
EVALUATION: The TEC system is not an Essential system and does not cool any Essential equipment. It is not an accident initiation or mitigation system for USAR evaluated accidents. The TEC system, although important for normal operation, is not needed for safe shutdown of the reactor. If the modified seals leak, the TEC head tank will make up the loss of water immediately. It would only cause leakage to the Turbine Building that would be contained by the floor and equipment drains and would be processed by radwaste treatment systems. Seals typically fail gradually with a slow leak; however, a major loss of water would cause an alarm to sound in the Control Room on low head tank level. If a pump seal leak is determined to be excessive, the affected pump will be shut down, valved out, and another pump will be placed in service without the loss of the TEC system. The modified seals should be more reliable and decrease the likelihood of problems with the TEC system. The TEC system is not mentioned in the Technical Specifications and there are no Technical Specifications that the TEC system directly affects. If the seals were to fail, the leakage would be less than the leakage from evaluated pipe failures.

MP 97-004

TITLE: Installation of Personal Computer Workstations

DESCRIPTION: Under this modification package, two personal computer workstations were installed in the plant to support field activities. The workstations function as look-up terminals for plant personnel to access station procedures and equipment data files. One workstation is located on the ground elevation of the Reactor Building, near Motor Control Center "Q," and the other in the Controlled Corridor that separates the Reactor Building and Turbine Building. The support stand for the workstation in the Reactor Building meets Class I seismic requirements, and the equipment is tethered to the stand to prevent interaction with safety-related equipment during a seismic event. The Controlled Corridor is a Class II seismic structure, therefore, the support stand for the second workstation is secured in accordance with plant housekeeping practices. Nonessential power for the workstations is provided by local lighting panels.

SAFETY
EVALUATION: The computer workstations function as look-up terminals for station procedures and equipment history, and cannot be used to operate or control plant equipment or systems. The equipment and support stands are installed per the governing seismic requirements and are powered by nonessential power. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR have not increased as a result of this activity. The cable and conduit modifications comply with National Electric Code requirements and the added electrical loads were confirmed to be within the rated capacity of the affected lighting panels. Therefore, the possibility of an accident or malfunction of a different type has not been created. As the workstations serve an administrative support function and have no interface with station equipment or systems, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

MP 98-021

TITLE: Addition of Two 100 Amp Fused Disconnect Switches to a 100 kVA Transformer Skid (Skid "J")

DESCRIPTION: This modification added two permanent 100 amp fused disconnect switches to a 100 kVA transformer skid (Skid "J") to provide power to temporary office trailers.

SAFETY
EVALUATION: Skid "J" is used to provide power to temporary office trailers and is supplied by the nonessential 12.5 kV ring bus electrical system. The installation is designed to the requirements of the 1996 National Electrical Code. The maximum calculated load added by this modification is adequately isolated from the remainder of the 12.5 kV system via the existing fuses in the 600 amp skid

disconnect. Though the 12.5 kV system does provide power for equipment defined as important to safety, it does not provide power for equipment defined as important to nuclear safety. Consequently, the loss of the 12.5 kV ring bus electrical system (e.g., from a fault associated with Skid "J") is not a plant event evaluated in the SAR and, hence, the probability of occurrence of accidents previously evaluated in the SAR has not been increased. Since the availability of the 12.5 kV system is not required for the mitigation of plant events described in the SAR, the consequences of previously evaluated accidents or equipment malfunctions have not increased. As noted above, the maximum calculated load added by this modification is adequately isolated from the 12.5 kV system via the existing fuses in the 600 amp skid disconnect. Therefore, the maximum potential voltage drop on the 12.5 kV system will not be affected by this modification should a fault associated with Skid "J" develop. For this reason, along with the fact that no safety-related equipment is supplied by the 12.5 kV system, the probability of occurrence of the malfunction of equipment important to safety is unaffected. For the same reasons, the possibility for an accident or equipment malfunction of a different type has not been increased. There is no margin of safety defined within the Technical Specifications regarding the nonessential 12.5 kV system. Accordingly, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1998-0011
(USQE 1999-0004)

TITLE: Dry Active Waste Shredder/Compactor Removal

DESCRIPTION: Under this CED, the shredder/compactor used for processing dry, radioactive waste was removed along with supporting area radiation and thermal fire detectors. For the past several years, the shredder/compactor was not used because of more cost effective methods available. This CED also addresses portions of the system removed under a previous maintenance work request. The shredder/compactor was removed to make the space available for other uses. Since radioactive materials are no longer processed or stored in the area, the area radiation monitors are no longer needed, and the remaining thermal fire detectors in the building are adequate to meet fire protection requirements.

SAFETY EVALUATION: The shredder/compactor was used to process dry, radioactive waste and had no safety function or impact on safe operation of the plant. Therefore, its removal had no impact on the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. Likewise, removal of the unused equipment does not create the possibility for an accident or malfunction of a different type. As this equipment was not addressed in the Technical Specifications, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1998-0060
(USQE 1998-0019, Revision 1)

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 taken under this CED to inject nitrogen into system high points at five locations in the Reactor Building. In doing so, 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.

SAFETY 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.

<u>CED 1998-0134 & CED 1998-0191</u>
(USQE 1998-0040 & 1998-0066)

TITLE: Appendix "R" Emergency Battery Lights Upgrade

DESCRIPTION: Under these CEDs, the existing Emergency Battery Light (EBL) units with a track record of unreliable operation and unsatisfactory vendor support were replaced with units demonstrating improved reliability.

SAFETY EVALUATION: The EBL units are provided to assist station personnel when responding to a postulated Appendix "R" fire event. Since the EBL units are not the initiators of any accidents, this modification does not increase the probability of a plant event previously evaluated in the SAR. The locations of Safe Shutdown equipment and egress routes that require emergency lighting have not been changed nor have the power supply and operating logic for the units been modified. As a result, the Emergency Lighting System continues to operate as originally designed, thus ensuring that the consequences of accidents previously evaluated remain unchanged. The new EBL units have been installed according to seismic requirements and the electrical loads connected to the lighting panels have been evaluated for the additional 15w per EBL unit loading. As a result, the probability and consequences of a malfunction of equipment important to safety previously evaluated in the SAR remain unchanged. The EBL units are not a source of a postulated plant event and, likewise, the failure of a lighting unit will not result in a postulated plant event. The transmitter used to activate the EBL for the monthly maintenance/inspection has been evaluated for areas with sensitive electronics and procedural controls have been established for its use. Accordingly, this modification does not create the possibility of an accident of a different type than evaluated in the SAR. As noted above, the additional electrical loads generated by the new EBL units have been evaluated and no changes have been made that would adversely affect circuitry or lighting levels. For these reasons, this modification does not create the possibility of malfunction of equipment important to safety of a different type than previously evaluated in the SAR. The EBL units and the normal plant lighting are considered to be nonessential and are not addressed in CNS Technical Specifications. The basis for the testing of the EBL units is

contained in Appendix R compliance procedures and is maintained by this activity. As a result, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1998-0140
(USQE 1998-0042)

TITLE: Westinghouse Office on the Turbine Operating Floor

DESCRIPTION: Under Plant Temporary Modification (PTM) 95-14, the Westinghouse outage support office was relocated from the north end of the turbine deck to the south end. The relocation was required to provide storage space for a spare low pressure turbine rotor. Under this CED, the installation was made permanent. In addition, this CED provided for added structural bracing, permanent power, phone and Gaitronics, and fire protection (i.e., a hand held extinguisher).

SAFETY EVALUATION: The Westinghouse office is a nonessential structure. It does not house, nor is it located in the vicinity of, equipment important to safety. The supplied power is from a nonessential source. For these reasons, the probability of occurrence of an accident or malfunction of equipment important to safety previously evaluated in the SAR has not increased. The structure is not used to respond to accidents or malfunctions, nor is it used to store equipment needed to respond to said events. The power is not provided by circuits supplying safety-related equipment, or equipment used to monitor or mitigate accidents or malfunctions (e.g., radiation monitors). Therefore, the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR have not increased. For the same reasons listed above, the possibility for an accident or malfunction of a different type has not been created. As the structure does not tie into, and is not located near any equipment or systems governed by the Technical Specifications, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1998-0197
(USQE 1999-0034)

TITLE: Depleted Zinc Oxide (DZO) Injection Skid Installation

DESCRIPTION: This modification package provided for the installation of the DZO Injection Skid, a subsystem of the Optimum Water Chemistry System. The Optimum Water Chemistry System functions to modify the chemical makeup of the reactor feedwater to protect the reactor internals and attached piping from Inter-Granular Stress Corrosion Cracking (IGSCC). The DZO system consists primarily of an American Society of Mechanical Engineers (ASME) Code Section VIII vessel (dissolution column) and associated isolation and flow control valving. During operation, water is diverted from the Reactor Feed Pump (RFP) discharge header, through the unit, and returned back into the condensate system, upstream of the RFPs. The diverted flow is manually varied between 4 and 100 g.p.m., depending on feedwater conditions. The installation of the DZO system does not affect the ability of the feedwater system to perform its intended function of providing coolant to the reactor vessel. The installed materials and equipment meet or exceed the original design requirements of the feedwater system. The DZO system has been placed in service. However, a few hardware adjustments are still required prior to package closeout.

SAFETY EVALUATION: The design of the DZO injection system is consistent with materials of construction, installation processes and design temperature and pressure of the feedwater system. This system is a passive system and its flow path does not adversely affect the performance of the feedwater system. The skid instrumentation is for indication only and does not interface with any plant power or control and instrumentation systems. There are no safety-related components that could be affected by a line break in the injection system. The use of depleted zinc oxide to reduce radiation levels in the plant has been shown in Electric Power Research Institute (EPRI) reports and through plant operations to be

compatible with the exposed equipment and materials. Therefore, the probability of an accident or the malfunction of equipment important to safety previously evaluated in the SAR has not increased. The feedwater system is not credited in any accident analysis and the diversion of flow from the feedwater going to the reactor vessel has no effect on the ability of the reactor vessel level instrumentation to respond to an accident (i.e., initiation of safety/Emergency Core Cooling System). Further, the systems required to mitigate the accidents are not affected by the zinc solution. Accordingly, the consequences of an accident previously evaluated in the SAR have not increased. The installation does not increase the consequences of a malfunction of equipment important to safety. The system itself is passive in nature and does not interface electrically with any system and it does not hydraulically affect any function of the feedwater system in response to an accident or transient. As noted above, the installation is consistent with system design pressures, temperatures and stress levels. However, because of the supply tap location, a reverse flow condition will develop through the Reactor Feed Pump "B" flow element if it is out of service. While this flow would be undetectable by Control Room personnel, there would be a small impact on the Core Thermal Power calculation. This impact was shown to be conservative. There would also be a small change in reactor vessel water level when operating in 3-element control. However, the level of change would be indistinguishable from other process input noise. The possibility for an accident or malfunction of a different type has not been created by the DZO system installation. As noted above, the skid is considered to be a passive component and any pressure boundary failure is bounded by existing analyses. Also, as noted above, the use of depleted zinc oxide to reduce radiation levels in the plant has been shown in EPRI reports and through plant operations to be compatible with the exposed equipment and materials. Finally, the installation of the zinc injection skid does not reduce the margin of safety as defined in the basis for any Technical Specification. There are no Technical Specifications related to the function of the skid and the chemistry parameters governed by the Technical Requirements Manual (i.e., reactor coolant system chemistry, dissolved oxygen concentration, conductivity, chloride concentration and pH) are unaffected.

CED 1998-0217
(USQE 1998-0067, Rev. 1)

TITLE: Overpower Relays Setting Change - EE-REL-1FE(32) and EE-REL-1GE(32)

DESCRIPTION: During the performance of the emergency diesel operability run, it was recognized that the overpower relay setpoint was below the required operations procedural requirements for surveillance testing. A revision to Nuclear Engineering Department Calculation (NEDC) 91-0220 was performed which changed the value from 100 percent to 115 percent of the normal 4000 kW loading. This CED raised the setpoint to the value in NEDC 91-0220 in order to ensure that spurious trips of the emergency diesel generator (EDG) do not occur during surveillance testing. This CED was included in the 10 CFR 50.59(b) Summary Report dated June 17, 1999 (NLS990061) but, as a result of comments from the off-site safety review board, the supporting USQE was revised. While additional justification was provided in the revision, the conclusions of the original USQE were not changed.

SAFETY EVALUATION: Performance of the change will not affect any of the factors that contribute to the probability of an accident since the buses, undervoltage protection, load shedding, and load sequencing circuitry that are important to mitigate an accident are not accident initiators. The relays affected by the setpoint change are used to protect the EDGs by isolating them from nonessential buses during surveillance testing should transients on the grid occur that could produce overloading conditions. Since the safety function and emergency response of the EDGs are unchanged by this modification, the probability and consequences of an accident previously evaluated in the SAR are also unchanged. The setpoint change has no effect on EDG reliability and, therefore, the probability of a malfunction of equipment important to safety has not been increased. As the setpoint change will also have no effect on redundancy or independence, the consequences of a malfunction of equipment important to safety have not been increased. The setpoint change will have no impact on the control logic of the overpower

relay. As a result, the setpoint change will not introduce any new or different types of accidents, abnormal occurrences, abnormal operational transients, special events, or design basis events. Since the effects of a relay failure have already been evaluated, the possibility of a different type of malfunction has not been introduced. As noted above, safety function and emergency response of the EDGs are unchanged by this modification. Accordingly, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1998-0225
(USQE 1998-0108, Revision 1)

TITLE: Appendix "R" Emergency Battery Lights Upgrade

DESCRIPTION: Under this CED, a portion of the Emergency Battery Light (EBL) units with a track record of unreliable operation and unsatisfactory vendor support were replaced with units demonstrating improved reliability. (See CEDs 1998-0134 and 1998-0191 for similar activities.) In addition to replacing existing units, one new EBL location was installed under this CED. While the replacement units are slightly larger and there is sufficient space for installation, there is not enough room to perform maintenance with the units installed. Accordingly, they must be removed and reinstalled once maintenance is complete. Initially, this was considered to be an acceptable condition. However, upon further review, it was determined that the EBL units supporting Station Blackout (SBO) should be relocated such that maintenance could be performed with the units in place. The CED and supporting USQE were subsequently revised to reflect this change. Under this revision, seismically qualified standalone supports were installed opposite the existing SBO EBL unit locations.

SAFETY EVALUATION: The EBL units are provided to assist station personnel when responding to a postulated Appendix "R" fire event. Since the EBL units are not initiators of any accidents, this modification does not increase the probability of a plant event previously evaluated in the SAR. The locations of Safe Shutdown equipment and egress routes that require emergency lighting have not been changed nor have the power supply and operating logic for the units been modified. As a result, the Emergency Lighting System continues to operate as originally designed, thus ensuring that the consequences of accidents previously evaluated remain unchanged. The new EBL units and support structures have been installed according to seismic requirements and the electrical loads connected to the lighting panels have been evaluated for the one new location and the additional 15w per EBL unit loading. As a result, the probability and consequences of a malfunction of equipment important to safety previously evaluated in the SAR remain unchanged. The EBL units are not a source of a postulated plant event and, likewise, the failure of a lighting unit will not result in a postulated plant event. The transmitter used to activate the EBL for the monthly maintenance/inspection has been evaluated for areas with sensitive electronics and procedural controls have been established for its use. Accordingly, this modification does not create the possibility of an accident of a different type than evaluated in the SAR. As noted above, the additional electrical loads generated by the new EBL units have been evaluated and no changes have been made that would adversely affect circuitry or lighting levels. Further, the added EBL and the support structures for relocating the SBO units comply with applicable seismic and loading requirements. For these reasons, this modification does not create the possibility of malfunction of equipment important to safety of a different type than previously evaluated in the SAR. The EBL units and the normal plant lighting are considered to be nonessential and are not addressed in CNS Technical Specifications. The basis for the testing of the EBL units is contained in Appendix R compliance procedures and is maintained by this activity. As a result, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1998-0239

TITLE: Security Barrier for Exterior Reactor Building Door R115

DESCRIPTION: Under this CED, modifications made under Temporary Design Change (TDC) 95-129 were accepted as permanent. The TDC was generated to address findings from an Operational Safeguard Response Evaluation (OSRE) and was previously reported (reference NLS970194, dated November 20, 1997). Although Reactor Building Door R115 had been previously welded shut under Design Change 88-239 to address security concerns, this modification was determined by the OSRE to be inadequate. As a result, additional concrete barrier slabs were added in front of the exterior door under the TDC.

SAFETY EVALUATION: As the concrete barrier slabs are placed outside and not attached to the Reactor Building, there are no essential or nonessential systems, structures, or components affected by this CED. Accordingly, the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR has not increased. Likewise, the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR have not increased. As the placement and weight of the slabs preclude them from becoming wind-driven missiles, the possibility for an accident or malfunction of a different type has not been created. Since the functionality of the Reactor Building (i.e., secondary containment) is unaffected, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1998-0268
(USQE 1998-0110)

TITLE: Addition of Throttling Control Capability for RHR-MOV-MO12A/B

DESCRIPTION: Under this CED, remote throttling control capability was provided for RHR-MOV-MO12A/B (heat exchanger A and B outlet valves, respectively). The Residual Heat Removal (RHR) system shutdown cooling mode is placed in operation during normal shutdown and cooldown. In the shutdown cooling mode with reactor pressure below 50 psig, reactor coolant is pumped by the RHR system pumps from one recirculation loop through the RHR heat exchanger(s) where heat is transferred to the service water system. Reactor coolant is then returned to the reactor vessel through connections to the recirculation loop(s). Temperature control is provided by varying flow rate in the RHR system and by use of the RHR heat exchanger bypass line. The valves used to control flow through the system during cooldown/heatup are the RHR outboard injection valve (RHR-MOV-MO27A/B) and the RHR bypass valve (RHR-MOV-MO66A/B). As an added option, the RHR heat exchanger outlet valve (RHR-MOV-MO12A/B) can be used to assist in controlling RHR system flow rate. During original plant design and operation, the cooldown rate was controlled by throttling the service water valves (SW-MOV-MO89A/B). It was discovered that, due to the erosive nature of service water, throttling SW-MOV-MO89A/B was not desirable and procedural changes were made to discontinue throttling SW-MOV-MO89A/B. However, RHR-MOV-MO12A/B were not originally provided with remote throttling capability, and prior to this CED, throttling was accomplished by locally de-energizing the power supply breaker when the valve reached the desired position. The new switches added under this CED provide remote throttling control capability from the Control Room without the need for extra Operators to be stationed at the associated motor control centers to de-energize the breakers when the valves are repositioned.

SAFETY EVALUATION: The replacement of the control switches in the Control Room for valves RHR-MOV-MO12A/B and at the Alternate Shutdown room for RHR-MOV-MO12B did not change the normal position or operation of these valves. The opening and closing rates are unchanged and no new automatic controls were added. Since the RHR-MOV-MO12A/B valves are not manipulated to support accident scenarios and are only used when the RHR system is aligned for the Shutdown Cooling mode of operation, the probability of an accident previously evaluated in the SAR has not been increased. Since the receipt

of a Low Pressure Coolant Injection (LPCI) flow initiation signal (reactor vessel low water level or drywell high pressure) when RHR is lined up for Shutdown Cooling will isolate the Shutdown Cooling injection line, the consequences of an accident or malfunction have not increased. As the new switches have the same voltage rating and current carrying characteristics as the previously installed switches, the probability of a malfunction of equipment important to safety as previously evaluated in the SAR has not increased. As noted above, the normal position and operation of these valves are unchanged, the opening and closing rates are unchanged, and no new automatic controls were added. Accordingly, the possibility for an accident or malfunction of a different type than any evaluated previously has not been created. While the Technical Specifications do address the operation of the RHR system in the Shutdown Cooling mode, there are no safety margins tied to the RHR heat exchangers or the heat exchanger outlet valves. Therefore, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1998-0270
(USQE 1999-0001)

TITLE: Removal of Miscellaneous Refueling Equipment

DESCRIPTION: This activity addresses several pieces of equipment associated with the Refueling Floor that were either removed or are no longer used as originally described in the Updated Safety Analysis Report (USAR). These items include the in-core neutron sources, the in-core flange seal test plug, and the fuel bail cleaner. The in-core neutron sources were used on initial core loading as the neutron source for initial reactor criticality and were removed during the fall 1976 refueling outage. An inventory taken for refueling service tools revealed the in-core flange test plug was missing from refueling area. As there are no approved procedures that require the use of this component, it will not be replaced. The fuel bail cleaner had the capability of being used in the fuel pool storage area, however this cleaner was never used due to the potential of a foreign material hazard created if the metal brushes were used above the channeled fuel. As all tools and servicing equipment necessary to meet the reactor general servicing requirements are supplied for efficiency and safe serviceability only, there are no safety objectives or requirements for the above listed components and their removal does not affect plant safety.

SAFETY EVALUATION: Per USAR Section 10.4, Tools and Servicing Equipment, all tools and servicing equipment necessary to meet the reactor general servicing requirements are supplied for efficiency and safe serviceability. These components are not part of, nor do they support, fuel movement activities. As such, there are no safety objectives or safety design bases identified with the servicing equipment being removed. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR have not increased. For the same reason, the possibility for an accident or malfunction of a different type has not been created. Since the use of the servicing equipment removed is not governed by the Technical Specifications, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0012
(USQE 1999-0077)

TITLE: Automatic Closure Modifications for CS-MOV-MO5A & MO5B

DESCRIPTION: This CED was generated to make modifications made under Temporary Design Change (TDC) 94-224 permanent. TDC 94-224 was developed to prevent spurious flow signals from closing the Core Spray minimum flow bypass valves CS-MOV-MO5A/B. The eight second time delay added to the closure circuitry under the TDC has been successful in preventing recurrence.

SAFETY
EVALUATION: Core Spray minimum flow bypass valves CS-MOV-MO5A/B are normally open valves that close after the Core Spray pumps start and a discharge path is established. The CNS SAFER/GESTR Loss of Coolant Accident (LOCA) Analysis and Nuclear Engineering Department Calculation (NEDC) 94-142 show that if the bypass valves did not close, the Core Spray pumps would still provide sufficient flow to protect the reactor core under the LOCA Design Basis Accident. The time delay relays used are essential components and divisional separation of the installation and wiring is maintained. The time delay function is not associated with the "open" function of valves. For these reasons, the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR has not increased. The use of the time delay function in the close circuit for CS-MOV-MO5A/B does not impact the ability of the Core Spray pumps to provide the minimum flow needed under the most severe Design Basis Accident, LOCA. Accordingly, the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the SAR have not increased. This time delay feature does not interface with any other control circuits or impact any interlocks associated with the valves. Further, the failure of the time delay relay is bounded by a failure of the valve to operate. Therefore, the possibility for an accident or malfunction of a different type has not been created. Finally, since the Technical Specification testing requirements for CS-MOV-MO5A/B are not affected by the time delay function, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0051
(USQE 1999-0022)

TITLE: Temporary Power for Reactor Equipment Cooling (REC) Heat Exchanger Cleaning

DESCRIPTION: Under this CED, a temporary welding receptacle was installed to facilitate cleaning of the REC heat exchangers. During the evaluation of a significant condition adverse to quality, it was determined that the welding receptacles installed in the Reactor Building had not been adequately evaluated for use during operating modes 1, 2, and 3. The temporary welding receptacle is powered from the nonessential 12.5 kV system which has no restrictions tied to the operating mode of the plant. Once the analysis to release the originally installed receptacles for use is completed, the temporary receptacle installed under this CED will be removed.

SAFETY
EVALUATION: The temporary welding receptacle is supplied by nonessential 12.5 kV power that is not an initiator of and has no impact on analyzed events. Therefore, the probability of an accident or the malfunction of equipment important to safety previously evaluated in the SAR has not increased. Since the installation does not interact with accident mitigation systems and the cable penetration for the temporary supply meets the applicable fire and secondary containment requirements, the consequences of an accident or the malfunction of equipment important to safety previously evaluated in the SAR have not increased. As the cable has been run in accordance with applicable separation requirements and the penetration seal is qualified for containment integrity, the possibility of an accident or malfunction of a different type has not been created. Since the 12.5 kV power system is not governed by the Technical Specifications and the secondary containment requirements have been maintained, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0063
(USQE 1999-0045)

TITLE: Floor Drain Inleakage

DESCRIPTION: Under this activity, the two major contributors of radioactive impurities to the Floor Drain Collector Tank were rerouted, resulting in a significant reduction in the liquid effluent generated by the Floor Drain Collector Tank system that is ultimately discharged into the Missouri River. The two sources

are: 1) the Elevated Release Point Sump (Z-Sump to Radwaste), and 2) the Radwaste Floor Drains from the High Integrity Container (HIC) dewatering discharge. These sources were rerouted to the Waste Collector Tank and Waste Sludge Tank, respectively. To provide for operational flexibility, the Z-Sump line includes a three-way valve so that the original drain path is still available as a backup to the Waste Collector Tank. This change in drain flow has no impact on the operation or availability of safety-related systems, structures, or components.

SAFETY EVALUATION: The materials used in the modification meet or exceed the design pressure rating of 150 psig, except the three-way valve which is rated at 125 psig. This rating meets the normal operating pressure requirements and is sufficiently above the maximum expected pressure of 20 psig that can be applied to the piping. Since the materials used meet or exceed normal operating pressure requirements and provide sufficient design margin above the maximum expected pressures, the probability of occurrence of an accident previously evaluated in the SAR has not increased. This change has no impact on the performance of the HIC dewatering discharge piping or Z-Sump, and the function of the Floor Drain and Radwaste systems have not been altered. Therefore, the consequences of an accident would be unchanged. From a safety perspective, the only equipment malfunction of concern would be the loss of Z-Sump which would make Standby Gas Treatment inoperable. The configuration of the three-way valve in the Z-Sump discharge piping will ensure that flow is maintained regardless of which processing tank is to be used and thus ensure that the probability of a malfunction of equipment important to safety previously evaluated in the SAR has not increased. The loss of Z-Sump has been evaluated and is bounding. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR have not increased. As noted above, the materials added meet or exceed normal operating requirements and provide sufficient margin above the maximum expected pressure, and the function of existing systems was not altered. Further, while operational flexibility was incorporated into the Z-Sump drain line, any failure would be bounded by existing analyses. For these reasons, a possibility for an accident or malfunction of a different type than previously evaluated in the SAR has not been created. This change has no effect on equipment or parameters governed by the Technical Specifications and, hence, has no effect on the margin of safety as defined in the basis of any Technical Specification.

CED 1999-0070
(USQE 1999-0060)

TITLE: HPCI ASD Flow Controller Enhancement

DESCRIPTION: This modification added a current-to-current (I/I) converter to the High Pressure Coolant Injection (HPCI) Alternate Shutdown (ASD) flow controller such that the control signal profile will match that of the primary HPCI flow controller located in the Control Room. Originally, the ASD controller provided a 4-20 ma signal while the Control Room provided a 10-50 ma signal. The HPCI governor is capable of operating on either control signal profile and was previously switched from one profile to the other, depending on the signal source. However, since the governor can only be calibrated to one control signal profile, it was difficult to establish and maintain acceptable as-left calibration readings for both loops.

SAFETY EVALUATION: The HPCI system is used for accident mitigation and is not considered to be an accident initiator. The I/I converter was added to the nonessential ASD flow controller circuit which is isolated from the essential portions of the HPCI control logic by a manual isolation switch. The systems, structures, and components (SSCs) that protect the barriers to radiation release will perform as previously analyzed. For these reasons, the probability and consequences of a previously analyzed accident have not been increased. The modification allows for the optimum calibration of the HPCI governor and eliminates one cable (i.e., previously used for switching the control signal profiles) included in the Appendix R Safe Shutdown Analysis Report. For these reasons, the probability of a malfunction of equipment important to safety has not been increased. As noted above, the I/I converter was added to the

nonessential ASD flow controller circuit which is isolated from the essential portions of the HPCI control logic. Therefore, the consequences of an equipment malfunction remain unchanged. Since the essential logic will remain unchanged, and loss of the ASD control signal is a bounded condition, the possibility for an accident or malfunction of a different type than previously evaluated has not been increased. This modification does not affect or require changes to the Technical Specifications. Accordingly, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0072
Operating License Change Request (OLCR) 2000-010
(USQE 2000-0007)

TITLE: Optimum Water Chemistry (OWC) System

DESCRIPTION: Modification package CED 1999-0072 provided for the installation of the Hydrogen/Oxygen Injection System, a subsystem of the OWC system. The OWC system functions to modify the chemical makeup of the reactor feedwater to protect the reactor internals and attached piping from Inter-Granular Stress Corrosion Cracking (IGSCC). This is done by injecting hydrogen into the reactor condensate/feedwater at the suction of the Condensate Booster Pumps. The injected hydrogen suppresses the radiolytic formation of hydrogen and oxygen in the core, thus reducing the concentration of oxygen, hydrogen peroxide, and other oxidizing agents in the reactor coolant. Outside the core, the hydrogen recombines with residual hydrogen peroxide and oxygen to form water. This changes the reactor coolant from an oxidizing environment to a reducing environment, mitigating IGSCC of vessel components. To offset the impact of excess hydrogen in the off-gas stream, oxygen is injected upstream of the Augmented Off-Gas (AOG) Recombiners.

Components installed under this CED include 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 generates and provides hydrogen and oxygen gas at the required injection pressures and flow rates. The Hydrogen Injection Module regulates the injection of hydrogen into the suction of the Condensate Booster Pumps. The Oxygen/Air Injection Module regulates the injection of oxygen or air into the off-gas system upstream the AOG Recombiners. The Condensate Oxygen Injection Module regulates the injection of oxygen into the condensate, and the Main Control Panel monitors and controls the OWC system operation.

To maintain consistent terminology, Section B3.3.6.1/2.d of the Technical Specification Bases was revised to define "full power background" as the radiation level corresponding to 100 percent power with hydrogen injection at the normal rate. This change was made under OLCR 2000-010.

The OWC system does not perform any safety-related functions and, with the exception of cable routed to the main Control Room, does not interface with any safety-related systems or components. The OWC system does interface with nonessential supply power, demineralized water, instrument air, feedwater flow signal, condensate booster pump logic circuits, Control Room annunciators, off-gas isolation signal, and Control Room Panel A. The hardware associated with the Hydrogen/Oxygen Injection system has been installed. However, pre-operational and start-up testing of the system are still required prior to package closeout.

SAFETY EVALUATION: OWC is a proven technology for the prevention of IGSCC crack initiation and growth in the recirculation piping and the lower vessel components. Prevention of IGSCC reduces the possibility of a large break or small break Loss of Coolant Accident (LOCA). The injection of hydrogen has no adverse effect on the condensate system, feedwater system, reactor recirculation system or the vessel and vessel components. The injection of oxygen maintains the dissolved oxygen concentration in a range consistent with fuel warranty requirements and the injection rate is adequate to prevent flow

accelerated corrosion. The design, installation, and testing of the OWC system are consistent with the existing systems' design codes and standards. In addition to solenoid valves, each injection pipe contains a check valve to provide an isolation boundary between the existing systems and the OWC system in case of an OWC system fault. For these reasons, the probability of an accident previously evaluated in the SAR has not increased. OWC does not adversely affect accident mitigation assumptions, equipment, or instrumentation. Since the main steam line radiation monitor alarm setpoints are based upon detecting an abnormal increase in reactor coolant activity as the result of fuel damage, the increased "background" radiation levels during normal operation as a result of the hydrogen injection will have no effect on the main steam line radiation monitor alarm function. Accordingly, the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR have not increased. As noted above, the design, installation, and testing of the OWC system are consistent with the existing systems' design codes and standards, and check valves provide system interface boundaries. The gas generation equipment is located in a separate building having a three-hour fire barrier rating and hydrogen leak monitors are provided at potential leak points. Therefore, the probability of a malfunction of equipment important to safety previously evaluated in the SAR has not increased. There are no new safety-related equipment interactions, failure modes, adverse operating scenarios or sequence of events created by the installation of the OWC system. Hence, the possibility for an accident or malfunction of a different type has not been created. OWC does not adversely affect primary containment oxygen concentration nor the thermal and mechanical properties of the fuel bundles. While there will be an increase in the main steam line radiation levels, the main steam line radiation monitor function is not affected. As a result, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0073
(USQE 1999-0037)

TITLE: Contractor Parking Lot Lighting

DESCRIPTION: Under this CED, permanent lighting was added for the parking lot designated for contractor personnel. Prior to adding lights, the lack of illumination in the parking lot was a personnel safety concern. The power is provided by the nonessential 12.5 kV system through an existing pole-mounted step down transformer. The combined existing and added loads are below the transformer rating and fault protection is provided by fused disconnect switches. A total of nine poles and twelve 400 watt lamps with photoelectric cells were added.

SAFETY EVALUATION: The nonessential 12.5 kV system is not an initiator of any analyzed accidents and does not provide power to any equipment important to safety. Therefore, the probability of an accident or the malfunction of equipment important to safety previously evaluated in the SAR has not been increased by this modification. Although the poles could become tornado-generated missiles, this consideration has already been factored into the design of the Seismic Class IS structures that house safety-related equipment. Therefore, the systems required to mitigate an accident (and thus the consequences of an accident) would be unaffected by this modification, even in the event of a tornado. As noted above, no equipment important to safety is powered by the 12.5 kV system and, therefore, the consequences of an accident or the malfunction of equipment important to safety previously evaluated in the SAR have not been increased should a loss of the 12.5 kV system occur. As the nonessential 12.5 kV system is not related to and does not supply electric power to any equipment important to safety, the possibility of an accident or malfunction of a different type has not been created. There are no Technical Specifications pertaining to the 12.5 kV system, therefore, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0074
(USQE 1999-0067)

TITLE: Optimum Water Chemistry Gas Generator (OWCGG) Building

DESCRIPTION: A nonessential concrete masonry block building was constructed to house the major components of the Hydrogen Water Chemistry System. The purpose of the building, referred to as the Optimum Water Chemistry Gas Generator (OWCGG) Building, and the associated equipment is to support optimum water chemistry for the primary reactor system. The building is located along the north Turbine Building wall and houses the nonessential Gas Generation Skid, Hydrogen Compression and Purification Skid, Oxygen Compression and Purification Skid, Rectifier Bank, and Control Panel. This CED provided for the installation/construction of the following: the OWCGG building structure, floor drains, air-conditioning unit, ventilation fans, louvers, fire dampers, heaters, interior/exterior lights, emergency lights, convenience power and OWC equipment power distribution. The building has no interface with any essential system, structure, or component (SSC) or any SSC used for safe shutdown. There is no interaction between the OWCGG Building and the Class II Turbine Building. The nonessential OWCGG Building was designed and constructed to meet the dead, live, wind, and seismic loads in accordance with the Uniform Building Code and Chapter XII of the USAR.

SAFETY EVALUATION: While the OWCGG building meets the requirements of a Class II structure, those components and portions of systems which can affect the probability of an accident are contained within a separate Class IIS structure. Therefore, the addition of the structure will not alter the probability of a Loss of Coolant Accident, Main Steam Line Break, Rod Drop, Recirculation Pump Seizure, or Refueling Accident. Since Class II structural requirements have been met, the probability of a transient which may be initiated by a wind generated missile such as a load reject with or without bypass or a station blackout event is not increased. Accordingly, the probability of an accident previously evaluated in the SAR has not increased. Since the structure has no interface with any essential components or equipment required for safe shutdown, mitigation of transients and accidents or special events, and since the structure does not alter any radiological barrier or release path, the consequences of an accident previously evaluated in the SAR have not increased. There is insufficient hydrogen stored in the building to create an explosive mixture and any small increase in fire potential due to the release of the contained oxygen in the building is restricted to the building. As the building is designed to provide a 3-hour fire barrier rating in accordance with National Fire Protection Association and the total volume of hydrogen or oxygen poses no creditable threat to the adjacent Turbine Building or nearby structures, the probability of a malfunction of equipment important to safety previously evaluated in the SAR has not increased. For the same reasons, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR have not increased. As noted above, the building has no interface with any essential SSC or any SSC used for safe shutdown, and there is no interaction between the OWCGG Building and the Class II Turbine Building. Further, the nonessential OWCGG Building was designed and constructed to meet the dead, live, wind, and seismic loads in accordance with the Uniform Building Code and Chapter XII of the USAR. Accordingly, the possibility for an accident or malfunction of a different type has not been created. Since the OWCGG Building and associated components installed under the CED have no relationship with or impact on any Technical Specification, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0077
(USQE 1999-0038)

TITLE: Temporary Leak Collection System for CW Inlet Valves

DESCRIPTION: The Circulating Water (CW) inlet valves to the main condenser have historically leaked water from the packing glands. In the past, this water was treated by the Radwaste System resulting in additional

operating costs. Under this activity, temporary equipment was added to existing taps on the packing glands to capture the leakage and route it back into the CW system. The result has been a significant decrease in the volume processed by the Radwaste System at a substantial dollar savings. This modification is considered temporary, until repairs to effectively stop the leakage can be made.

SAFETY EVALUATION: The interfacing boundary equipment used in this temporary modification has a design pressure rating well above that of the CW system, and this temporary alteration does not affect high energy lines. For these reasons, there will be no increase in the probability of any accident previously evaluated. Except for the capturing funnel, the system is a closed loop installation which would be unaffected by accidents or malfunctions previously evaluated in the SAR. Further, periodic sampling and surveys are conducted to ensure proper "radiological control" is maintained and to ensure the water being returned to the system is not contaminated above the required minimum levels. Accordingly, the consequences of accidents or malfunctions previously evaluated in the SAR have not been increased. The added equipment was attached to existing taps having manual isolation valves. As noted above, check valves with design ratings exceeding the CW system requirements were installed to assume the role of the boundary isolation. The operation of the CW inlet valves remains unaffected by the temporary modification. As a result, the probability of a malfunction of equipment important to safety previously evaluated in the SAR has not been increased. Since the system boundary has been maintained while capturing the leakage and diverting it back into the system, and the original analysis remains bounding with respect to equipment failure, the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR has not been created. This temporary modification has no impact on equipment governed by the Technical Specifications. Accordingly, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0081
(USQE 1999-0062)

TITLE: Mitigation Monitoring System (MMS) Installation

DESCRIPTION: This modification package provided for the installation of the Electrochemical Potential (ECP) Mitigation Monitoring System (MMS), a subsystem of the Optimum Water Chemistry System. The Optimum Water Chemistry System functions to modify the chemical makeup of the reactor feedwater to protect the reactor internals and attached piping from Inter-Granular Stress Corrosion Cracking (IGSCC). The function of the ECP/MMS is to monitor the effectiveness of the hydrogen injection and noble metal chemical deposition in the prevention of IGSCC. The ECP/MMS system primarily consists of two permanently mounted floor skids. The MMS taps into the nonessential portion of the Reactor Water Cleanup (RWCU) System outboard of containment isolation valves, downstream of the RWCU pumps and upstream of the regenerative heat exchanger, and diverts approximately 2 to 6 g.p.m. of the flow for monitoring purposes. The diverted flow is returned to a subcooling line that routes water from the outlet of the non-regenerative heat exchanger to an injection point between an outboard containment isolation valve and the inlet of the RWCU pump. This small diversion has an insignificant effect on the RWCU System ability to maintain primary water chemistry. The installed materials and equipment meet or exceed the original design requirements of the RWCU system. The MMS system has been placed in service. However, (1) a change to the method of operation, and, (2) verification of equipment performance during hydrogen injection must be completed prior to package closeout.

SAFETY EVALUATION: The 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 2 to 6 g.p.m. 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 1999-0082
Technical Requirements Manual Change Request (TRMCR) 1999-006
Special Procedure 99-003
(USQE 1999-0050)

TITLE: Noble Metal Chemical Addition

DESCRIPTION: This modification package provided for the implementation of the Noble Metal Chemical Addition of the Optimum Water Chemistry (OWC) System. The OWC System functions to modify the chemical makeup of the reactor feedwater to protect the reactor internals and attached piping from Inter-Granular Stress Corrosion Cracking (IGSCC). Under this CED, both the temporary installation of the equipment for chemical injection and the permanent aspect of the "treatments"are addressed. The NobleChem™ Application and Supporting Equipment, owned and operated by GE Nuclear Energy, consists of the following components: the NobleChem™ Injection System (drive water skid and injection skid); the Sampling System (sampling skid and ion chromatograph); the NobleChem™ Analysis System (inductively coupled plasma mass spectrometer); and the Data Acquisition System. This equipment is considered temporary and is removed after each chemical injection or "treatment." During a treatment, noble metal chemicals are injected into the reactor vessel via the Reactor Recirculation system. The noble metals deposited on the reactor vessel internals and recirculation piping work in conjunction with the depleted zinc oxide and injected hydrogen to lower the electrochemical corrosion potential (ECP) and stress corrosion cracking. The benefit of the NobleChem™ Application is that the targeted reduction of ECP is achieved at much lower levels of injected hydrogen (because of the catalytic behavior of the noble metals). This is desirable because the increased main steam line radiation levels due to hydrogen injection are also reduced. Special Procedure (SP) 99-003 was developed to support the NobleChem™ Application.

The first NobleChem™ treatment was performed during Refueling Outage 19, under Mode 3 (hot shutdown) conditions. In support of the process, changes to the reactor water conductivity and pH limits in the Technical Requirements Manual (TRM) were made under TRMCR 1999-006. (The noble metal chemicals are known to result in a temporary increase of conductivity and pH, typically for periods of time in excess of the limits specified in the TRM.) These changes were also addressed under this CED. Benchmark testing of the Noble Metal application will be performed after the hydrogen injection system is in service. This test is required prior to package closeout.

SAFETY EVALUATION: The USQE for this CED addressed all three parts of the NobleChem™ Application: the installation of temporary equipment necessary for the treatment; the temporary relaxation of the TRM chemistry limits; and the deposition of platinum and rhodium noble metals onto the wetted surfaces of the primary coolant system.

Installation of temporary equipment. The sampling skid is a nonessential passive monitoring system installed in the nonessential RWCU system. The installation was outboard of the containment isolation valves and the stainless steel hoses used to connect the sampling skid to the RWCU system are capable of handling the pressure and temperature requirements. Cooling water for the skid was provided by the nonessential portion of the Reactor Equipment Cooling system. The injection and drive skids were connected to the Reactor Recirculation system via instrument lines containing restricting orifices and an excess flow check valve. The manual containment isolation valves were placed under administrative control to provide redundant isolation capability. The connections to the Residual Heat Removal (RHR) system are also part of the containment boundary and were placed under administrative control. The temporary installation was done with the plant in a shutdown condition. For these reasons, the probability of an accident previously evaluated in the SAR has not increased. None of the skids have any relationship with or impact on any equipment used during an accident, transient or special event. Therefore, the consequences of an accident or malfunction of equipment previously evaluated in the SAR have not increased. All the skids (injection, drive, and sampling) and other equipment were seismically restrained. There was no impact on or relationship to any essential components other than the connection to the RHR instrument lines and the temperature and pressure rating of the connecting hoses greatly exceeded the conditions experienced during use. Therefore, the probability of a malfunction of equipment important to safety previously evaluated in the SAR has not increased. Since any leak that could have developed would have been bounded by the small break Loss of Coolant Accident analysis, the possibility of an accident or malfunction of a different type has not been created. Finally, as the temporary equipment has no impact on equipment contained in the Technical Specifications, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

Temporary relaxation of the TRM chemistry limits (TRMCR 1999-006). During NobleChem™ treatments, conductivity increases as a result of the cations and anions of the noble metal compounds. These ionic species are known not to be problematic with respect to crack initiation and propagation by IGSCC mechanisms. The reactor water pH changes are due to the hydroxide releases from the platinum compound and the nitrate releases from the rhodium compound. There are no initiators of transients or special events associated with or affected by the temporary changes in reactor water chemistry. Further, the temporary relaxation of limits during the injection period has no effect on the reactor coolant primary boundary or associated instrumentation. For these reasons, the probability of an accident or the malfunction of equipment important to safety previously evaluated in the SAR has not increased. The temporary changes do not alter the sequence of events for accidents, transients, or special events as described in the SAR. Since reactor water pH and conductivity have no influence on source terms, the radiological barriers and equipment used in the removal of radiological isotopes assumed in the SAR are unaffected. Accordingly, the consequences of an accident or the malfunction of equipment important to safety previously evaluated in the SAR have not increased. The change in chemistry limits does not change plant operations and no new equipment interactions are being created (except the temporary skids which have been evaluated separately). The change in chemistry limits does not create any new accident scenario or sequence of events. Although the corrosion rate may increase slightly, the consequences are insignificant due to the short time the limits are relaxed (approximately 48 hours). Therefore, the possibility for an accident or malfunction of a different type has not been created. Since the temporary change in the chemistry limits has no impact on any limiting condition for operation or safety limit, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

Deposition of platinum and rhodium noble metals onto the wetted surfaces of the primary coolant system. Noble metal chemical addition (NobleChem™) prevents the start of IGSCC and stops the growth of existing IGSCC within the reactor coolant pressure boundary. NobleChem™ does not adversely affect any contacted equipment. The injection of NobleChem™ induces a short-term increase in main steam line radiation upon return to power, and a second short-term radiation increase along with an increase in conductivity when hydrogen is first injected. These effects are nonrecurring

and produce no adverse consequences. The application of noble metals on primary surfaces cannot cause the initiation of a transient nor can it have any effect on the occurrence of special events. The effects on equipment important to safety within the reactor cooling system will be either negligible or beneficial. Therefore, the probability of an accident or the malfunction of equipment important to safety previously evaluated in the SAR has not increased. As the source terms do not change and the accident mitigation equipment will function as assumed in the accident safety analyses, the consequences of an accident previously evaluated in the SAR have not increased. Similarly, because radiation dose mitigation equipment will function as assumed in the safety analyses, the consequences of the malfunction of equipment important to safety previously evaluated in the SAR have not increased. NobleChem™ does not adversely affect plant operating conditions, plant operations, or equipment environment, operating ranges or loadings (e.g., stresses or pressure). Further, the NobleChem™ chemicals do not adversely interact with the contacted surfaces or equipment. Since no new equipment was added to the plant, there are no new equipment interactions, equipment failure modes, adverse operating scenarios or sequence of events created by treatment. As a result, challenges to equipment important to safety will not be increased and the transient analyses in the SAR will not be affected. For these reasons, the possibility for an accident or malfunction of a different type has not been created. The use of NobleChem™ has no effect on normal plant operations, and industry experience and testing to date demonstrates that NobleChem™ has no adverse effect on the thermal or mechanical properties of the fuel bundle components. Therefore, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0083
(USQE 1999-0041)

TITLE: HPCI-MOV-MO58 Insulation

DESCRIPTION: Upon reevaluation of the potential for pressure locking using conservative methodology and assumptions (reference Generic Letter 95-07), it was determined that HPCI-MOV-MO58 could become pressure locked under certain postulated conditions. In the event HPCI-MOV-MO58 becomes pressure locked, it would fail to open upon an Emergency Condensate Storage Tank low level signal. As a result, the High Pressure Coolant Injection (HPCI) pump would lose suction as the automatic transfer to the Suppression Pool would fail to occur. To alleviate this potential, a blanket of insulation was added to retard the valve heat-up rate to within acceptable values to ensure the HPCI system remains operable under required conditions.

SAFETY EVALUATION: The HPCI suction line, which includes HPCI-MOV-MO58, has been previously analyzed as not being an accident initiator per USAR Chapter XIV. Therefore, the possibility of an accident previously evaluated in the SAR has not been increased by this activity. The ability for HPCI-MOV-MO58 to be able to perform its safety-related function remains unchanged. By adding insulation to retard the valve's heat-up, the capability of the HPCI system to satisfy its safety function will be ensured for the duration of an event or accident. As a result, the consequences of an accident have not been increased. Adding insulation to the outside of the valve will not increase the valve's internal stress levels, nor will it increase the interconnecting piping system's stresses beyond code allowables. The addition of this insulation will also not result in adverse temperature conditions, internally or externally, ensuring that the valve's elastomeric components do not suffer from premature aging. Accordingly, the probability of a malfunction of equipment important to safety previously evaluated in the SAR has not been increased. Since the failure of HPCI-MOV-MO58 is bounded by existing analyses, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR have not increased. This activity does not create or delete any existing system interfaces, add or delete any equipment, components, or structures, or change the interface between a system and the plant's operators. Accordingly, the possibility for an accident or malfunction of a different type than any previously

evaluated has not been created. Since the performance, reliability, and function of HPCI-MOV-MO58 remain unchanged, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0086
(USQE 1999-0048)

TITLE: Evaluation of Civil/Structural Unauthorized Modification NAIT 2-20965-4

DESCRIPTION: Nuclear Action Item Tracking (NAIT) Identification Number 2-20965 Action Number 4, identified a potential unauthorized modification for the work performed on door R402, located between the Reactor Building Elevation 976' Clothing Change Area and the stairwell per Maintenance Work Request (MWR) 75-12-151. The work request per MWR 75-12-151 was to install a steel framed, expanded metal type door to replace the existing wood door and to provide a locking device. The "work performed" per MWR 75-12-151 indicated that the purchase order for the replacement door was disapproved. The fix was installing expanded metal in the existing door. However, field verification revealed that door R402 no longer exists. The only indications of the presence of a door are parts of the hinges and door closer that are still attached to the door frame. No documentation can be found for the removal of the door. This CED provides the required documentation to show that permanent removal of door R402 is acceptable.

SAFETY EVALUATION: Door R402 was never a part of the Secondary Containment boundary, and was not included in the Fire Hazards Analysis or the Internal Flooding analyses. It is not a Security barrier, nor does its removal have any affect on systems, structures, or components important for accident mitigation. Therefore, the door is considered to be an architectural component and its removal does not increase the probability or consequences of an accident previously evaluated in the SAR. For the same reasons, the probability or consequences of a malfunction have not increased. In addition to the above considerations, the door did not interface with any other plant piping or ventilation systems. Therefore, permanent removal of the door does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the SAR. Since Secondary Containment is not impacted by the permanent removal of the door, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0090
(USQE 1999-0042)

TITLE: Continuous Sampling from the CRD Pumps' Suction Piping

DESCRIPTION: Sampling lines and fittings necessary to hook up analysis hardware were added downstream of existing sample valves on the Control Rod Drive (CRD) suction lines to assist station Chemistry personnel in determining the cause for the premature clogging of the CRD suction filters. Following analysis, the sample flow is directed to the Reactor Building equipment drain system.

SAFETY EVALUATION: This activity does not affect the design, maintenance, or operation of any system, structure or component associated with accidents or transients previously evaluated, nor does it result in the operation, either continuously or intermittently, of any components, systems, or appurtenances beyond their intended limits. Consequently, the probability of an accident previously evaluated in the SAR has not been increased. Since this portion of the CRD system is not credited, either directly or indirectly, in any accident or event mitigation analyses, the consequences of any previously evaluated event or accident have not been increased. Further, a maximum sampling rate of 2 g.p.m. has been specified to ensure that the consequences of a Loss of Coolant Accident or a feedwater line break remain unchanged. The 2 g.p.m. maximum sampling rate also ensures that this activity does not result in the flooding of equipment important to safety. This, along with a specified weight limit for the

interconnecting hardware to ensure pressure boundary integrity of the sample lines, ensures that the probability of a malfunction of equipment important to safety has not been increased. This activity does not introduce any new failure modes, nor does it alter the design or operation of any existing plant equipment. Therefore, the consequences of a malfunction of equipment important to safety have not been increased. The possibility for an accident or malfunction of a different type has not been created because this activity does not alter the reactor core or any of its protective features, or change the interface between the CRD system and the plant's operators. Further, this activity does not alter the design or function of any components or alter their normal and accident parameters. Since the performance, reliability, and function of the CRD system remain unchanged, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0098
(USQEs 2000-0001, 2000-0002, and 2000-0003)

TITLE: Unauthorized Mechanical Modifications Resolutions

DESCRIPTION: As a result of review and field verification activities, several modifications to the facility were identified and classified as unauthorized. As such, these modifications were not previously evaluated per the requirements of 10 CFR 50.59. Under this CED, these modifications were documented and evaluated for safety impact. The following modifications addressed as part of this CED were subject to the requirements of 10 CFR 50.59.

1. Under Maintenance Work Request (MWR) 90-1916, additional valves were installed in the Service Air (SA) system downstream of SA-V-128 and SA-V-138. During field verification activities, it was observed that the valve downstream of SA-V-138 apparently had been removed at a later date. The valve downstream of SA-V-128 remained and was verified to have been incorporated into the necessary drawings and procedures as SA-V-623. The installed valve satisfies the design requirements of the SA system and is acceptable for use-as-is. SA-V-128 is a SA system station valve. Its purpose is to supply service air for general plant use or maintenance as needed. The SA system is a nonessential system that does not support any accident initiating or accident mitigating systems. SA-V-623 was originally added because SA-V-128 was leaking through.

2. During field verification activities, it was observed that a Service Water (SW) system valve was installed upstream of the Turbine Equipment Cooling Heat Exchanger (although the drawings show a flow instrument test connection at that location, no valve was shown). As the valve is compatible with the SW system and satisfies the design requirements of the system, it is acceptable for use-as-is. The flow instrument test connection was installed to support start-up testing and is no longer used. The valve is installed in a nonessential portion of the SW system.

3. During field verification activities, it was observed that a valve was installed on the discharge of the Reactor Water Cleanup (RWCU) Phase Separator Sludge Pump, just below the discharge flange. The 1/4 inch valve is rated for 3000 psig up to 400 degrees F, which exceeds the system design requirements. The brass material is compatible with the RWCU system, making the valve acceptable for use-as-is. The RWCU Phase Separator Sludge Pump is a nonessential pump. The pump came from the factory with a threaded hole in the discharge for the purpose of installing a pressure indicator or vent, if required. It is assumed that this valve was installed during initial plant startup.

SAFETY EVALUATION: Item 1 was evaluated under USQE 2000-0001. The improper functioning of the nonessential SA system is not a contributor to any analyzed accident listed in USAR Section XIV.6.0, but could potentially affect the Instrument Air system and contribute to initiation of some of the abnormal operational transients listed in USAR Table XIV-4-1. Since installing an additional manual valve at

a service connection downstream of a previously installed and normally closed manual isolation valve cannot result in improper functioning of the system, the probability of an accident or malfunction of equipment previously evaluated in the SAR has not been increased. Since the service connection is used only for general plant activities and maintenance, the addition of a second valve has no effect on the ability of any system or equipment to perform safety functions necessary to mitigate the consequences of analyzed accidents or malfunctions. Therefore, the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the SAR have not increased. As the valve satisfies the design requirements of the SA system, does not affect any system functions or the way in which the system is operated, and does not create any new credible failure modes for the affected equipment, the possibility for an accident or malfunction of a different type has not been created. Finally, there are no Technical Specifications associated with SA service connections. Therefore, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

Item 2 was evaluated under USQE 2000-0002. Since the nonessential SW header is designed to automatically isolate if the essential SW header pressure drops to a given setpoint, adding a valve to the test connection will have no effect on any safety-related system or analyzed failure modes for affected equipment. Additionally, no equipment important to safety interfaces with this valve. Therefore, the probability of an accident or malfunction of equipment important to safety as previously evaluated in the SAR has not increased. Since the added valve will not affect the ability of any system or component to perform safety functions necessary to mitigate the consequences of analyzed accidents or malfunctions, the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the SAR have not increased. As the valve meets the design requirements of the SW system, does not affect any system functions or the way the system is operated, and does not create any new credible failure modes for affected equipment, the possibility for an accident or malfunction of a different type has not been created. Finally, there are no Technical Specifications associated with the nonessential portion of the SW header. Therefore, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

Item 3 was evaluated under USQE 2000-0003. The improper functioning of the nonessential RWCU Phase Separator Sludge Pump will not initiate any accident or abnormal operational transient. Adding a valve for a vent or pressure indicator will not result in improper functioning of this system and does not affect any analyzed failure modes for the affected equipment. Therefore, the probability of an accident or malfunction of equipment important to safety as previously evaluated in the SAR has not increased. Since the added valve will not affect the ability of any system or component to perform safety functions necessary to mitigate the consequences of analyzed accidents or malfunctions, the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the SAR have not increased. As the valve satisfies the design requirements associated with the RWCU Phase Separator Sludge Pump, does not affect any system functions or the way the system is operated, and does not create any new credible failure modes, the possibility for an accident or malfunction of a different type has not been created. Finally, there are no Technical Specifications associated with the RWCU Phase Separator Sludge Pump. Therefore, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

<u>CED 1999-0110</u>
(USQE 1999-0047)

TITLE: RFPT "A" Steam Supply Drain Line Leak

DESCRIPTION: This CED provided for the temporary repair of a steam leak on the Reactor Feed Pump Turbine (RFPT) "A" high pressure steam supply drain line. The 3/4 inch drain line is not isolable during power operation. A clamping mechanism was placed around the affected area and sealant was injected to temporarily contain the leak. The clamping mechanism was left in place until permanent repairs

were made during shutdown conditions. The temporary repair was performed by a contractor having expertise with this repair methodology. The RFPTs and the associated drains are nonessential.

SAFETY EVALUATION: The temporary repair was made by a qualified contractor using a proven methodology. The repair clamp and sealant used meet or exceed the original piping material and strength requirements. The sealant material and an injection limit of six cubic inches were selected to ensure that the reactor chemistry requirements would not be exceeded, even if all of the sealant entered the main condensate system, and to minimize the possibility of blocking the drain line. Operations personnel were monitoring plant conditions during the repair and were prepared to remove the RFPT from service should abnormal conditions have developed. Additionally, the temporary repair does not affect the operation of the feedwater pump or feedwater control system. As a result, the probability of an accident or equipment malfunction previously evaluated in the SAR was not increased during or as a result of the temporary repair. Any event caused during or as a result of the repair would be bounded by the loss of feedwater and steam line break events, which are evaluated in the SAR. Therefore, the consequences of an accident or equipment malfunction previously evaluated in the SAR were not increased. The temporary repair had no effect on the function of the RFPT, and the materials used were compatible with the drain piping and boundary conditions. Further, the RFPT drains have no interface with the feedwater control system, and the volume of sealant was controlled to ensure chemistry limits would not be exceeded. For these reasons, the possibility for an accident or malfunction of a different type was not created. Since system operation was unaffected by the temporary repair and measures were taken to avoid exceeding chemistry limits, the margin of safety as defined in the basis for any Technical Specification was not reduced.

CED 1999-0111
(USQEs 1999-0080, 1999-0081, 1999-0082, 1999-0083, 1999-0084, and 1999-0085)

TITLE: Evaluation of Miscellaneous Mechanical Unauthorized Modifications

DESCRIPTION: As a result of review and field verification activities, several modifications to the facility were identified and classified as unauthorized. As such, these modifications were not previously evaluated per the requirements of 10 CFR 50.59. Under this CED, these modifications were documented and evaluated for safety impact. The modifications addressed as part of this CED are further described below.

1. Under Maintenance Work Request (MWR) 75-12-201, a ½ inch drain valve was installed in one of the instrument taps on each of the downstream flanges of flow elements FE-47A and FE-47B. The flow elements are located in the Aftercondenser Off-Gas line located in the Steam Jet Air Ejector Room. Although the installation operated without failure since 1975, the materials used did not comply with the design requirements for the Air Removal system. Accordingly, the valves and attaching pipe nipples were replaced under this CED with components meeting system requirements. During unit operation, air and noncondensible gases are collected in the main condenser, then exhausted through the steam jet air ejectors. The Air Removal system is classified as nonessential. The drain lines in question function only as a pressure boundary during system operation, and are available for draining any accumulated water from the line, if required, to support maintenance activities while the equipment is out of service.

2. MWR N79-12-4 was initiated because the plug in Fire Protection Header Vent valve, FP-V-217, was frozen in place and could not be removed. Under the MWR, the plug was successfully removed. However, since FP-V-217 is in a relatively inaccessible location, the piping was extended under the MWR to an accessible location and a second valve added. Procedurally, the existing valve was relabeled as a vent root valve and left in the open position. The new valve was numbered FP-V-949 and procedurally controlled as the vent valve. The piping and valve meet the necessary seismic qualifications and were confirmed to satisfy the design requirements of the

Fire Protection system. As a result, the modification is acceptable for use-as-is. The Fire Protection system is classified as nonessential.

3. Under MWR 76-8-191, an exhaust duct from the "TIP tent" was tied into the Reactor Building ventilation system. (The "TIP tent" is a term used to describe the metal building over the drive mechanisms for the Traversing Incore Probe (TIP) neutron monitors.) The installation consists of a filter and short ventilation duct run from the top of the TIP tent to an existing 22" x 10" exhaust duct in the Reactor Building. The duct was installed to maintain a negative pressure in the TIP tent. The gauge and material type of the sheet metal could not be verified without disassembly and testing (based on 22 years of operation with no signs of deterioration, this was not considered to be necessary). Since there is no essential equipment that could be affected by a failure of the added duct, seismic qualification is not required. The added exhaust duct effectively reduces the amount of air flow from other areas. However, the design requirements are associated with maintaining a slight negative pressure as opposed to satisfying predetermined flow rates, and these requirements are being maintained. Therefore, the installation is acceptable for use-as-is. During the walkdown it was determined that the filter was clogged. It was subsequently determined that the filter was not in any preventive maintenance program. Because of the filtering provided by the Reactor Building Ventilation system, the filter at the TIP tent was removed by this CED.

4. During field verification activities, two undocumented valves in the cooling water piping for Service Air Compressor 1C were identified. The valves are located on the supply and return Turbine Equipment Cooling (TEC) system lines to the compressor. No documentation could be found authorizing the installation of these valves. However, under MWR 79-4-108 temporary emergency cooling water lines were installed to this air compressor. It is assumed that the valves were installed under this MWR, but not removed as they are convenient for draining the TEC supply piping and the intercoolers of the air compressor. The valves exceed the design requirements for the TEC system and were confirmed to be compatible for use within the system. Accordingly, the installed valves satisfy the design requirements of the TEC system and are acceptable for use-as-is.

5. MWR N79-3-61 was issued to repair valve DW-V-73 or install a new valve downstream of valve DW-V-73. As a result, a new valve (DW-V-596) was added. Since no additional documentation was generated to support the installation of the new valve, this was considered to be an unauthorized modification. The valve exceeds the design requirements for the Demineralized Water system and was confirmed to be compatible for use with demineralized water. Accordingly, the installed valve satisfies the design requirements of the Demineralized Water system and is acceptable for use-as-is. DW-V-73 is used to supply demineralized water on as needed basis for general plant use or for maintenance. DW-V-596 provides additional isolation capability.

6. During field verification activities, it was observed that undocumented and unlabeled valves had been installed in various Demineralized Water and Service Air systems locations downstream of the isolation valves. Specifically, additional valves had been installed downstream of DW-V-124, DW-V-125, DW-V-127, DW-V-129, DW-V-147, DW-V-148, SA-V-126, and SA-V-137. The added valves meet the design requirements for, and were confirmed to be compatible for use in, the Demineralized Water and Service Air systems, as appropriate. However, the walkdown also revealed that some of the associated piping used in the Demineralized Water system was carbon steel (i.e., not acceptable for use in the Demineralized Water system). This piping was replaced with materials meeting the Demineralized Water system design requirements. With the replacement of the discrepant piping, the modifications satisfy the design requirements of the Demineralized Water and Service Air system, respectively, and are acceptable for use-as-is.

SAFETY
EVALUATION: Item 1 was evaluated under USQE 1999-0080. The improper functioning of the nonessential Air Removal system is not a contributor to any analyzed accident listed in USAR Section XIV.6.0. However, improper functioning could potentially cause the loss of condenser vacuum transient listed in USAR Table XIV-4-1. Since the installed drain valves serve as a pressure boundary only and will not cause the Air Removal system to function improperly, the probability of an accident or malfunction of equipment previously evaluated in the SAR has not been increased. As the valves are only used to drain condensate from the lines in support of maintenance activities when the equipment is out of service, there is no effect on the ability of any system or equipment to perform safety functions necessary to mitigate the consequences of analyzed accidents or malfunctions. Therefore, the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the SAR have not increased. As the existing and replacement valves do not degrade the performance or reliability of the Air Removal Air system, have no effect on the process flow, and pressure boundary integrity is maintained, the possibility for an accident or malfunction of a different type has not been created. Finally, the Technical Specifications only address the radiological limits associated with noble gases contained in the Air Removal system process flow. As the installed valves have no effect on these limits, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

Item 2 was evaluated under USQE 1999-0081. The improper functioning of the nonessential Fire Protection system will not initiate any accident, abnormal operational transient, or the malfunction of equipment important to safety. Since extending the vent line to an accessible location and adding a second manually operated valve will not result in improper functioning of the system, this change does not increase the probability of an accident or malfunction of equipment important to safety as previously evaluated in the SAR. Since the added piping and valve will not affect the ability of any system or component to perform safety functions necessary to mitigate the consequences of analyzed accidents or malfunctions, the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the SAR have not increased. As the modification complies with design requirements of the nonessential Fire Protection system and the vent is isolated except during system outages or maintenance, the possibility for an accident or malfunction of a different type has not been created. Since there are no Technical Specifications associated with the nonessential Fire Protection system venting configuration or operation, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

Item 3 was evaluated under USQE 1999-0082. The improper functioning of the nonessential portion of the Reactor Building Heating and Ventilating system will not initiate any accident, abnormal operational transient, or the malfunction of equipment important to safety. Therefore, the probability of an accident or malfunction of equipment important to safety as previously evaluated in the SAR has not increased. Since the Reactor Building ventilation radioactivity monitoring devices and the Secondary Containment isolation logic are unaffected by the modification, the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the SAR have not increased. The additional exhaust duct satisfies all design requirements, is qualified for the installed application, does not affect any system function or the way in which any system is operated, and does not create any new credible failure modes. Therefore, the possibility for an accident or malfunction of a different type has not been created. A review of the Technical Specifications and other licensing basis documents reveals that there are no defined margins of safety associated with the Reactor Building ventilation system. Therefore, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

Item 4 was evaluated under USQE 1999-0083. The improper functioning of the nonessential TEC system will not initiate any accident, abnormal operational transient, or the malfunction of equipment important to safety. The added valves meet the system design requirements and are maintained in the normally closed position. Therefore, the probability of an accident or malfunction of equipment

important to safety as previously evaluated in the SAR has not increased. As noted above, the valves comply with the system design requirements and are maintained in the normally closed position. Therefore, the modification will not affect the ability of any system or component to perform safety functions necessary to mitigate the consequences of analyzed accidents or malfunctions and, hence, the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the SAR have not increased. For the same reason, the possibility for an accident or malfunction of a different type has not been created. Since there are no Technical Specifications associated with the TEC system or the affected air compressor, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

Item 5 was evaluated under USQE 1999-0084. The improper functioning of the nonessential Demineralized Water system will not initiate any accident, abnormal operational transient, or the malfunction of equipment important to safety. Therefore, the addition of a second manually operated isolation valve does not increase the probability of an accident or malfunction of equipment important to safety as previously evaluated in the SAR. Since the added valve will not affect the ability of any system or component to perform safety functions necessary to mitigate the consequences of analyzed accidents or malfunctions, the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the SAR have not increased. As the added valve complies with design requirements of the nonessential Demineralized Water system and is normally maintained closed except during outage or maintenance activities, the possibility for an accident or malfunction of a different type has not been created. Since there are no Technical Specifications associated with the nonessential Demineralized Water system or associated service connections, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

Item 6 was evaluated under USQE 1999-0085. The improper functioning of the nonessential Demineralized Water system will not initiate any accident, abnormal operational transient, or the malfunction of equipment important to safety. The improper functioning of the nonessential Service Air system is not a contributor to any analyzed accident listed in USAR Section XIV.6.0, but could potentially affect the Instrument Air system and contribute to initiation of some of the abnormal operational transients listed in USAR Table XIV-4-1. Since the addition of second manually operated isolation valves downstream of previously installed and normally closed manual isolation valves will not result in the improper operation of either system, the probability of an accident or malfunction of equipment important to safety as previously evaluated in the SAR has not increased. Since the added valves will not affect the ability of any system or component to perform safety functions necessary to mitigate the consequences of analyzed accidents or malfunctions, the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the SAR have not increased. The added valves comply with the design requirements of the affected systems. Further, they do not affect any system function or how the systems are operated. Therefore, the possibility for an accident or malfunction of a different type has not been created. Since there are no Technical Specifications associated with Demineralized Water or Service Air system service connections, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

<u>CED 1999-0114</u>
(USQE 1999-0046)

TITLE: Feedwater Heater A-3 Relief Valve MC-RV-18RV Flange Leak Repair

DESCRIPTION: This CED provided for the temporary repair of a condensate leak from the inlet flange of the Feedwater Heater A-3 relief valve. The 1 inch by 1-1/2 inch relief valve is isolable when the A-3 heater is isolated and condenser vacuum is not required. A clamping mechanism was placed around the leaking flange and sealant was injected to temporarily contain the leak. The clamping mechanism was left in place until permanent repairs were made during shutdown conditions. The temporary repair

was performed by a contractor having expertise with this repair methodology. The feedwater heaters, including the associated condensate relief valves, are nonessential.

SAFETY EVALUATION: The temporary repair was made by a qualified contractor using a proven methodology. The repair clamp and sealant used meet or exceed the original piping material and strength requirements. The sealant material and the injection limit of 13.9 cubic inches were selected to ensure that the reactor chemistry requirements would not be exceeded, even if all of the sealant entered the condensate system. The relief valve function is to protect the A-3 heater tube side from an over pressure condition when removing the feedwater heater from or returning the feedwater heater to service. Therefore, a blockage of the relief valve would be acceptable under normal plant operation since A-3 heater can relieve pressure through the A-4 heater relief valve if needed. For these reasons, the probability for a loss of feedwater heating or a loss of feedwater flow as previously evaluated in the SAR was not increased during or as a result of the temporary repair. Any event caused during or as a result of the repair would be bounded by a loss of feedwater heating or a loss of feedwater flow. Therefore, the consequences of an accident or equipment malfunction previously evaluated in the SAR were not increased. The temporary repair had no effect on the function of the feedwater heater or condensate system, and the materials used were compatible with the relief valve and pressure boundary conditions. And, as noted above, the volume of sealant was controlled to ensure chemistry limits would not be exceeded. For these reasons, the possibility for an accident or malfunction of a different type was not created. Since system operation was unaffected by the temporary repair and measures were taken to avoid exceeding chemistry limits, the margin of safety as defined in the basis for any Technical Specification was not reduced.

<u>CED 1999-0117</u>
<u>Cycle 20 COLR</u>
<u>OLCR 2000-003</u>
<u>TRMCR 2000-001</u>
(USQE 1999-0091, Revision 1)

TITLE: Cycle 20 Reload (GE14, MELLL, ICF)

DESCRIPTION: This CED documented the analysis and design review associated with Reload 19 fuel bundles for Cycle 20 operation, and implemented the associated changes to the Core Operating Limits Report (COLR), the Technical Requirements Manual (TRM), and the Technical Specifications Bases. The evaluation was performed to ensure that Cycle 20 core design meets all of the applicable safety and regulatory requirements. Reload 19 fuel bundles are of the GE14 family of designs. General Electric (GE) provided reload design and licensing documents summarize the reload design and licensing calculations performed by GE in accordance with their NRC-approved methodology documented in NEDE-24011-13-P-A. In addition, this CED implements operation with Maximum Extended Load Line Limit (MELLL) and replaced the Flow Control Trip Reference Unit circuit cards in the Average Power Range Monitor (APRM) system to accommodate MELLL operation. The NRC granted an amendment on April 11, 2000 (TAC No. MA7705) revising the CNS Technical Specifications regarding the APRM neutron flux-high (flow-biased) allowable value, allowing CNS to operate in the MELLL domain, along the 121-percent rod line, and adding the Increased Core Flow (ICF) up to 105-percent flow domain. Therefore, the subject safety evaluation covers the configuration changes associated with the Cycle 20 reload, GE14 fuel type, MELLL, and ICF (although ICF has not yet been implemented), and the Cycle 20 COLR.

SAFETY EVALUATION: The Cycle 20 reload core thermal limits have been developed with NRC-approved methodologies as described in the General Electric Standard Application for Reactor Fuel (GESTAR) II and required by Technical Specification 5.6.5. The Cycle 20 reload core will not increase the probability of previously evaluated accidents because the analyses of the core have been carried out with NRC-approved methodology and have been correctly implemented as documented in the Cycle 20 COLR.

Similarly, the change to GE14 fuel and the implementation of the MELLL and ICF strategies were each analyzed in accordance with NRC-approved methodologies. Further the replacement of the APRM circuit card is equivalent in design and function to the previous component, with the exception of the added flow-biased rod block clamp. The added clamp is not an initiator for nor does it impact mitigation of any accident. The operating limits specified in the COLR ensure that the consequences of an accident or equipment malfunction previously evaluated in the SAR have not increased. Although the basis for the source terms used in the evaluation is still under review by the NRC, there is reasonable assurance that the source terms used will be bounding for approximately one operating cycle (as stated in the NRC Safety Evaluation Report associated with the Issuance of Amendment on Design Basis Accident Radiological Assessment Calculational Methodology Revision, TAC MA7758). Further, until NRC review and approval of the fuel handling accident analysis is completed, limitations on the movement of any irradiated GE14 fuel and on the movement of loads over irradiated GE14 fuel have been imposed. The analyses and thermal limits address all bounding previously evaluated core design related equipment malfunctions. The mechanical stability and maintenance of coolable geometry as required by General Design Criteria are assured by adherence to these core thermal limits for the evaluated core design related equipment malfunctions. Although specific analysis done in support of MELLL operation identified a peak drywell airspace temperature that was six degrees higher than the temperature assumed in equipment qualification, the short period of time at these higher temperatures has no adverse impacts on drywell equipment or the probability for equipment failure. The consequences of a malfunction of equipment important to safety previously evaluated in the SAR have not increased because the mechanical, thermal-hydraulic, and Loss of Coolant Accident design criteria imposed on the fuel and core maintain the level of protection previously achieved during anticipated operational occurrences and accidents. Further, the flow-biased rod block clamp added by the replacement APRM circuit card has no effect on the probability or consequences of a card failure. The possibility of a different type of accident is not created because this change does not allow for a new fission product release path, result in a new fission product release failure mode, or create a new sequence of events that results in fuel cladding failures or fission product release. In addition, the components added will not adversely affect the safety function of the APRM system (flow-biased high flux scram) nor will they impact the ability of the plant to cope with special events, as the added function (flow-biased rod block clamp) will not initiate events that could lead to an accident. Operation of a mixed GE9B and GE14 core has been evaluated using NRC-approved methodologies. In addition, evaluations show that the plant operating envelope can be safely extended to include the MELLL and ICF operating strategies. The equipment or systems required to load and provide safe operation for the Cycle 20 fuel are identical to those used in previous cycles. Other than for the addition of the flow-biased rod block clamp, no plant modifications are required to accommodate the new fuel or operating strategies. For these reasons, the possibility of a malfunction of a different type than previously evaluated has not been created. The margin of safety is defined by the thermal limits in the Cycle 20 COLR. Since CNS is operated within the thermal limits specified in the COLR, the margin of safety for each limit is not reduced. Additionally, the flow-biased rod block clamp function is not addressed in the Technical Specifications. Therefore, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

<u>CED 1999-0121</u>
<u>CED 1999-0122</u>
(USQE 1999-0052, Revision 1/USQE 1999-0057, Revision 1)

TITLE: 125 VDC Battery Cell Replacement/250 VDC Battery Cell Replacement

DESCRIPTION: A methodology was developed to allow for the replacement of one or two cells in the 125 VDC and 250 VDC batteries without de-energizing the battery. CED 1999-0121 addresses the specifics pertaining to the Division A and B 125 VDC batteries and CED 1999-0122 addresses the specifics pertaining to the Division A and B 250 VDC batteries. To summarize, the affected battery is temporarily modified by 1) connecting identical cells in parallel to the cells being replaced, 2)

disconnecting and replacing the degraded cells, and 3) removing the parallel cells, thus restoring the battery to the original qualified configuration. Since it was later determined that the seismic qualification of the affected battery could not be maintained for the duration of the 125 VDC battery cell replacement process, or for certain conditions associated with the 250 VDC battery cell replacement (i.e., depending on the 250 VDC battery cell replacement option required), the CEDs and supporting USQEs were revised to prompt the entry into the appropriate Technical Specification Limiting Condition for Operation (LCO) when required. It was also recognized that the inter-cell voltage (ICV) requirements may not be continuously met. Therefore, the aforementioned revisions also required entry into the appropriate Technical Specification LCO should the ICV limits not be continuously maintained. While the affected battery may not remain operable for the entire cell replacement process as originally intended, the revisions made to the CEDs ensure that the period of time the applicable Technical Specification LCO is in effect is minimized.

SAFETY EVALUATION: USQE 1999-0052 was developed to support CED 1999-0121, and USQE 1999-0057 was developed to support CED 1999-0122. 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. While the affected battery may be declared inoperable and the appropriate LCO entered for short periods of time due to seismic issues, the battery will remain energized for the duration of the activity. As this activity is only applied to one 125 VDC 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 serviced under this activity. Since all materials used in establishing the parallel configuration are safety-related, rated appropriately with respect to electrical design requirements, and seismically qualified, and since the electrical connection hardware will support battery operability in all critical parameters, the probability of an equipment malfunction previously evaluated in the SAR is not increased by 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. The interim configuration of the battery string plus the parallel set of cells will ensure the continuous performance of all required battery safety functions. 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. Since this activity was developed to support the replacement of cells that may degrade below Technical Specification limits, the objective of this activity is to maintain or restore the margin of safety as defined in the basis for any Technical Specification. Although implementation of this activity will require entry into a Technical Specification LCO (with the exception of one option available for the 250 VDC battery cell replacement activity), the duration is minimized such that the time requirements of the LCO are not exceeded. Since compliance with the established LCOs ensures that the margin of safety is maintained, the margin of safety as defined in the basis for any Technical Specification is not reduced by this activity.

<u>CED 1999-0144</u>
(USQE 1999-0075, Revision 1)

TITLE: Replacement of Valves SW-MOV-89A and B

DESCRIPTION: Under this CED, Service Water valves SW-MOV-89A and B were replaced with valves having improved throttling characteristics. In addition, spool pieces were installed as a contingency so that additional butterfly valves can be easily installed should the new valves degrade over time to the point that they can no longer perform their isolation function. The Residual Heat Removal Service Water (RHR SW) Booster system has two pumps in each of the two trains of Service Water (SW) providing

cooling water to one Residual Heat Removal (RHR) heat exchanger per train. The SW system back pressure and flow control is maintained by throttling the subject valves, located on the RHR heat exchanger outlet. As noted above, these valves also perform an isolation function when their respective RHR heat exchanger is not in service. Because of the high fluid velocity and erosive nature of the Service Water, valve erosion problems caused the valves to leak through after a short run duration (typically less than a month), making them unable to perform their isolation function. Further, the originally installed valves had been derated to an intermediate pressure valve. The valves installed under this CED comply with the original system design criteria.

SAFETY EVALUATION: The SW system and RHR SW booster system are required to support the accident mitigation function of the plant, but are not considered accident initiators in the USAR Chapter XIV Accident Analysis. Therefore, the probability of an accident previously evaluated in the SAR has not increased. Since the loss of one SW train is bounded by the assumed single failure of one Emergency Diesel Generator (EDG), the consequences of an accident previously evaluated in the SAR have not increased. Silting or plugging of the new valves is not considered a credible failure since the valve design includes a portion of the trim that has no pressure reduction stages, but 1-1/2 inch diameter holes to provide for full flow requirements. This, plus the restoring of the original system pressure design criteria, ensures that the probability of a malfunction of equipment important to safety previously evaluated in the SAR has not increased. Since failure of either valve is bounded by the failure of one EDG to start, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR have not increased. The failure of one of these valves is not considered an accident initiating event in the Chapter XIV Accident Analysis and no new active components were added. Therefore, the possibility of an accident of a different type than any previously evaluated in the SAR has not been created. Since the new components comply with the original system design specifications, the possibility of a different type of malfunction than any previously evaluated in the SAR has not been created. This is further supported by the fact that the accident mitigation function and the operation of the valves and the RHR SW booster system are not being changed. And, since the function and operation of the SW and RHR SW booster systems as described in the Technical Specifications has not changed, the margin of safety as described in the basis for any Technical Specification has not been reduced.

<u>CED 1999-0158</u>
(USQE 1999-0066)

TITLE: Temporary Installation of Stem Locking Device on CW-MOV-103MV

DESCRIPTION: CW-MOV-103MV is a 96" butterfly valve in the circulating water supply to the main condenser. The valve is used primarily during condenser backwashing activities. Under this CED, a temporary stem clamping device was installed on the shaft between the valve body and actuator for CW-MOV-103MV to minimize fatigue stress on other valve components by restricting the magnitude of valve disc movement and aligning the shaft until plant shutdown conditions permitted permanent repairs. When investigating excessive valve disc oscillations, the valve bushings were found to be excessively worn causing shaft-to-actuator misalignment, and the spline adaptor which mates the valve shaft to the actuator shaft was found to be cracked. These conditions contributed significantly to the observed oscillations and, if not checked, would have lead to actuator damage and possible valve failure. The clamping device, in effect, locked the valve shaft in place to prevent further degradation of the valve and actuator. Since the temporary clamping device locked the shaft in place, administrative controls were established to ensure the clamp was first loosened whenever valve actuation was desired. (The clamping device was to be subsquently replaced under CED 1999-0171 with an external bearing similar in design to the stem clamp. The new design would align the shaft and permit valve rotation without having to first loosen the clamp. However, Forced Outage 99-01 enabled actuator and spline adaptor replacement. The installation of a spacer ring in the packing area to provide adequate packing tightness was the only portion of CED 1999-0171 implemented after the Forced Outage repairs.)

SAFETY
EVALUATION: Although CW-MOV-103MV is classified as nonessential and has no safety function, mis-operation or valve failure could contribute to a loss of condenser vacuum, which is an analyzed operational transient. Clamp failure results in the valve returning to its operational condition prior to implementing this modification, and the consequences of not being able to remotely actuate the valve in case of an emergency is bounded by the loss of condenser vacuum analysis. Therefore, the probability of an accident previously evaluated in the SAR was not increased by this temporary modification. As noted above, CW-MOV-103MV is classified as nonessential and has no safety function. Therefore, the probability of the malfunction of equipment important to safety previously evaluated in the SAR was not increased. Since normal condenser operation is not affected by the modification and backwashing is not an activity required to mitigate an accident or transient, the consequences of an accident or equipment malfunction previously evaluated in the SAR were not increased. Since no new failure mechanisms were introduced by the clamping device, the possibility for an accident or malfunction of a different type was not created. As the operation of the Circulating Water system is not governed by the Technical Specifications, the margin of safety as defined in the basis for any Technical Specification was not reduced.

CED 1999-0164
(USQE 1999-0088)

TITLE: Turbine Generator Fluid (TGF) System Tubing Modification

DESCRIPTION: Under this CED, the TGF System tubing that supplies the turbine intercept and reheat stop valves was rerouted to the walls next to the low pressure (LP) turbines. Previously, the tubing ran over the top of both LP turbine casings, requiring its removal prior to removing the LP turbine covers for maintenance. Since a 5 to 10 day flush is required when the TGF tubing pressure boundary is broken and, previously, this flush could not be performed until the LP turbines were fully reassembled, there was a great potential to extend the scheduled time for the Turbine Generator outage. By rerouting the tubing such that it does not need to be removed to service the LP turbine casings, the duration of turbine maintenance outages could potentially be reduced by an estimated eight days. Additionally, under this CED, the fittings were replaced with ones more suited to repeated disassembly and reassembly. Due to repeated use, the original fittings were showing signs of wear and were often a source of minor leakage.

SAFETY
EVALUATION: The events that could be affected by this activity are a turbine trip with or without bypass, a generator load reject with or without bypass and turbine missile generation, or reactor feedwater controller failure. Since the modification meets original design requirements and has no adverse effect on valve stroke times, the potential for a turbine overspeed and resultant potential for missile generation has not increased. Likewise, the modification has no impact on the function or response of the bypass valves. Therefore, the probability of an accident previously evaluated in the SAR has not increased. The stop, governor, and bypass valve stroke times have not been altered, and the assumptions and conclusions concerning previously analyzed events remain valid and bounding. Accordingly, the consequences of an accident previously evaluated in the SAR have not increased. Equipment important to safety which could be affected includes the governor valve fast closure Reactor Protection System (RPS) pressure switches (which provide a trip signal to RPS) and fuel cladding. Since no physical alterations will be performed in the vicinity of the RPS pressure switches and the reconfigured tubing is physically separated from the RPS switches by a shield wall, the RPS switches remain unaffected by this activity. As noted above, the stop, governor, and bypass valve stroke times have not been altered. Therefore, the fuel reload analysis and core operating limits are unaffected. Accordingly, the probability of occurrence and the consequences of a malfunction of equipment important to safety previously evaluated in the SAR have not increased. The configuration and design are consistent with guidance provided in original design of the TGF system, hence no new failure modes have been created. Further, this activity also does not change any normal or accident parameter

for any plant equipment, nor does it add or remove any new equipment, processes, or interfaces. For these reasons, the possibility for an accident or malfunction of a different type has not been created. The Technical Specifications establish performance criteria and surveillance requirements for the turbine bypass valves. Neither the performance criteria nor the surveillance requirements will be affected by this activity. Therefore, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0170
(USQE 1999-0073)

TITLE: RW-LS-Z4 HI-HI Contact Bypass

DESCRIPTION: During surveillance testing of the "Z" sump, it was determined that the HI-HI switch in the control circuit for the Z1 sump pump was not operable. As a result, the operability of the Z sump was degraded and the associated Technical Specification Limiting Condition for Operation (LCO) was entered. To ensure the continued availability of the Z1 pump during the LCO period, the contacts for the HI-HI switch were bypassed by installing a temporary jumper until plant shutdown conditions permitted permanent repairs.

SAFETY EVALUATION: The Z sump is not an accident initiator and, therefore, the addition of the temporary jumper has no effect on the probability of accidents previously evaluated in the SAR. Since the temporary jumper prevented the Z sump from challenging other safety systems due to an inoperable HI-HI switch, the consequences of an accident previously evaluated in the SAR were not changed. The jumper did not restore sump operability, but did make the sump available for use during non-accident conditions. Additionally, the jumper was compatible with the circuit where it was installed. Since the jumper did not cause the sump pump to operate outside of normal design parameters, the probability of occurrence and the consequences of an equipment malfunction previously evaluated in the SAR were unaffected. As no new failure modes were introduced and the LCO requirements were being observed, the possibility of an accident or equipment malfunction of a different type was not created by the jumper. Since the jumper did not restore operability of the sump and the LCO remained in effect, the margin of safety as defined in the basis for any Technical Specification was not reduced.

CED 1999-0171
(USQE 1999-0069)

TITLE: CW-MOV-103MV Stem Shaft External Bearing Temporary Installation

DESCRIPTION: CW-MOV-103MV is a 96" butterfly valve in the circulating water supply to the main condenser. The valve is used primarily during condenser backwashing activities. Under CED 1999-0158, a temporary stem clamping device was installed on the shaft between the valve body and actuator for CW-MOV-103MV to decrease the magnitude of valve disc movement and align the valve shaft, following the discovered spline adaptor damage. While the clamp installed under CED 1999-0158 achieved the desired result of preventing shaft movement, the valve could not be operated without first loosening the clamp. Under this CED, the original clamp design was modified into an external bearing for the valve shaft. With the modified design, the valve shaft would be properly aligned but not locked in position. Intervention to first loosen the clamp would not be required to operate the valve. However, the external bearing was never installed because Forced Outage 99-01 allowed for the replacement of the actuator and spline adaptor. Satisfactory post-outage valve performance did not require the use of the external bearing. This CED also added a spacer ring in the valve stuffing box to compensate for the packing degraded by the excessive valve oscillations first addressed under CED 1999-0158. The spacer ring was installed during the Forced Outage repairs.

SAFETY
EVALUATION: Although CW-MOV-103MV is classified as nonessential and has no safety function, mis-operation or valve failure could contribute to a loss of condenser vacuum, which is an analyzed operational transient. Failure of the external bearing results in the valve returning to its operational condition prior to implementing this modification. Therefore, the probability of an accident previously evaluated in the SAR was not increased by this temporary modification. As noted above, CW-MOV-103MV is classified as nonessential and has no safety function. Therefore, the probability of the malfunction of equipment important to safety previously evaluated in the SAR was not increased. Since normal condenser operation is not affect by the modification, the consequences of an accident or equipment malfunction previously evaluated in the SAR were not increased. Since no new failure mechanisms were introduced by the external bearing, the possibility for an accident or malfunction of a different type was not created. As the operation of the Circulating Water system is not governed by the Technical Specifications, the margin of safety as defined in the basis for any Technical Specification was not reduced.

CED 1999-0210
(USQE 1999-0078)

TITLE: Isophase Bus Cooler Leak

DESCRIPTION: Under this CED, a temporary repair was made to the main generator isophase bus duct cooling coils. Cooling water (supplied by the Turbine Equipment Cooling (TEC) system) was found leaking from a pin hole at a supply tube joint. Since removing the cooling coil from service for repairs would require a power reduction to 50 percent, a temporary repair was made by clamping a rubber patch over the affected area. This temporary measure was left in place until permanent repairs were made during shutdown conditions. The isophase bus duct cooling coils and the TEC system are both considered to be nonessential.

SAFETY
EVALUATION: Neither the TEC nor the bus duct cooling systems are initiators for any abnormal transients described in the SAR. Therefore, the temporary repair to minimize TEC inventory loss and eliminate moisture concerns with the main generator isophase buses does not increase the probability of an accident previously evaluated in the SAR. Since neither the TEC system nor the main generator isophase buses are relied upon to mitigate any transients or accidents as analyzed in the SAR, the consequences of an accident or equipment malfunction are unchanged. The rubber material used to make the temporary repair is compatible with the tubing material, so there was no potential for the further degradation of the cooling coil. Therefore, the probability for a malfunction of equipment important to safety as previously evaluated in the SAR has not increased. As this temporary repair did not affect the operation of the TEC system or the bus duct coolers, the possibility for an accident or malfunction of a different type was not created. Since neither system is included in the Technical Specifications, the margin of safety as defined in the basis for any Technical Specification was not reduced.

CED 1999-0219
(USQE 1999-0087)

TITLE: Fire Protection Sprinkler System 35, High Temperature Sprinklers

DESCRIPTION: Under this CED, all 175 degrees F and 220 degrees F sprinkler heads in Fire Protection Sprinkler System 35 (23 in total) were replaced with 286 degrees F high temperature sprinkler heads. The sprinkler heads were replaced to decrease the effects of a High Energy Line Break (HELB) of a High Pressure Coolant Injection (HPCI) steam line in the Injection Valve Room. Had this particular HELB occurred, the Fire Protection system would have actuated because of the lower temperature sprinkler

heads previously used and the Environmentally Qualified (EQ) equipment in the northeast corner of the Reactor Building would have been sprayed down. The higher temperature sprinklers will not actuate during this postulated HELB scenario, thus ensuring the EQ equipment in the area will remain functional.

SAFETY EVALUATION: The Fire Protection system is not an initiator of any accidents, abnormal transients or other events evaluated in the SAR. As this modification did not change any of the initial conditions or assumptions relative to accidents, abnormal transients or other events evaluated in the SAR, the probability of an accident previously evaluated in the SAR has not increased. Since the Fire Protection system is not relied upon to mitigate the consequences of an accident, the consequences of an accident previously evaluated in the SAR have not increased. Both the function and the design basis for the Fire Protection system remain unchanged by this modification. In addition, this modification assures the reliability of EQ equipment in the event of the postulated HELB. Therefore, the probability of occurrence and the consequences of a malfunction of equipment important to safety previously evaluated in the SAR have not increased. Since all component failures which are analyzed bound any potential failures of the higher temperature sprinklers and no new failure modes are being introduced, the possibility of an accident of a different type than any previously evaluated in the SAR has not been created. And, since the reliability of EQ equipment in the event of a HELB of a HPCI steam line is assured, the possibility of a different type of malfunction than any previously evaluated in the SAR is not created. While the Technical Specifications do not address Fire Protection system requirements, the modification complies with the design parameters of National Fire Protection Association 13, 1976. Hence, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 1999-0236
(USQE 1999-0090)

TITLE: Removal of the General Electric Integrated Equipment Monitoring System (GEIEMS) Computer

DESCRIPTION: This CED removed the GEIEMS computer and related equipment. The GEIEMS was designed to monitor feedwater nozzle bypass leakage by analyzing Plant Management Information System (PMIS) data. It was originally installed by Design Change 90-360 to meet NUREG-0619 commitments. Due to advances in ultrasonic testing (UT), these commitments were revised (reference NLS990060 dated June 24, 1999). Since the GEIEMS computer system was not Y2K compliant, and since it was no longer required for feedwater nozzle bypass leakage monitoring, there was sufficient justification for system removal. The 16 feedwater nozzle resistance temperature detectors (RTDs) that provided input to the GEIEMS will continue to provide input to the PMIS computer and were unaffected by this modification.

SAFETY EVALUATION: The GEIEMS is a monitoring system that uses feedwater line nozzle and piping temperature data from the PMIS system. Since this data is obtained non-intrusively by externally mounted RTDs, the GEIEMS is not considered to be an accident initiator. As such, the probability of an accident previously evaluated in the SAR will not be increased by this activity. The consequences of an accident previously evaluated in the SAR will not be increased by this activity because data from the GEIEMS were not used for accident mitigation. As a result of advances in UT monitoring capabilities, the GEIEMS was no longer required for the detection of cracking on the inner surface of the feedwater nozzles. This, along with the fact that the GEIEMS was considered to be non-safety-related and the data gathered non-intrusively, assures that removal of the system has no effect on the probability of a malfunction of equipment important to safety. As noted above, the GEIEMS was classified as non-safety-related and functioned to analyze data gathered non-intrusively by PMIS. For these reasons, removal of the system has no impact on the consequences of a malfunction of equipment important to safety. For these same reasons, removal of the monitoring system does not create the possibility

for an accident or malfunction of a different type than any evaluated previously in the SAR. Since the condition of the feedwater nozzles will continue to be monitored, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 2000-0009
(USQE 2000-0005, Revision 1)

TITLE: Isolation of Drywell Fan Coil Unit (FCU)

DESCRIPTION: During normal plant operation, Drywell (DW) Fan Coil Unit (FCU) FC-R-1C developed a leak resulting in a forced outage. Under this CED, a temporary modification was implemented to isolate the FCU by installing blank flanges and return the plant to operation. (The FCU was subsequently repaired and returned to service during the following scheduled refueling outage.) Initially, the CED removed both the fan and the leaking cooling coils from service. However, as a result of localized "hot spots" in the Drywell, the CED and supporting USQE were revised to provide for the operation of the fan to increase air movement. The FCUs have no safety function and are not relied on to function in the event of a design basis event or accident. Cooling water for the FCUs is supplied by the non-critical loop of the Reactor Equipment Cooling (REC) system. The REC system is designed to isolate the non-critical loop during low system pressure conditions. The FCUs operate to maintain the average DW temperature at or below 150 degrees F. This limit ensures containment integrity and equipment operability during any event or accident scenario. If the average DW temperature limit is reached (e.g., as a result of a steam leak or loss of FCUs), actions are taken per established abnormal and emergency operating procedures to maintain the plant in a safe configuration.

SAFETY EVALUATION: With one FCU out of service, the average DW temperature will still be below the 150 degrees F limit and the failure of the blank flanges is bounded by a REC piping failure inside the DW. Therefore, the probability of an accident previously evaluated in the SAR has not increased. Since the FCUs are not relied upon during an accident and the average DW temperature will still be below the 150 degrees F limit, the consequences of an accident previously evaluated in the SAR have not increased. The increased loading will have no impact on the three remaining operational FCUs and the blind flanges used to isolate the cooling coils comply with system and seismic requirements. For these reasons, the probability of the malfunction of equipment important to safety previously evaluated in the SAR has not increased. Since no credit is taken during a design basis accident/design basis event for the operation of Drywell FCUs, and since one FCU out of service has no impact on the abnormal and emergency operating procedures, the consequences of the malfunction of equipment important to safety previously evaluated in the SAR have not increased. The possibility for an accident or malfunction of a different type has not been increased because, as noted above, the 150 degrees F average DW temperature limit has not changed and the installed blind flanges meet REC system and seismic requirements. Finally, since the 150 degrees F average DW temperature limit has not changed, the margin of safety as defined in the basis of any Technical Specification has not been reduced.

CED 2000-0011, Revision 1
(USQE 2000-0004, Revision 1)

TITLE: Bypass Valve "Open Signal Permissive" Alteration

DESCRIPTION: Under this CED, the control circuitry for the Main Turbine Bypass Valves (BPVs) was modified to ensure that accident and transient analysis assumptions will be valid at all levels of power operation. Per accident and transient analysis assumptions, the BPVs go to greater than or equal to 80 percent open within 300 milliseconds of a turbine stop valve closure signal on a turbine trip at any reactor thermal power level greater or equal to 25 percent. During original design and construction, the 25 percent permissive "setpoint" was established at 100 percent reactor thermal power using an input signal from the main generator output. However, due to increased inefficiencies in the main turbine

and balance-of-plant systems at lower power operation, the relationship between reactor thermal power and generator output is not linear. Field data taken at the original 25 percent generator load permissive level were found to correlate to 30.9 percent reactor thermal power. Consequently, the BPV feature of concern would not have been available until reactor thermal power exceeded 30.9 percent. To correct this non-conservatism, a resistor was replaced in the BPV control circuitry, thus electronically reducing the Open Signal Permissive from 25 percent generator load to approximately 16 percent of generator load. (Under Revision 1, this value was reduced to approximately 14 percent of generator load, which equates to 20 to 23 percent reactor thermal power, to provide added conservatism.) The new value is high enough to inhibit the function during low power turbine testing while assuring the assumed 25 percent permissive is available at required levels of power operation. Also, under Revision 1, a Control Room annunciator having redundant Control Room annunciation was modified to indicate when the BPV fast-open permissive is inhibited.

SAFETY EVALUATION: Since the BPVs and annunciation system are not initiators of any accidents, the probability of an accident previously evaluated in the SAR has not increased. Shifting the permissive setpoint in the conservative direction ensures that the consequences of an accident previously evaluated in the SAR have not increased and explicit indication of the permissive status will assist operators in responding to events or transients. The probability of a malfunction of equipment important to safety has not increased because the function and failure modes of the BPVs are unchanged and because the revised annunciator does not deviate from the original design considerations. Additionally, while the reliability of the control circuitry was not reduced, any failure of the control circuitry or loss of pressure control is bounded by the Digital Electro-Hydraulic Pressure Control Failure - Open. Shifting the permissive setpoint in the conservative direction also ensures that the consequences of a malfunction of equipment important to safety previously evaluated in the SAR have not increased. As neither the replacing of a resistor in the control circuitry nor the resulting shift in the permissive setpoint has an impact on the function or failure mechanisms of the BPVs, the possibility for an accident or malfunction of a different type has not been created. This is further supported by the fact that failure modes of the control circuitry and BPVs are bounded by existing analyses. The same is true for possible failures of the annunciator. Since shifting the permissive setpoint in the conservative direction ensures that accident and transient analysis assumptions will be valid at required levels of power operation, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 2000-0037
(USQE 2000-0009)

TITLE: Alternate Refueling Floor Shield Plug Laydown Arrangement

DESCRIPTION: As part of the original design of the facility, a refueling floor laydown arrangement and authorized "Safe Load Paths" for movement of heavy loads on the refueling floor were evaluated and established for use. During Refueling Outage 18, an alternate refueling floor laydown arrangement was developed and evaluated under Engineering Evaluation 1998-0059 (USQE 1998-0054). Since the alternate arrangement resulted in improved space utilization, this activity was initiated to make the change a permanent option within the Maintenance Procedures. The "Safe Load Paths" originally established are unchanged by this activity. USQE 1998-0054 was included in the 10 CFR 50.59(b) Summary Report for Cooper Nuclear Station dated June 17, 1999 (NLS990061).

SAFETY EVALUATION: The components affected by the rearrangement are currently shown to be stored on the refueling floor and cannot damage the fuel, reactor vessel, or any other structure, system or component. Accordingly, the components affected by the rearrangement are not considered to be accident initiators, and therefore the probability of an accident has not been increased. As the Reactor Building structure was confirmed to remain within design limits, the probability of a malfunction of equipment important to safety previously evaluated in the SAR has not increased. Since rearranging the components has no

impact on the operation of any system, structure, or component assumed to function during an accident or malfunction, the consequences of an accident or malfunction as previously evaluated in the SAR have not been increased. As this activity can only be implemented during outage conditions and since the existing fission product barriers are not affected, the possibility of an accident of a different type than previously evaluated has not been created. Also, since no new failure modes have been introduced, the possibility of a malfunction of a different type has not been created. There are no Technical Specifications associated with the arrangement of the refueling floor. This, along with the fact that no new "Safe Loads Paths" are proposed, assures that the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 2000-0059
(USQE 2000-0013)

TITLE: Replacement of Buchanan Terminal Blocks in Drywell

DESCRIPTION: Under this CED, eight Buchanan-0241 terminal blocks qualified to 311 degrees F were replaced with Weidmuller SAK-6N terminal blocks qualified to 450 degrees F. Although not complete at the time, it was anticipated that a reevaluation of the small break Loss of Coolant Accident analysis would result in higher peak drywell temperatures that would challenge the temperature rating of the existing components. As a result, seven terminal blocks in Main Steam Isolation Valve limit switch and solenoid operated valve circuits, and one terminal block in the RCIC-MO-MO15 power circuit, were replaced with units qualified to temperatures that would bound the anticipated results. Except for temperature qualification, the Buchanan and Weidmuller terminal blocks are equivalent in function and design.

SAFETY EVALUATION: This activity replaced existing terminal blocks with similar components having a higher qualification relative to their ability to withstand postulated post-accident environmental parameters. In that the design and function of the terminal blocks are unchanged, this activity does not affect any accident initiators, and the probability of an accident previously evaluated in the SAR has not increased. Since, as noted above, the new components are equivalent in design to the old, no new failure modes have been introduced by this activity. Accordingly, the probability of a malfunction of equipment important to safety previously evaluated in the SAR has not increased. As the new components are better able to withstand the effects of adverse conditions, the probability of any postulated failure of the terminal blocks is reduced. Therefore, the function and ability of any mitigating systems or equipment are also unchanged by this activity. For this reason, the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR have not increased. Further, as no new failure modes have been introduced, the possibility for an accident or malfunction of a different type has not been created. Finally, since the terminal blocks perform a passive electrical function, the replacement has no effect on the operation or performance of any Technical Specification equipment. Consequently, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 2000-0107
(USQE 2000-0025)

TITLE: Installation of Z1 Sump Hi-Hi Level Bypass Switch (RW-SW-Z4)

DESCRIPTION: Under this CED, a temporary modification was made to the Z1 sump control circuitry to return the sump to an operable status. During surveillance testing, the safety-related Hi-Hi level switch was found to be inoperable. Through investigation it was determined that the safety-related low-low level switch, operating from the same device, remained operable. As a result, a temporary bypass switch contact was installed around the Hi-Hi level switch contact and the Z1 sump returned to an operable status. By design, the Hi-Hi switch serves as a backup to the nonessential level control circuit. With

the bypass switch in the "closed" or "bypass" position, the Z1 sump pump cycles on the Low-Low level switch to control sump level; however, without the benefit of the Hi-Hi switch, the sump pump cycles more frequently. (This increased frequency was shown to be of no consequence to the design limits of the pump or motor.) During normal plant operation, the bypass switch is maintained in the "open" position and the sump level was controlled in the normal fashion by the nonessential circuitry. If the nonessential alternator circuit were to become inoperable, approved procedures directed Operations personnel to monitor the operability of the Z2 sump pump. If there was a concurrent failure of the safety-related control circuit for the Z2 sump pump, Operations personnel would then switch the bypass switch to the "closed" position, allowing the Z1 pump to control sump level using the safety-related Low-Low switch as described above.

This temporary modification was necessary to support continued plant operation. With one of the Z sump pumps inoperable, the corresponding Standby Gas Treatment system must be declared inoperable, placing the plant in a 7-day shutdown Limiting Condition for Operation. Since the Z sump is a potential hydrogen environment during normal operation, troubleshooting of the inoperable Hi-Hi switch would ideally be done during shutdown conditions.

SAFETY EVALUATION: The components used in the temporary modification were qualified for use in safety-related applications. In addition, procedural controls were developed and approved to support the temporary configuration change and engineering evaluation determined that the increased frequency in pump cycling was of no consequence to the design limits of the pump or motor. Accordingly, the probability of occurrence of an accident or malfunction of equipment previously evaluated in the SAR was not increased. Since the Z1 sump pump was returned to an operable status as a result of the modification, the consequences of an accident or equipment malfunction previously evaluated in the SAR were unchanged. During normal plant operations, the bypass switch was maintained in the "open" position under procedural control. Further, this temporary modification introduced no new automatic functions in the Z sump pump control circuitry. Therefore, the possibility of an accident or malfunction of a different type was not created. By returning the Z1 sump pump to an operable condition, the margin of safety as defined in the basis for any Technical Specification was maintained.

CED 2000-0114
Setpoint Change Request (SCR) 2000-16
TRMCR 2000-002
(USQE 2000-0019)

TITLE: PC-PS-119A, B, C, D Upgrade to EQ including SCR and TRMCR

DESCRIPTION: When evaluating discrepancies between in-containment temperature values used in the Environmental Qualification (EQ) program, temperatures higher than those previously assumed resulted. More specifically, temperatures resulting from a small break Loss of Coolant Accident (LOCA) exceeded those previously assumed. One method of minimizing the in-containment temperature during and following a LOCA is to manually initiate the Containment Spray Mode of the Residual Heat Removal (RHR) System. As such, actions were initiated to qualify the Containment Spray Mode of the RHR system. (This mode of RHR operation was not previously credited with accident mitigation and, therefore, components in the Reactor Building which may be exposed to a harsh environment were not included in the EQ program.) As a result of this effort, four pressure switches (PC-PS-119A, B, C, and D) and associated terminal blocks were identified as not being qualified under the EQ program. These pressure switches and terminal blocks were replaced under this CED with EQ qualified equivalent components.

While evaluating the Containment Spray Mode of the RHR system, a discrepancy relative to the Technical Requirements Manual (TRM) Allowable Value for the subject switches was identified. (Originally, a value of less than or equal to 2.0 psig was specified; the correct value is greater than or

equal to 2.0 psig.) The resolution of this issue, and the resulting setpoint and TRM change requests, were implemented in conjunction with this CED.

SAFETY EVALUATION: Since the Containment Spray Mode of the RHR system is designed for mitigation, the component upgrades and setpoint changed implemented by this activity cannot increase the probability of an accident previously evaluated in the SAR. As a result of this activity, the Containment Spray Mode of the RHR system is qualified to mitigate the consequences of a small break LOCA, and the revised setpoint provides additional margin to the -2 psig containment design pressure. Accordingly, the consequences of an accident or equipment malfunction previously evaluated in the SAR have not increased. The probability of an equipment malfunction previously evaluated in the SAR has not increased because the replacement components are equivalent in design and function; the setpoint change will add margin to containment integrity and reduce the probability that the containment will be inadvertently sprayed; and, the containment spray function will be available when required to assist in the mitigation of a LOCA. Since the pressure switches were replaced with equivalent components, no new failure modes have been introduced and the possibility of an accident or malfunction of a different type has not been created. The setpoint and TRM changes for the subject pressure switches provide additional design margin as noted above. Therefore, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

CED 2000-0118
(USQE 2000-0020)

TITLE: Cable Replacement for EQ Valves SW-MOV-650MV & SW-MOV-651MV

DESCRIPTION: As a result of qualification issues surrounding the discovery of degraded environmentally qualified (EQ) cable splices, an effort was initiated under CED 2000-0088 to replace existing splices with qualified Raychem splices. The scope of this effort included the splices in Service Water valves SW-MOV-650MV and SW-MOV-651MV. However, the actuator compartments for the subject valves are physically too small to accommodate the larger sized Raychem splices. To address this condition, the short section of 1000 VAC rated cable between the local terminal box and the motor for the subject valves was replaced with 600 VAC rated cable. (The 600 VAC cable has a smaller diameter and allows for a much smaller bend radius. As a result, the 600 VAC cable and splices can be accommodated in the actuator compartments.) The cable replacement was documented and implemented under CED 2000-0118.

The subject valves are powered by the 480 VAC distribution system. Although the design criteria for the 480 VAC distribution system calls for the use of 1000 VAC rated power cable, most of the end devices (including the subject valves) are only rated for 600 volts. To avoid changes to the overall system impedance, the new cables were cut to match the length of cables replaced. (The lengths of cable involved are approximately 4 feet for SW-MOT-650MV and 8 feet for SW-MOT-651MV.) The existing splices in the local terminal boxes were already included in the scope of CED 2000-0088.

SAFETY EVALUATION: The functions of the motor operated valves, the terminal boxes, cables and splices were unchanged by this activity. Additionally, the number of splices remained the same and the overall system impedance was unaffected. For these reasons, the probability of occurrence and the consequences of an accident or equipment malfunction previously evaluated in the SAR remain unchanged. Although the voltage rating for the power cables between the local terminal box and the subject valves was reduced to 600 VAC, this is the same voltage rating as the valve operators. Accordingly, no new failure modes were introduced and the possibility for an accident or equipment malfunction of a different type was not created. As the safety function of the subject valves was unchanged, the margin of safety as defined in the basis of any Technical Specification was not reduced.

SPECIAL PROCEDURES (SPs) AND SPECIAL TEST PROCEDURES (STPs)

STP 88-005 Amendment 4
SP 99-008

TITLE: Specimen Irradiation Capsule Test Program

DESCRIPTION: STP 88-005, Amendments 1-4, were written to document the installation and removal of test specimens (capsules) in the reactor core as part of a joint effort by General Electric Company (GE) and Nebraska Public Power District. The objectives of the program were to demonstrate the improved resistance to Irradiated-Assisted Stress Corrosion Cracking (IASCC) of special stainless steel materials and the improved corrosion resistance of Zirconium materials. Amendment 4 to the STP allowed the insertion of three new GE test capsules into the reactor core for up to three cycles. The specimen irradiation capsules contained samples of special stainless steels, Inconels, Zircaloys and Zirconium-based alloys, and boron carbide. SP 99-008 was written to replace STP 88-005, Amendment 4, to document removal of the last two test specimens from the vessel during Refueling Outage 19 which were previously installed as part of that STP.

SAFETY EVALUATION: Since the installation and use of the irradiation capsule and coupons does not affect the core operation, core monitoring, instrumentation of the plant, or safety systems, the probability of an accident or the malfunction of equipment important to safety previously evaluated in the SAR was not increased. Analyses showed that the installation of the test capsule did not result in violating any of the fuel system damage limits (FSDLs) contained in General Electric Standard Application for Reactor Fuel (GESTAR), and there is no significant difference between the original source holder design and the irradiation capsule that would affect plant performance. Accordingly, the consequences of an accident or the malfunction of equipment important to safety previously evaluated in the SAR were not increased. As stated above, the installation and use of the irradiation capsule and coupons does not affect core operation, core monitoring, or the instrumentation of the plant, and therefore does not create the possibility of an accident of a different type. Further, because the capsule is installed in the same manner as a standard source tube holder, the impact on the core support plate and the upper guide are acceptable and the spring forces are great enough to ensure the capsule will not dislodge during operation. Therefore, the activity does not create the possibility of a malfunction of a different type. Since the design failure point of the fuel rod is not changed by the installation and use of the irradiation capsule and coupons and the change does not result in violating any of the FSDLs contained in the GESTAR, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

SP 98-029
(USQE 1998-0081, Revision 1)

TITLE: Secondary Containment Differential Pressure Measurements

DESCRIPTION: This SP provided instructions for obtaining absolute pressure measurements at various locations within the Reactor Building, outside at grade level, and on the Reactor Building roof. The test was non-intrusive as measurements were taken using a portable pressure gage which was not connected to plant systems or equipment. The data was gathered to measure the stack effect on the Reactor Building. Prior to the SP, Procedure 6.SC.501, Secondary Containment Leak Rate Test, required the addition of a stack effect correction factor to ensure there is -0.25 inches water gage at the Reactor Building roof. The test data obtained by this SP was evaluated under Engineering Evaluation 1998-0124. The SP and supporting USQE were previously reported in the 10 CFR 50.59(b) Summary Report dated June 17, 1999 (NLS990061). As a result of comments from the off-site safety review board, the USQE was revised.

SAFETY
EVALUATION: The unit was in Mode 4 or 5 when this test was performed and core alterations were suspended while the roof hatch was opened. The test is non-intrusive and no components were installed in the plant. Therefore, the probability of occurrence or consequences of an accident or malfunction previously evaluated in the SAR were not increased. The user was in attendance with the test equipment to prevent it from interacting with essential plant equipment. Current plant procedures were followed to access the Reactor Building roof and to run Standby Gas Treatment. Therefore, no new types of accidents of equipment malfunctions were introduced. When the Reactor Building hatch is opened, Procedure 6.SC.502 directs Operations to declare Secondary Containment inoperable. As the Limiting Condition for Operation 3.6.4.1 was not met, core alterations were suspended. Since compliance with the Technical Specification Limiting Condition for Operation was maintained, the margin of safety as defined in the basis for any Technical Specification was not reduced.

SP 99-001, Revision 2
(USQE 1999-0109, Revision 1)

TITLE: Steam Jet Air Ejector (SJAE) Performance Testing

DESCRIPTION: Under this SP, on-line performance testing and monitoring procedures were implemented for the Steam Jet Air Ejectors (SJAEs). The SJAEs are used to draw noncondensible gases off the main condenser and have a direct impact on main condenser vacuum. The SJAEs consist of two redundant trains (with one train color coded "red" and the other "blue") for each of the two main condenser sections (i.e., there is a red and blue train for condenser 1A and a red and blue train for condenser 1B). Each train consists of two first and two second stage ejectors with one first and one second stage ejector in service on each SJAE unit. The plant was originally designed to operate with two first and two second stage ejectors in service on one SJAE unit, with the other SJAE unit in backup. However, SJAE performance was not meeting design levels of operation and research indicated that air ejector performance may have been poor since initial plant startup. Accordingly, under this SP, guidance was provided for three distinct operating variations to evaluate overall SJAE performance and design. The first operation closed the first stage drain valves of the operating train to isolate the drain loop seal from the first stage suction housing. (The loop seals were added following initial plant startup to address SJAE surging. The loop seals had little effect on the surging phenomena and may have actually compromised efficiency.) The second operation varied the steam supply pressure to the first and second stage nozzles to assess injector performance. The third operation placed the out-of-service second stage air ejectors into service. Based on the results, the SJAEs are being operated in the optimized configuration under SP 00-003 until permanent changes to the system operating procedures are put into effect. Revision 1 to the SP (which had no effect on the USQE) added flexibility to allow individual sections to be performed or re-performed in any order as recommended by the Thermal Performance Engineer. Revision 2 to the SP combined two sections into one, allowing both sets of second stage ejectors for condensers 1A and 1B to be placed in service at the same time. This resulted in a minor revision to the USQE.

SAFETY
EVALUATION: This SP did not perform any activities which will initiate an accident and, therefore, the probability of an accident previously evaluated in the SAR was not increased during the performance of the SP. However, they can contribute to a loss of condenser vacuum transient, which has been previously evaluated. Since the SP contains sufficient controls to monitor and recover condenser vacuum should an adverse trend occur, the probability of a loss of vacuum transient was also not increased. As the off-gas isolation function was unchanged as a result of this activity, the consequences of an accident previously evaluated in the SAR were not increased. No hardware changes were made under this SP and equipment and system design parameters were not exceeded by the operational variations. Therefore, the probability of a malfunction of equipment important to safety previously evaluated in the SAR was not increased. While operational parameters and system configuration were varied, no action setpoints or mitigation activities were altered by this activity. Therefore, the consequences of

a malfunction of equipment important to safety previously evaluated in the SAR were not increased. As previously noted, all activities were performed within the normal design of the equipment and system. Accordingly, the possibility for an accident or equipment malfunction of a different type was not created. Since the radioactivity rates specified for the Augmented Off-Gas are not being altered, and the highest expected operating gross gamma activity rate is not being altered, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

SP 99-002
(USQE 1999-0016)

TITLE: Main Turbine Governor Valve Number 2 Valve Curve Adjustment On-Line

DESCRIPTION: To improve main turbine efficiency, this SP provided guidance for adjusting the main turbine governor valve number 2 (GV2) sequential valve curve characteristic while on-line at 85 percent reactor power. The main turbine control valves can operate in two different modes of operation: single or sequential valve control. In single valve control, all governor valves open or close together at the same rate, while in sequential valve control the valves open or close in a sequence as the load demand increases or decreases. Discussions with various turbine control vendors indicate that the most efficient valve position is at a point where all of the control valves are either fully opened or fully closed, such that valve throttling is at a minimum. Prior to implementing the SP, at 100 percent reactor thermal power GV1 and GV3 were fully open, GV4 was about 80 percent open, and GV2 was only 6 to 8 percent open. If the flow through GV2 were to increase, then flow through GV4 would decrease and a net decrease in throttling losses would result. Under the SP, the valve curve for GV2 was varied incrementally at six different levels between 5 and 20 percent open (at full power) so that data could be taken to determine the most efficient operating curve. (No changes were made to the GV4 valve curve as it will automatically compensate for changes in throttle pressure resulting from changes in the GV2 position.) In brief, the SP was implemented by dropping to 85 percent power, transferring valve control to the single mode, adjusting the GV2 curve on the valve servo card, returning to sequential mode, and slowly increasing power to 100 percent while monitoring turbine performance.

SAFETY EVALUATION: The governor valve servo cards are independent of each other and, therefore, changes to the GV2 valve curve have no effect on the operation of the other governor valve servo cards or the digital electro-hydraulic (DEH) pressure controllers that provide input signal to the servo cards. Since the pressure controllers are unaffected, any failure of GV2 during performance of the SP would be compensated for by the other governor and bypass valves. Additionally, changes to the GV2 opening characteristics have no impact on the main turbine hydraulic trip system and, therefore, the turbine response to accidents or transients would be unchanged. For these reasons, the probability of occurrence and the consequences of an accident previously evaluated in the SAR were unchanged. Similarly, the probability of occurrence and the consequences of an equipment malfunction previously evaluated in the SAR were unchanged. Since this SP did not affect the function of the DEH pressure controllers, GV2 will function as designed in response to DEH controller output, and no new failure modes were introduced by the changes to the GV2 valve curve, the possibility of an accident or equipment malfunction of a different type was not created. Since the reload licensing analyses are based on single valve control at full power (i.e., the most conservative assumption), transferring to single control at 85 percent power and any changes to the GV2 valve curve are bounded. Therefore, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

SP 99-004, Revision 2
SP 99-005, Revision 1
(USQE 1999-0053, Revision 2/USQE 1999-0055, Revision 1)

TITLE: 125 VDC Station Battery Cell Replacement/250 VDC Station Battery Cell Replacement

DESCRIPTION: Under CED 1999-0121 and CED 1999-0122, a methodology was developed to allow for the replacement of one or two cells in the 125 VDC and 250 VDC batteries while maintaining battery availability and minimizing periods of inoperability. The subject SPs provided a means of replacing one or two cells in the affected 125 VDC and 250 VDC battery subsystems without de-energizing the battery subsystem. SP 99-004 addresses the specifics pertaining to the Division A and B 125 VDC batteries and SP 99-005 addresses the specifics pertaining to the Division A and B 250 VDC batteries. To briefly summarize, the affected battery is temporarily modified by: 1) connecting identical cells in parallel to the cells being replaced, 2) disconnecting and replacing the degraded cells, and 3) removing the parallel cells, thus restoring the battery to the original qualified configuration. The revisions to the SPs changed the number of parallel cells that could be installed and, hence, the number of cells that could be replaced during SP performance. These revisions had minimal impact on the supporting USQEs.

SAFETY EVALUATION: USQE 1999-0053 was developed to support SP 99-004, and USQE 1999-0055 was developed to support SP 99-005. 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. While the affected battery subsystem is declared inoperable and the appropriate Limiting Condition for Operation (LCO) is entered during SP performance, the battery will remain available for the duration of the activity. As this activity is only applied to one 125 VDC 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 serviced under this activity. Since the affected battery is considered inoperable and the appropriate LCO is entered, the probability of an equipment malfunction previously evaluated in the SAR is not increased by 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 is bounded by the SAR evaluations, the consequences of an equipment malfunction previously evaluated in the SAR are not increased by this activity. The failure of a 125 VDC or 250 VDC battery is not an accident initiator and loss of a battery subsystem is previously analyzed and would not prevent safe shutdown. Since the circuit breaker installed along with the parallel cells met applicable ratings, no new failure modes were created. Additionally, the parallel cells were restrained and the possible malfunction of the battery subsystem due to the installed parallel cells is no different than the failure of any other battery cell. For these reasons, the possibility of an accident or malfunction of a different type cannot be created by this activity. Since compliance with the established LCOs ensures that the margin of safety is maintained during SP performance, the margin of safety as defined in the basis for any Technical Specification is not reduced by this activity.

SP 99-010
(USQE 1999-0092)

TITLE: Diesel Generator 2 On-Line Troubleshooting (Division 2)

DESCRIPTION: During the performance of a routine monthly surveillance test, Diesel Generator 2 (DG2) experienced excessive power oscillations causing a breaker trip. DG2 was subsequently declared inoperable. This SP was developed to assist station personnel in troubleshooting the problem. The procedure consisted of starting the diesel and operating it under varying load conditions while recording data and monitoring the governor for oscillations. The SP included recommendations provided by the vendor and specified the criteria and actions to be taken to secure the diesel should significant oscillations be

observed. Although the diesel was operated while connected to the plant grid, all normal protective relaying functions were in place to protect the bus and diesel. During the SP the speed setting on the DG2 hydraulic actuator was adjusted to bring the frequency within specifications. The test continued with no oscillations noted and all operational parameters were normal.

SAFETY EVALUATION: Since the DG is not an accident initiator, the probability of an accident previously evaluated in the SAR was not affected by either the DG or the performance of the SP. As the appropriate Technical Specification (TS) Limiting Condition for Operation (LCO) was entered due to DG inoperability, and as the accident analyses assume the loss of one DG, the consequences of an accident previously evaluated in the SAR were unchanged. With all normal protective relaying functions in place during SP performance, the probability of an equipment malfunction previously evaluated in the SAR was unchanged. The consequences of an equipment malfunction are bounded by the single failure design of the safety-related power supply systems. Since the DG equipment is not associated with any type of accident initiators and precautions were put in place so that the activity could not impact safe operation of plant equipment, the possibility for an accident or malfunction of a different type was not created. As noted above, the DG was inoperable and the appropriate LCO had been entered. In that compliance with the TS LCO was maintained, the margin of safety as defined in the basis for any Technical Specification was not reduced.

SP 00-002
(USQE 2000-0006)

TITLE: Post MP 97-068A & 97-100 AOG Operation with Recombiner B

DESCRIPTION: This special procedure provided for the start up of Augmented Off-Gas (AOG) Train B following the implementation of modifications MP 97-068A and MP 97-100. As part of the special procedure, the effectiveness of the modifications was evaluated and data gathered to establish the operating parameters used in generating permanent revisions to the applicable station procedures. MP 97-068A added an equalizing line between the 48 inch holdup drain line loop seal high point in the Z sump and the inlet side of the off-gas pre-filter to eliminate any air binding present in the loop seal (air binding is suspected to have caused excessive water retention in the 48 inch holdup line). MP 97-068A also installed a differential pressure indicator to allow determination of whether Standby Gas Treatment (SGT) operability is challenged during AOG operation. MP 97-100 provided essential power to the Z sump and documented the seismic qualification of the Z sump system as Class IS-Operating Basis Earthquake design. Performance of the SP demonstrated that AOG Train B could be operated without excessive water retention in the 48 inch holdup line. This was accomplished by throttling AOG-AOV-PCV1675 (AOG inlet to after filters) to maintain differential pressure between the 48 inch holdup line and the Z sump at less than 50 inches water gage.

SAFETY EVALUATION: The SGT, Z sump, and AOG systems are not plant event initiators for accidents, transients, and other events as described in USAR Chapter XIV. Accordingly, the probability of an accident previously evaluated in the SAR has not increased. The Z sump system is required to remain operable in order to support SGT system operation during a Loss of Coolant Accident concurrent with a Loss of Offsite Power or a Refueling Accident. Design calculations determined the maximum allowable differential pressure between the 48 inch holdup line and the Z sump as being 50 inch water gage before water will accumulate in the 48 inch holdup line, thus challenging Z sump operability. Since this special procedure directed the operators to secure the AOG system if the 50 inch water gage differential pressure limit was exceeded, the operability of the Z sump and SGT systems was not challenged by performance of this special procedure. Therefore, the consequences of an accident previously evaluated in the SAR have not increased. Hydrogen entering the Z sump was not an issue since that event involves the 48 inch holdup line becoming positively pressurized, whereas this SP was countering a condition where the holdup line becomes slightly negative. The SP also required the AOG system be secured if the differential pressure dropped below 0 (zero) inches water gage (such

that the holdup line pressure would become positive with respect to the Z sump). This limit, along with the previously discussed 50 inch water gage limit, ensures that the probability and consequences of a malfunction of equipment important to safety have not increased. This is further supported by the fact that slight changes in AOG back pressure (which could result from operation between the two limits discussed above) will not adversely impact the off-gas effluent radiation monitoring, main condenser vacuum, or off-gas system isolation functions. Since the limits established by the special procedure will not result in the operation of plant systems or equipment outside of normal operating parameters, and since no new failure mechanisms are introduced by the special procedure, the possibility of an accident or equipment malfunction of a different type has not been created. Similarly, the margin of safety as defined by the basis for any Technical Specification has not been reduced.

SP 00-003
(USQE 2000-0015)

TITLE: Steam Jet Air Ejector (SJAE) Startup Using Four Second Stage Ejectors

DESCRIPTION: Under SP 99-001, the steam jet air ejectors (SJAEs) were operated in varied configurations to determine if overall plant efficiency could be improved. Based on the results of SP 99-001, it was demonstrated that improved SJAE and main condenser performance is achieved by operating the SJAEs with two first stage and all four second stage ejectors (total of six) in service. As the SP was implemented during normal operation at steady state conditions, there was no guidance provided for starting up the plant with all second stage ejectors aligned for service. (This is desirable in order to reduce personnel exposure and eliminate the minor plant perturbations experienced under SP 99-001 when valving in the second set of ejectors.) Accordingly, this SP was developed to determine if the SJAEs could be successfully placed in service at a reactor pressure of approximately 500 psig. The SP included provisions to remove one set of second stage ejectors and proceed with a normal startup if adequate steam supply pressure (i.e., the limiting variable) could not be maintained. Should this occur, steps from SP 99-001 were included to return the second set of ejectors to service once sufficient reactor pressures were achieved.

SAFETY EVALUATION: This SP did not perform any activities which would initiate an accident. Additionally, since the SP contains sufficient controls to monitor and recover condenser vacuum should an adverse trend occur, the probability of a loss of vacuum transient was not increased. Therefore, the probability of an accident previously evaluated in the SAR was not increased. As the off-gas isolation function was unchanged as a result of this activity, the consequences of an accident previously evaluated in the SAR were not increased. Since no new equipment and the associated new failure modes were introduced, and the activity maintained all air removal system parameters within their normal range, the probability of a malfunction of equipment important to safety previously evaluated in the SAR was not increased. No action setpoints or mitigation activities were altered by this activity. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR were not increased. As previously noted, all activities were performed within the normal design of the equipment and system. Accordingly, the possibility for an accident or equipment malfunction of a different type was not created. This activity did not alter the margin between Technical Specification gross gamma rate or the reactor cooldown rate and specified operating limits for both. Accordingly, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

USAR CHANGE REQUESTS (UCRs), OPERATING LICENSE CHANGE REQUESTS (OLCRs), AND TECHNICAL REQUIREMENTS MANUAL CHANGE REQUESTS (TRMCRs)

UCR 99-018
(USQE 1999-0040)

TITLE: USAR Personnel Monitoring Device Requirements

DESCRIPTION: The personnel radiation monitoring section of the USAR (Volume III, Section VII-15.1.2) was revised to remove information that has been superseded by revisions to 10 CFR 20 and specific exemptions to the requirements of 10 CFR 70.24. This section was also written in more generic terms so that future changes in 10 CFR 20 personnel monitoring requirements will not require associated changes in the USAR. Personnel radiation safety is unaffected by the change since the monitoring requirements and practices, controlled by the Radiation Protection Program Station Procedures, reflect current 10 CFR 20 and Technical Specification requirements, including the conditional requirements associated with the aforementioned exemption.

SAFETY EVALUATION: The changes to personnel radiation monitoring requirements are administrative in nature and do not involve any physical changes to the design, operation or analyses governing the facility or systems, structures, or components assumed to operate in the SAR accident mitigation analysis. Additionally, the changes do not affect any accident mitigation assumptions. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased. Likewise, since the changes also do not create a new plant operating condition or mode, the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created. Technical Specification Amendments 174 and 178 previously incorporated the specific changes to 10 CFR 20. Therefore, the margin of safety as defined in the basis for any Technical Specification is unaffected by this USAR revision.

UCR 99-025
(USQE 1999-0036)

TITLE: USAR Change to Delete Appendix E and Incorporate the ODAM by Reference

DESCRIPTION: This change deleted the information in USAR Appendix E in its entirety, and replaced it with the Offsite Dose Assessment Manual (ODAM) by reference. (Previously reported USQE 1998-003 provides the generic acceptability of incorporating documents into the USAR by reference.) Editorial changes were also made to Chapter IX, Section 4.5 to refer to the ODAM for establishing Elevated Release Point release rates rather than Appendix E, and a change was made to the reference in Appendix F. Finally, a reference to the Technical Specifications was deleted from Chapter XIII, Section 13.1. This was a carryover from the change that implemented the Improved Technical Specifications, and should have been deleted at that time.

Appendix E to the Final Safety Analysis Report (FSAR), Elevated Release Point Release Rate Limit Calculations, was submitted as Amendment No. 8 to the License Application. The appendix included an analytical model for calculating the annual gamma dose rate at ground level resulting from a continuous release of radioactive materials, as well as calculations for establishing stack release limits based on the model. The purpose of Appendix E was to meet the FSAR content requirement of 10CFR50.34(b) (3) and the 1966 AEC Guide for the Organization and Content of Safety Analysis Reports. Subsequent to the receipt of the Operating License, the NRC required changes to the License and Technical Specifications in order to comply with Section V of Appendix I to 10CFR50. These new requirements were referred to as "Radiological Effluent Technical Specifications (RETS)." The

RETS submittal for CNS used the title of "Offsite Dose Assessment Manual." The NRC approved the RETS submittal as Amendment 89 to the CNS Operating License.

Appendix E to the USAR contains obsolete information which has been superseded by changes to the CNS Operating License and the implementation of the ODAM. The deletion of the obsolete information and incorporation of the ODAM by reference is administrative in nature, and does not involve any physical change to the facility or change to the procedures or methods by which the plant is operated. The use of the ODAM is approved by the NRC and included in the CNS Technical Specifications.

SAFETY EVALUATION: Appendix E to the USAR became obsolete at the time the ODAM was approved for use to determine alarm setpoints and dose rates from radiological releases. This change deletes the obsolete information and incorporates the ODAM by reference. Accordingly, this revision does not involve a physical change to the facility, and does not affect or alter any procedures or methods by which the plant is operated. As a result, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased. Likewise, the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created. The Technical Specifications require the use of the ODAM. Since this revision deletes obsolete information from the USAR which is in conflict with the methods described in the ODAM, and the ODAM is incorporated into the USAR by reference, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

UCR 2000-001
(USQE 1999-0063)

TITLE: USAR Rebaseline Project - Chapter VIII, Binder 8

DESCRIPTION: This UCR documents more than 200 changes made to Chapter VIII, Sections 1 - 4. All changes except [11], [58], [59], [61], [62], [146], [149], [152], and [175] were verified not to constitute a change to the facility or its operation as described in the SAR and thus screened as not requiring further 10CFR50.59 evaluation. This USQE documents the results of the aforementioned changes requiring further evaluation under 10CFR50.59.

Changes [11], [59], and [152] revised the USAR to indicate that not all of the loads will be fast transferred upon a trip of the main generator or Normal Station Service Transformer (NSST) supply. The previous USAR description stated that "each" 4 kV bus will execute an automatic fast transfer to the startup power source in the event that unit power is lost for any reason. This was not accurate as Buses 1C and 1D will not fast transfer. This logic configuration has been present since plant startup.

Changes [58], [146], and [149] revised the USAR to leave one of the buses 1C or 1D powered from the Startup Station Service Transformer (SSST) during normal operation. Based on actual procedures in use, not all auxiliary loads are transferred to the NSST during startup, above approximately 20 percent power. Per plant procedures, either Bus 1C or 1D is left on the SSST at full power while the other bus is transferred to the NSST. This operating configuration avoids the loss of both Recirculation Pumps upon a generator trip.

Change [61] describes an existing feature of the startup power source breaker logic. The change revised the USAR to add information stating that the startup transformer cannot be inadvertently interconnected with the normal AC power source. This logic configuration has been present since plant startup.

Change [62] added the startup AC power source in addition to the normal AC power source with which the diesel generators can be synchronized. Per the existing safety design basis for the standby AC power source, the diesel generators must be capable of parallel operation with the normal AC power source for system performance testing. This information was duplicated in this section and the practice of parallel operation between the startup AC power source and the diesel generators was added.

Change [175] increased the upper voltage level of normal operation for the 4 kV buses. This change was evaluated in Nuclear Engineering Department Calculation 87-132A, and is shown to be within the acceptable operating voltage for the affected plant equipment. This change reflects the results of detailed evaluations of the 4 kV system normal operating voltage conditions. Previously, the USAR listed the normal operating voltage range as being from 3950 to 4250 VAC. The revised range is 3950 to 4400 VAC.

SAFETY EVALUATION: Changes [11], [59], and [152] - The USAR was revised to indicate that not all of the loads will be fast transferred upon a trip of the main generator or NSST supply. Since the fast transfer is only required after an event has occurred, the change cannot increase the probability of an event previously evaluated in the SAR. The fast transfer capability of the safety-related loads is unaffected by this change and, therefore, the consequences of an event are unchanged. For the same reason, the probability and consequences of an equipment malfunction are unchanged. Loss of power to the Reactor Recirculation Pumps due to buses 1C and 1D not transferring upon a trip of the main generator or NSST has no effect on any accident initiator and thus cannot introduce any accident not evaluated previously in the SAR. As noted above, this operating configuration has no impact on the fast transfer of equipment important to safety and, as a result, there is no increase in the probability of a malfunction of a different type than any previously evaluated in the SAR. The only applicable Technical Specification is 3.8.1, which requires that two qualified circuits between the offsite transmission network and the onsite Class IE AC Electrical Power Distribution System be OPERABLE. Since the fast transferred capabilities of the safety-related buses 1A and 1B are unaffected, there is no impact to the Technical Specifications requirements and thus the margin of safety has not been reduced.

Changes [58], [146], and [149] - The USAR was revised to leave one of the buses 1C or 1D powered from the SSST during normal operation. Since the SSST meets all of the requirements for power operation, these changes do not increase the probability of an accident previously evaluated in the SAR. Safety-related systems and equipment are supplied from the 1A and 1B buses which are unaffected by the change. Therefore the accident and transient analyses in the SAR are also unaffected by the changes as are the consequences of these accidents and transients. Accordingly, the probability and consequences of an equipment malfunction are unchanged. As noted above, the SSST meets all of the requirements for power operation. Therefore, the probability for an accident or malfunction of a different type has not been created. Similar to changes [11], [59], and [152] discussed above, the only applicable Technical Specification is 3.8.1, which requires that two qualified circuits between the offsite transmission network and the onsite Class IE AC Electrical Power Distribution System be OPERABLE. Since the fast transferred capabilities of the safety-related buses 1A and 1B are unaffected, there is no impact to the Technical Specifications requirements and thus the margin of safety has not been reduced.

Change [61] - This revision describes an existing feature of the startup power source breaker logic. This feature is provided to increase the reliability of the AC electrical distribution system by preventing the propagation of a fault and is not required until after a power source has been lost. Therefore, this feature does not increase the probability of a power loss and, consequently does not increase the probability of a transient or accident. In actuality, by increasing system reliability, the likelihood of a Station Blackout event is decreased. The breaker controls for the startup AC power source were always arranged to prevent the additional inadvertent connection of the NSST but that

feature was not described. Mitigation systems credited with coping with accidents previously evaluated in the SAR are not affected by this change. The change described is provided to increase the reliability of the AC power distribution system and hence the availability of power and, therefore, does not increase the consequences of accidents previously evaluated. Since none of the equipment or sources are considered important to safety and are not credited in plant analysis, the probability and consequences of a malfunction of equipment important to safety have not been increased. The only credible failure is a loss of power which is already evaluated in the SAR. No interconnections between electrical divisions which could possibly result in a new failure over and above that already evaluated is introduced by this change. Moreover, the loss of power is not an initiator of any accident. Therefore, this change does not create the possibility for an accident or malfunction of a different type than any previously evaluated in the SAR. As with the previously discussed changes, the only applicable Technical Specification is 3.8.1. Since the configuration increases the reliability of the AC electrical distribution system by preventing the propagation of a fault, it contributes to satisfying this Technical Specification requirement. Hence, the margin of safety as defined in the basis for this Technical Specification has not been reduced.

Change [62] - This USAR revision added the startup AC power source in addition to the normal AC power source with which the standby power source can be synchronized. Operation of a diesel generator in parallel with the startup power source by itself does not increase the probability of an accident, transient or special event since the bounding loss of offsite power event is already assumed. Plant procedures, and the existing SAR, do not allow the paralleling of both diesel generators with offsite sources simultaneously, consequently, even in the case where a loss of offsite power occurs and the associated diesel generator is coincidently lost, current analysis requires only one diesel generator and this condition is bounded. For the same reasons, the consequences of an accident previously evaluated in the SAR are not increased. The standby power source is designed for parallel operation with offsite power sources and operation in this configuration is generally considered a safe and reliable practice. However, when the diesel generators are operating in parallel with offsite power sources, no credit is taken for their ability to support a loss of offsite power coincident with a Loss of Coolant Accident (or any other lesser event). Credit is taken for the opposite division's standby power source which has no interconnection with the non-credited source. Consequently, this change does not increase the probability of a malfunction of equipment important to safety previously evaluated in the SAR. As noted above, the existing analysis assumes a loss of offsite power and the availability of only one train of mitigation systems. Consequently, even in the worst case where a loss of the startup power source occurs while in this configuration, and assuming the associated diesel generator is coincidently lost, the condition is bounded by existing analysis and the consequences of a malfunction of equipment important to safety previously evaluated in the SAR is not increased. The only credible failure is a loss of power which is already evaluated in the SAR. No interconnections between electrical divisions which could possibly result in a new failure over and above that already evaluated is introduced by this change. Moreover, the loss of power is not an initiator of any accident. Therefore, this change does not create the possibility for an accident or malfunction of a different type than any previously evaluated in the SAR. As with the previously discussed changes, the only applicable Technical Specification is 3.8.1. During the time that a diesel generator is in parallel operation with an offsite source, no credit is taken for that diesel generator (i.e., there is no reduction beyond that already assumed in Technical Specifications). Hence, the margin of safety as defined in the basis for this Technical Specification has not been reduced.

Change [175] - This USAR revision increased the upper voltage level of normal operation for the 4 kV buses. This change was evaluated in Nuclear Engineering Department Calculation 87-132A, and is shown to be within the acceptable operating ranges for the affected plant equipment. Therefore, there is no increased potential for a loss of power to a critical station bus and the probability of an accident or equipment malfunction previously evaluated in the SAR has not been increased. Since the voltage levels involved are within acceptable limits for all affected equipment and no protective functions are impacted, the consequences of an accident or equipment malfunction have not increased. Since no

changes in system use, process pressures or temperatures or other potential challenges to boundary integrity are introduced, there is no potential for new accident sequences to be created. Further, since the affected equipment is not subjected to electrical conditions or configurations beyond design and there are no wiring changes or control logic changes introduced, there is no potential for a different type of malfunction to be created. The revision does not allow for the operation of the 4 kV buses closer to their undervoltage setpoint as given in Technical Specification 3.3.8.1 and the response of the system to degraded voltage conditions or faults is not impacted. Therefore, the margin of safety is not affected by this change.

UCR 2000-004
DC 92-141B

TITLE: USAR Rebaseline Project-Chapter VIII, Binder 13, Change [4]

DESCRIPTION: As part of the USAR rebaseline project, Change [4] of Binder 13 affecting Chapter VIII incorporated changes resulting from DC 92-141B. The DC 92-141B related changes were incorporated under USAR Change Request (UCR) 2000-004. Under the design change (DC), the control power circuit for the Low Pressure Coolant Injection (LPCI) and Reactor Recirculation (RR) pump discharge valves was changed from 125 VDC to 250 VDC, to match the motive power for the valves. These actions were taken to address potential single failure vulnerabilities as documented in Nonconformance Report (NCR) 92-90.

SAFETY EVALUATION: Under this activity, the control power source was changed to match the motive power source for the LPCI and RR pump discharge valves. All components exposed to the higher voltage levels were verified to be compatible for 250 VDC service. As the change was limited to the control power source, there was no effect on the safety function of the valves or interfacing systems. Therefore, the probability of an accident previously evaluated in the SAR has not increased. As noted above, all components were verified to be compatible for the higher voltage service. Consequently, the probability of an equipment malfunction previously evaluated in the SAR has not increased. (Since the number of power supply dependencies has been reduced, overall reliability has been improved and the probability of failure has actually been reduced slightly.) Also, as noted above, this activity had no impact on the safety function of the valves or interfacing systems. Accordingly, the consequences of an accident or equipment malfunction have not increased. As no new failure modes were introduced, the possibility of an accident or malfunction of a different type has not been created. Finally, since the minimal loads added to the 250 VDC system were confirmed to be acceptable, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

UCR 2000-005
(USQE 1999-0051)

TITLE: Environmental/Industrial Safety Supervisor - Responsibility Change

DESCRIPTION: Under this activity, USAR Figure XIII-2-2 detailing the Nuclear Power Group (NPG) management organization was revised to delete the Nuclear Environmental/Industrial Safety Supervisor position. To make more efficient utilization of personnel, the position was changed to Industrial Safety Coordinator. The new position, although still reporting directly to the Radiological Manager, is without supervisory responsibilities and is therefore no longer considered to be part of the management organization. In addition to the figure, Section XIII-2.1.4.5 was revised to clarify the responsibilities of the Radiological Manager with regard to the environmental and industrial safety programs. As the change is administrative in nature, and does not impact the overall functional aspects of existing programs, processes, procedures, activities, or plant operating systems, structures, or components, it has no effect on nuclear safety.

SAFETY
EVALUATION: As the above described organizational change is administrative in nature and does not affect the design, function or operation of systems, structures or components important to safety, the probability of occurrence and the consequences of an accident or equipment malfunction previously evaluated in the SAR have not increased. Likewise, the possibility for an accident or malfunction of a different type has not been created. As defined staffing requirements and qualifications are unaffected by the change, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

UCR 2000-009
DC 87-169
MDC 81-009

TITLE: Change to USAR Chapter V, Sections 2.3.5.2 and 2.3.7

DESCRIPTION: As part of the USAR rebaseline project, Section V-2.3 revisions incorporated changes resulting from Design Change (DC) 87-169 and Minor Design Change (MDC) 81-009. The DC and MDC related changes were incorporated under USAR Change Request (UCR) 2000-009. DC 87-169 was implemented in support of Change Number 31 to the CNS Technical Specifications (TS). The TS change required a check once per operating cycle to ensure that the four primary containment purge and vent butterfly valves were limited to a maximum opening rotation of 60 degrees (to ensure that the maximum closing time of 14 seconds could always be met). Under the DC, the mechanical stops were set to limit the opening rotation to 45 degrees. In addition, the motor operators were modified (through changes to motor speed and gear ratios) to eliminate the need for magnetic motor brakes. The vent and purge valves receive an auto closure signal from the Primary Containment Isolation System upon a Group II or VI isolation signal. MDC 81-009 was implemented in support of NRC IE Bulletin 80-25. The Bulletin identified a potential condition where two-stage Target-Rock relief valves could actuate as a result of a pneumatic supply overpressure condition. As a result, a redundant relief valve was installed under the MDC on the Automatic Depressurization System (ADS) safety relief valve pneumatic supply piping, and the setpoints were reduced slightly to 120 psig. In addition, the relief valve settings on the instrument air dryers (the backup pneumatic supply) were reduced from 150 to 130 psig. The safety evaluations for the modifications are individually summarized below.

SAFETY
EVALUATION: DC 87-169: No changes were made in the operational characteristics of the purge and vent valves other than the slightly increased closure time and the increased available torque margin of the modified operators. This enhances the capability for isolating the purge and vent lines against the differential pressure and flow conditions following a Design Basis Loss of Coolant Accident. Consequently, the probability and consequences of an accident or malfunction of equipment important to safety have not increased. Since the modification did not result in any new mode of operation, the possibility for an accident or malfunction of a different type has not been created. As no operating limits or trip setpoints are affected, and the TS closure time limits continue to be met, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

MDC 81-009: As the ADS function has been evaluated assuming a complete loss of instrument air, the addition of a redundant relief valve does not increase the probability of an accident previously evaluated in the SAR. By adding the redundant relief valve and reducing the relief valve setpoints on the ADS pneumatic supply header and the backup instrument air dryers, the probability of the equipment malfunction postulated by the Bulletin has been reduced. As the system operation is unchanged, the consequences of an accident or equipment malfunction previously evaluated in the SAR have not been reduced. Since no new failure modes or equipment or system interactions were introduced, the possibility of an accident or malfunction of a different type has not been created. Finally, since relief valves were not addressed in the Technical Specifications, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

UCR 2000-011
DC 85-15L, Amendment 1

TITLE: USAR Chapter X, Section 10 Revisions: DCRDR Modification to Control Room Panel VBD-R

DESCRIPTION: As part of the USAR rebaseline project, Section X-10 revisions incorporated changes resulting from Design Change (DC) 85-15L, Amendment 1. The DC related changes were incorporated under USAR Change Request (UCR) 2000-011. DC 85-15L was implemented as part of the Detailed Control Room Design Review (DCRDR) project performed pursuant to NUREG-0737, Item I.D.1. The primary focus was to rearrange the indicators and controls on Vertical Board R in accordance with established Human Factors Engineering guidelines. While the DC provided for numerous changes, UCR 2000-011 focused on the elimination of the automatic advance feature for the pre-filters associated with the Reactor Building, Turbine Building, and Control Building ventilation supply units. As a result of this change, the filter media are manually advanced locally by station operators based on visual observation of particulate loading or increased differential pressure readings. The pre-filter units are classified as nonessential and are intended to protect the nonessential supply fans for the above listed building ventilation systems.

SAFETY EVALUATION: As noted above, the ventilation supply unit pre-filters are nonessential and therefore have no modes of operation that can affect accident precursors, or accident or transient response. Further, there are no credible failure modes that could adversely affect other safety-related equipment. Accordingly, the probability of occurrence or the consequences of an accident or equipment malfunction previously evaluated in the SAR have not increased. Likewise, the possibility of an accident or malfunction of a different type has not been created. There are no Technical Specifications pertaining to the pre-filters. Hence, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

UCR 2000-012
(USQE 1999-0074)

TITLE: USAR Section X-10.3.4.4 Revision to Allow Operating Four Turbine Building Exhaust Fans, As Needed, for Cooling and to Control Turbine Building Differential Pressure

DESCRIPTION: Under this activity, the USAR was revised to provide for additional operational flexibility with respect to the Turbine Building Ventilation system. As initially designed, four exhaust fans, each having one third capacity, were installed. In the normal line up, three would be running with the fourth in a standby condition. However, operating experience dictates that under certain weather conditions, operation of the fourth fan is required to achieve the desired differential pressures (DP) or temperatures. Accordingly, Section X-10.3.4.4 was revised to allow the normal configuration to include the operation of the fourth fan on an as-needed basis. The Turbine Building Ventilation system is classified as nonessential.

SAFETY EVALUATION: As the alignment of the Turbine Building Ventilation system cannot contribute to the initiation of an accident or equipment malfunction, the probability of an accident or equipment malfunction previously evaluated in the SAR has not increased. (In actuality, the decreased operating temperatures resulting from increased ventilation will improve equipment reliability and, in effect, reduce the probability of an equipment malfunction.) Since the Turbine Building Ventilation system is not used to mitigate the consequences of an accident, and the response to a failure of an exhaust fan remains unchanged, the consequences of an accident or equipment malfunction previously evaluated in the SAR have not increased. The change does not operate equipment outside current design limits and failure mechanisms are unaffected; as a result, the possibility for an accident or malfunction of a different type

has not been created. Since the functionality of the Turbine Building radiation monitors is not affected by the change in exhaust flow, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

UCR 2000-019
(USQE 2000-0011)

TITLE: NPG Organization and Responsibility Revisions

DESCRIPTION: This UCR documents various administrative changes in job titles, responsibilities, and reporting relationships within the Nuclear Power Group (NPG). The specific changes are summarized as follows:

- The Risk & Regulatory Affairs Manager position was added with responsibility for Licensing and Risk Management. This position reports to the Senior Manager of Technical Services.
- The reporting relationship of Risk Management changed from Engineering Support to Risk & Regulatory Affairs.
- The position title of Nuclear Information Services Supervisor was changed to Nuclear Information Technology Manager. This position reports directly to the Senior Manager of Site Support and is responsible for managing CNS computer-related activities.
- The Nuclear Licensing and Safety Supervisor position was deleted.

A review against other Licensing documents (i.e., Quality Assurance Policy Document, Safeguards Plan, and Emergency Plan) verified that there are no reductions in commitment or effectiveness as a result of the changes.

SAFETY EVALUATION: The changes in organizational structure and responsibilities do not delete or affect the overall functional aspects of existing programs, processes, procedures, or activities at CNS; nor do the changes affect the design, function, method of performing a function, system interface, or operating parameters of plant systems, structures, or components important to safety. Therefore, the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR has not increased. Since the changes are administrative in nature, existing fission product barriers and established mitigative actions are unaffected. Further, the changes do not reduce the effectiveness of the CNS Emergency Plan or have a negative impact on the facility's ability to mitigate the consequences of an accident. Therefore, the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR have not increased. As previously noted, the changes are administrative in nature and the possibility of an accident or malfunction of a different type has not been created. Since the overall administrative responsibilities, organizational requirements, and unit staff qualifications are unaffected by this change, the margin of safety as defined in the basis of any Technical Specification has not been reduced.

UCR 2000-025
MDC 77-46

TITLE: USAR Rebaseline Project - Chapter X, Binder 18

DESCRIPTION: As part of the USAR Rebaseline Project, Chapter X was updated to reflect changes made to the facility under Minor Design Change (MDC) 77-46. These changes were documented and incorporated by UCR 2000-025. Under MDC 77-46, the original fuel grapple was replaced with a modified Boiling Water Reactor 6 (BWR-6) type fuel grapple featuring a redundant hook fuel grapple head. The modified BWR-6 design provides improved protection against the mis-engagement of the hook such that the potential for a dropped fuel bundle during refueling activities is not possible.

SAFETY
EVALUATION: While the consequences of a fuel handling accident are unchanged by the modification, use of the redundant hook fuel grapple head reduces the potential for a dropped fuel bundle. Likewise, the modification has no impact on the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the safety analysis report. Since the new fuel grapple does not present any different types of safety concerns than previously analyzed, the possibility of an accident or malfunction of a different type has not been created. The new grapple provides safer operation than the existing design. Consequently, the margin of safety as defined in the basis for any Technical Specification is unchanged as a result of the modification.

UCR 2000-032
(USQE 2000-028)

TITLE: USAR Change, Chapter VII, Section 3.4.5

DESCRIPTION: Under this USAR change, two portions of Section VII-3.4.5 were revised to more accurately reflect the actual design of the facility. Specifically, the USAR stated that a trip of an isolation trip system channel is annunciated in the control room. This was not totally correct as only a trip of the Primary Containment Isolation System (PCIS) Group 1 channel is annunciated, while a trip of the remaining PCIS channels is indicated by the extinguishing of panel indicating lights. Accordingly, the USAR was clarified to state that a trip of an isolation trip system channel is indicated (i.e., by annunciation or by the extinguishing of indicator lights) in the control room. The second clarification dealt with a statement that read: "The positions of air-operated isolation valves are displayed in the same manner as motor operated valves." The actual plant design does not include position indicating lights on the PCIS mimic for valves RR-AOV-740AV and RR-AOV-741AV. Accordingly, the original USAR statement was expanded to note this exception. There were no physical changes to the facility or procedures associated with this USAR change.

SAFETY
EVALUATION: As noted above, this activity implemented changes to the USAR to more accurately reflect the actual configuration of the plant; there were no physical changes to the facility or procedures and this change did not affect the control of reactor operations, accident initiators, or capability of mitigation systems. As a result, the probability of occurrence and the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the SAR remain unchanged. Similarly, the possibility for an accident or malfunction of a different type was not created. None of the changes associated with this activity involved Technical Specifications or the basis for any Technical Specification. Therefore, the margin of safety as defined in the basis of any Technical Specification was unchanged.

OLCR 99-021
(USQE 1999-0043)

TITLE: Administrative and Sample Station Revisions to the Offsite Dose Assessment Manual (ODAM)

DESCRIPTION: This change to the ODAM revised the Radiological Environmental Monitoring Program (REMP) sample locations based on the annual review of REMP data as required by the ODAM. In addition, this revision implemented numerous editorial and human factor changes to improve the overall quality, accuracy, and useability of the ODAM. The changes made to the ODAM maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50 Appendix I, and did not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations. Additionally, the changes did not alter the radiological effluent controls and radiological environmental monitoring activities required by the CNS Technical Specifications (TS).

SAFETY
EVALUATION: The changes to the ODAM are administrative and do not affect the design, function, system interface or operating parameters of plant systems, structures and components (SSCs) important to safety. The proposed changes do not affect any assumptions for accidents evaluated in the SAR and do not cause previously evaluated accidents to shift to a higher frequency class. Therefore, the probability of an accident evaluated in the SAR has not increased. Since the changes do not degrade or prevent mitigative actions described or assumed in the SAR and do not impact fission product barriers, the consequences of an accident previously evaluated in the SAR have not increased. The probability of a malfunction of equipment important to safety previously evaluated in the SAR has not increased because the performance of safety systems assumed to function in the SAR accident analysis has not been affected, challenges to safety systems such that system performance is degraded below design basis have not increased, and there are no revisions to system lineups such that system performance is impacted. As stated above, the changes are administrative and do not affect the design, function, system interface or operating parameters of plant SSCs important to safety. Since no equipment was modified or installed by this change, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR have not increased. For these same reasons, the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR has not been created. The proposed changes do not affect the margins of safety evaluated against Safety Limits, Limiting Safety System Settings, Limiting Conditions of Operation, and design parameters for SSCs. Initial conditions, time calculational models and codes, allowances for instrument drift and system response during operating, transient and accident conditions are unaffected by this change. Administrative and programmatic responsibilities outlined in TS are also unaffected by this change. Accordingly, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

OLCR 99-027
(USQE 1999-0065)

TITLE: Technical Specification (TS) Bases Surveillance Requirement (SR) 3.7.3.4

DESCRIPTION: This revision to TS Bases SR 3.7.3.4 removed the nominal setpoint values that were listed for three Reactor Equipment Cooling (REC) pressure switches (which provide for automatic isolation of non-critical loads on low header pressure) and replaced them with the analytical limits. Nuclear Engineering Department Calculation (NEDC) 92-050X, NEDC 97-087, and Design Change 93-057 were also added to the references section, as the design change is the source of the analytically determined limit, NEDC 97-087 validates this limit, and NEDC 92-050X is the source of the actual (or nominal) setpoint values. Since nominal setpoints can change depending on experience, drift, refined instrument inaccuracies, etc., the analytically determined limits are considered to be more appropriate from a human factors and useability standpoint. This revision was an administrative change that did not alter the safety function of the pressure switches, nor the method of performing that function.

SAFETY
EVALUATION: This revision is considered to be administrative in nature. The nominal and analytical setpoints were neither generated nor changed by this activity. Instead, the focus was shifted in the TS Bases Surveillance Requirement from the nominal setpoints, which take into account numerous variables, to the analytical limit which services as the primary input to the nominal setpoints. The function of the REC pressure switches is to initiate valve closure on low header pressure (indicating a possible passive failure of a non-critical line) such that only the critical loops receive REC water. The switches themselves are not accident or transient initiators and will continue to perform their design function. Accordingly, this activity has no effect on the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. Similarly, the possibility for an accident or malfunction of a different type has not been created. Since the switches will continue to perform their design function, and including the analytical limits in the TS

Bases assures that the function of REC to provide cooling water to supported systems is maintained, the margin of safety as defined the basis for the TS has not been reduced.

OLCR 2000-007
(USQE 2000-0012)

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

DESCRIPTION: Under this activity, the Technical Specification Bases were revised to accommodate High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) system testing at reactor pressures below that at which system operability is required. Previously, "adequate reactor steam pressure" for testing was defined as greater than or equal to 150 psig, which was also the reactor pressure at which system operability is required. "Adequate reactor steam pressure" for testing was redefined as greater than or equal to 145 psig (while the pressure at which operability is required was not changed). As a result of this change, HPCI and RCIC functionality can be demonstrated before system operability is required. This enhances nuclear safety in that operational problems can be identified and corrected before the systems are required to be operable.

SAFETY EVALUATION: Since this change does not alter the design function or requirements of HPCI or RCIC and does not affect accident or transient precursors, there is no increase in the probability of occurrence or the consequences of an accident previously evaluated in the SAR. As the reduced pressure for testing is still within the design basis for HPCI and RCIC, the probability of occurrence or the consequences of an equipment malfunction previously evaluated in the SAR have not increased. Likewise, testing the systems prior to when they are required to be operable does not create the possibility for an accident or malfunction of a different type. Finally, since the acceptance criteria for operability are unchanged, the margin of safety as defined in the basis for any Technical Specification is unchanged.

TRMCR 1999-003
(USQE 1999-0020)

TITLE: New TRM Section 3.1

DESCRIPTION: This revision to the Technical Requirements Manual (TRM) added a new section (i.e., Section 3.1) to establish acceptance criteria for control rod scram time testing for reactor pressures less than 800 psig. Specifically, acceptance criteria for scam time testing at zero reactor pressure was added, which provides a high probability of meeting the Technical Specifications (TS) acceptance criteria for insertion at reactor pressures greater than or equal to 800 psig. The acceptance criteria specified originates from CNS Startup Test #5, which provides actual test data for single rod scrams at reactor pressures of 0, 600, 800, 930 and 1000 psig. This TRM change did not affect the TS scram limits for reactor pressure greater than or equal to 800 psig, which are used in accident analyses.

SAFETY EVALUATION: Accidents and transients that terminate in or assume a reactor scram were considered for possible effect. Of special importance is the Control Rod Drop Accident, because it can be more severe in the low pressure reactor startup range, where the new TRM limit applies. While the new TRM limit is adequate to demonstrate that the Control Rod Drive is capable of rapid insertion with only the accumulator pressure, it is deemed conservative because it is significantly faster than the Technical Specification limit used in the Control Rod Drop Accident analysis. Other accidents would not be affected since they are analyzed at full power conditions. Since scram timing at zero reactor pressure will not cause the system to be operated outside of its design limits or degrade equipment, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR have not been increased. Adding timing criteria for a depressurized scram does not involve an accident initiator and no new failure modes are being introduced. Therefore, the possibility for an accident or malfunction of a different type than any

evaluated previously has not been created. As previously noted, the actual Technical Specification limits are not affected. As a result, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

PROCEDURE CHANGES

Procedure 2.2.6, Revision 41
(USQE 1999-0033)

TITLE: Condensate System

DESCRIPTION: This change proceduralized the installation and removal of oxygen injection equipment for the Condensate and Feedwater systems, thus eliminating the need for a Temporary Configuration Change whenever it is necessary to inject oxygen in these systems. Oxygen injection is required to maintain dissolved oxygen levels in condensate and/or feedwater effluents within fuel warranty limits.

SAFETY EVALUATION: The small amount of oxygen injected does not alter the manner in which the Condensate or Feedwater systems operate and is insignificant compared to the amount of oxygen produced during normal BWR operation. The injection pathway utilizes a spare sample line and valves that are part of the system and rated to withstand the requirements of oxygen injection. The injection apparatus is located in an area where seismic concerns are not present and equipment important to safety is not present. Therefore, the probability of occurrence or consequences of a previously evaluated accident, transient, or plant event are not increased. Since the manner of system operation is not changed and the chemical parameters of the systems will be maintained within the requirements stated in station procedures, the probability of occurrence or consequences of a malfunction of equipment important to safety are not increased. The requirements for dissolved oxygen control are based on the minimization of corrosion rate to ensure system longevity. The systems are being maintained to perform their designed functions and the oxygen injection equipment is not associated with any other plant system functions or operation; therefore, the possibility of a different type of accident or equipment malfunction than previously evaluated is not introduced. The level of oxygen in the condensate or feedwater does not impact the margin of safety as defined in the basis for any Technical Specification.

Procedure 2.2.27, Revision 28
(USQE 1999-0054)

TITLE: Equipment, Floor, and Chemical Drain System

DESCRIPTION: This procedure was revised to provide guidance for operation of Primary Containment sump pumps in the event that both "G" sump pumps fail. The failure of both "G" sump pumps will cause "G" sump (identified leakage) to overflow into "F" sump (unidentified leakage). The "G" sump is the Drywell Equipment Drain Sump and "F" sump is the Drywell Floor Drain Sump. In the event that both "G" sump pumps fail to operate, identified leakage into "G" sump will not be measurable by the normal method. The unidentified leakage rate going into "F" sump would increase by the amount added from "G" sump and the identified leakage rate will be considered as zero. Therefore, the final result is an increase in unidentified leakage rate, a zero identified leakage rate, and no change in total leakage rate. The use of "F" sump to determine unidentified and total leakage, without separate determination of identified leakage (other than considering it zero) is considered a conservative action to ensure operability of the reactor coolant pressure boundary leakage detection system and compliance with Technical Specification 3.4.4 limits.

SAFETY EVALUATION: The loss of both "G" sump pumps and allowing "G" sump to overflow to "F" sump does not initiate, nor is it a precursor to, any accident or abnormal transient evaluated in the SAR. All leakage going to "F" sump will continue to be monitored and subject to Technical Specification 3.4.4 limits. Primary Containment Isolation Valves RW-AOV-AO94 and RW-AOV-AO95 will function as designed on a Group 2 isolation signal. In addition, the Post Accident Sampling System and all associated valves will function as designed. In the event that both "G" sump pumps fail, the Drywell Equipment Sump

Heat Exchanger will not perform its function of cooling sump liquid prior to entering the Radwaste Equipment Drain Process System. However, the heat exchanger does not provide a safety function. The "G" sump temperature indicator will be degraded and not provide an accurate temperature indication of the liquid in the sump. However, the drywell equipment sump temperature indicator is just one of many indications the operators have available that could assist them in identifying changes in reactor coolant inventory. The loss of the heat exchanger and temperature indicator will not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety. Increased flow to "F" sump will cause the "F" sump pumps to cycle more frequently. However, the increase in "F" sump pump cycles will not present any additional burden on these pumps. "F" sump will continue to perform its function as described in the USAR. No additional or different types of accidents or equipment malfunctions are created by allowing "G" sump leakage to flow to "F" sump. There will be a higher demand placed on the Radwaste Floor Drain Process System; however, based on bounding calculations, the discharge is below any 10CFR50 Appendix I limits. In-leakage to both sumps will still be monitored per current specifications and there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

2.2.65, Revision 42
2.2.65.1, Revision 21
6.LOG.601, Revision 23
6.LOG.602, Revision 10
(USQE 2000-0030)

TITLE: REC Temperature Limits

DESCRIPTION: Technical Specifications (TS) Surveillance Requirement (SR) 3.7.3.2 requires verifying that the Reactor Equipment Cooling (REC) Heat Exchanger Discharge Temperature is less than or equal to 95 degrees F every 24 hours. This SR is performed by reading the indicated temperature. Prior to this change, an administrative limit was placed on the indicated temperature which was less than the 95 degrees F to account for instrument uncertainties. However, analysis has shown that REC will continue to perform its safety-related design function of providing cooling water for essential equipment and maintain temperatures within limits even at actual REC temperatures of up to 99.4 degrees F. Therefore, this change removes the administrative limits.

SAFETY EVALUATION: The normal operation of the REC System is not an accident initiator. The change to the administrative limit for REC temperature will not prevent the REC system from fulfilling its support function of providing cooling to the Emergency Core Cooling System compartments and maintaining the temperatures in the compartments below limits, nor will it impact the analyses or performance of containment isolation systems. Accordingly, the probability of occurrence and consequences of an accident or malfunction of equipment important to safety are not increased. As stated above, REC is not an accident initiator and will continue to perform its safety design function. Since no new components are installed and the relationship between existing components and the methods of operating any components is not changed, the possibility for an accident of a different type has not been created. Additionally, the change does not increase the possibility of a failure of the individual components in the REC system. The piping, pumps, valves, heat exchangers, and associated instrumentation have temperature ratings above the normal and expected temperatures in the REC System. Therefore, the possibility of a malfunction of a different type has not been created. Calculations demonstrate that the safety support function of REC can still be performed with the combination of instrument uncertainty and the analyzed REC temperature limits and the margin of safety has not been reduced.

Procedure 2.4.5.2.1, Revision 6
Procedure 2.4.5.2.2, Revision 6
Procedure 2.4.5.2.3, Revision 6
Procedure 2.4.5.2.4, Revision 6
(USQE 1999-0029)

TITLE: DEH Malfunctions Mode I (2.4.5.2.1)
DEH Malfunctions Mode II (2.4.5.2.2)
DEH Malfunctions Mode III (2.4.5.2.3)
DEH Malfunctions Mode IV (2.4.5.2.4)

DESCRIPTION: The subject procedures were revised in accordance with General Electric Service Information Letter (SIL) 614, Revision 1, Backup Pressure Regulator. This SIL revision states that a downscale pressure regulator failure (without backup) from full power has been shown to be bounded by other transient events analyzed for establishing full power fuel operating limits. Therefore, a downscale pressure regulator failure at full power (greater than or equal to 90 percent power) does not represent a safety concern. Should a plant discover a backup regulator problem while operating in this high power range, continued operation near full power is justifiable until the next planned power reduction. Therefore, the subject procedures were revised to indicate that the unanalyzed power range is 25 percent to 90 percent, and if a failure occurs at greater than or equal to 90 percent power, the transient is bounded by transient events analyzed for establishing the full power fuel operating limits and continued operation is allowed.

SAFETY EVALUATION: The Digital Electro-Hydraulic (DEH) pressure control system plays a role in two analyzed abnormal transients (DEH Pressure Controller Output Fails Low, and DEH Pressure Controller Output Fails High). These changes provide guidance to the operator on dealing with a malfunctioning of a DEH pressure controller; however, this guidance does not increase the probability of a DEH pressure controller failure. This change has no effect on the probability or consequences of any other analyzed accidents, abnormal events, or special events. Continuing to operate the plant with reactor power greater than or equal to 90 percent power after one of the two independent pressure controllers has failed does not increase the probability that the other controller will fail. Failure of one or both DEH pressure controllers does not increase the probability that any other equipment important to safety will fail. This change does not create any new initiating events or failure modes that could degrade the barriers to fission product release. No equipment additions or modifications are being made. There are no Technical Specifications applicable to the DEH pressure controllers. The bounding analysis has not been changed and this assures that the margin of safety has not been reduced.

Procedure 3.35.1 (Revision 0)
(USQE 1998-0039)

TITLE: Installation of Quick Connects on Air Operated Valves

DESCRIPTION: This procedure provides for the installation of quick connects and isolation valves on air operated valves (AOVs) on an as-needed basis. The quick connects will facilitate rapid connection of test equipment to AOVs for performing testing, troubleshooting, or calibration activities. The isolation valve provides for leak tight redundancy during normal operation and minimizes air loss and the potential effects associated with connecting the test equipment. The procedure requires an engineering evaluation of each application prior to installation and further requires the components used and resulting configuration meet or exceed the safety and seismic classification of the affected AOV. The most common points to be monitored on the AOV are air supply pressure, actuator pressure, and pneumatic control signal. The procedure scope specifically excludes the Main Steam Isolation Valves and the Control Rod Drive inlet and outlet AOVs.

SAFETY
EVALUATION: The AOVs that are initiators of an abnormal operational transient, incident of moderate frequency, anticipated operational occurrence or accident and limiting fault as described in the USAR are excluded from the scope of the procedure. Although the quick connects and isolation valves impact the pressure boundary of the air operator, engineering evaluation on a case-by-case basis is required to ensure that the components added meet or exceed the system design and classification for the affected AOV. As a result, the probability of an accident previously evaluated in the SAR has not been increased. The majority of AOVs required to mitigate an accident are designed to fail to the safe position or are equipped with a safety-related accumulator. Since the addition of a quick connect and isolation valve would not change or interfere with this fundamental design philosophy, the consequences of an accident previously evaluated in the SAR have not been increased. As the addition of the quick connect and isolation valve does not affect the function of the AOV or the probability of failure, the probability or consequences of a malfunction of equipment important to safety previously evaluated in the SAR have not increased. This is further supported by the fact that, as noted above, the components added meet or exceed the system design and classification for the affected AOV. This same fact, along with the redundancy provided by the isolation valve, ensures that the possibility of a different type of accident or malfunction has not been created. As the operating characteristics of the affected AOVs will not be changed, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

Procedure 6.EE.604, Revision 7
(USQE 1999-0031)

TITLE: 125V Battery Charger Performance Test

DESCRIPTION: This procedure was revised to ensure adequate voltage is available to safety-related components during the duration of the battery charger performance test. This change specifically provides for removal of a plant air compressor from service to reduce the loading on the Motor Control Center in order to accommodate the increase in loading caused by the addition of the battery charger. This will provide additional voltage margin to prevent overloading the bus if a design basis event were to occur during the charger performance testing.

SAFETY
EVALUATION: Removing a single air compressor from standby service is bounded by the accident analysis which assumes failure of all air compressors. Air operated equipment required for engineered safeguards and reactor protection are provided with accumulators that provide compressed air for continued operation or have been designed to fail-safe in the event of an air system failure. Therefore, removing a single air compressor from service does not increase the probability of an accident evaluated in the SAR. Both the electrical AC system and air system responses remain bounded by existing analysis; therefore, no new failure modes or failures of barriers to fission product release are introduced that would increase the probability or consequences of a previously evaluated accident. This change does not install any new equipment nor does it change the form-fit-function of existing equipment. Removing a single air compressor from standby service does not prevent the air system from performing its design function. The air compressors are not associated with accident mitigation as described in the SAR. Removing an air compressor from service does not increase the probability or consequences of a malfunction of equipment important to safety. No new types of accidents or equipment malfunctions are introduced. The margin of safety is not reduced as the air system will still be capable of performing its design basis functions and the electrical system response remains bounded by existing analysis.

Procedure 7.2.82, Revision 0
(USQE 1999-0056)

TITLE: Removal and Installation of Heating Coil (AC-T-1A)

DESCRIPTION: This is a new procedure to control the removal and reinstallation of the heating coil within the Instrument Shop Heating, Ventilating, and Air Conditioning (HVAC) unit as needed in hot weather. Removing the heating coil for the summer months increases the efficiency of the fan coil unit and is a change to the plant as described in the USAR. The Instrument Shop is also the Alternate Operational Support Center (AOSC) as governed by the Emergency Plan. The housing of the fan coil unit acts as a barrier to mitigate the intrusion of contamination into the AOSC during post accident conditions. The affected HVAC system will be inoperable during coil removal and reinstallation.

SAFETY EVALUATION: This change does not affect any component or system that is a precursor to a previously evaluated accident or used to mitigate the consequences of a previously evaluated accident. Other than during the actual process of removing and reinstalling the coil, the availability of the AOSC will be unaffected. There are no plant control devices located in the AOSC. The change does not affect any component or system that is important to safety or used to mitigate the consequences of a malfunction of a component important to safety. No different types of failure modes are introduced. Any accidents that could occur would be bounded by previously analyzed steam leaks or fires. This change does not involve nuclear safety-related components and does not affect any component or system that is not a normal industrial device. No new types of malfunctions are created. The change does not affect any component or system that is in contact with any system that would have an effect on a margin of safety. Therefore, the margin of safety as defined in the basis for any Technical Specification cannot be reduced.

Procedure 10.15 (Revision 14)
(USQE 1999-0035)

TITLE: TIP System Reproducibility and Core Power Symmetry Test

DESCRIPTION: CNS Procedure 10.15, Traversing Incore Probe (TIP) System Reproducibility and Core Power Symmetry Test, was revised to remove the requirement to conduct the test every cycle. This testing was GE-specified for the initial plant startup to confirm that the system readings and symmetries were within the expected range. The tests were continued on a cycle basis as an NRC commitment made in 1979. Plant improvements have reduced the need for such periodic testing. Computer model enhancements have increased the fuel vendor's ability to design and predict fuel cycle behavior. The TIP system has transitioned from neutron to gamma flux detectors, which increases accuracy. On-line computer processing and monitoring have improved with the 3D Monicore system. A commitment change was processed in support of this revision.

SAFETY EVALUATION: This revision will not increase the probability of an accident, transient, or other event evaluated in the SAR because the likelihood of these analyzed events occurring does not depend upon the performance of this testing. Further, the TIP system performance characteristics will not change due to the performance frequency of the procedure. Since the vendor's calculations of fuel cycle thermal limits are based on the system design characteristics and not on the performance of any cycle specific testing, no increase in the consequences of an accident, transient, or other event evaluated in the SAR has resulted. The TIP system operability is not contingent upon the performance of this testing. Therefore, this revision does not increase the probability of a malfunction of equipment important to safety previously evaluated in the SAR. Fuel thermal limits help ensure that one or more fuel rods do not experience fuel cladding failure during evaluated equipment malfunctions, and the TIP system assumed accuracy is one of many inputs to these limits. But as noted above, these system accuracy assumptions are based on the system design characteristics and not on the performance of any cycle

specific testing. Accordingly, this revision does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR. For these same reasons, the possibility for an accident or malfunction of a different type has not been increased. Since the cycle-specific results of the TIP uncertainty and symmetry tests are not used to adjust the Technical Specification limits for Minimum Critical Power Ratio (MCPR), Average Planar Linear Heat Generation Rate (APLHGR), and Linear Heat Generation Rate (LHGR), this revision does not reduce the margin of safety as defined in the basis for any Technical Specification.

Procedure 14.7.2.1, Revision 3
Procedure 14.7.2.2, Revision 3
Procedure 14.7.3, Revision 4
(USQE 1999-0032)

TITLE: DEH Throttle Pressure Transmitter MS-PT-81A Calibration (14.7.2.1)
DEH Throttle Pressure Transmitter MS-PT-81B Calibration (14.7.2.2)
DEH Backup Pressure Controller Removal and Return to Service During Operation (14.7.3)

DESCRIPTION: These procedures were revised to include a step to inform the operator that one Digital Electro-Hydraulic (DEH) controller will be removed from service and to enter the appropriate abnormal procedure for DEH malfunction, if necessary. A note was added to state that analysis does not exist to support thermal limits at power operation between 25 percent and 90 percent with one DEH pressure controller failed or excessive difference between in service and backup pressure controllers. Furthermore, if a failure occurs at greater than or equal to 90 percent power, the transient is bounded by other transient events analyzed for establishing the full power fuel operating limits. (Reference General Electric SIL 614, Revision 1).

SAFETY EVALUATION: The DEH pressure control system plays a role in two analyzed abnormal transients (DEH Pressure Controller Output Fails Low, and DEH Pressure Controller Output Fails High). These changes provide guidance to the operator on dealing with a malfunctioning of a DEH pressure controller; however, this guidance does not increase the probability of a DEH pressure controller failure. This change has no effect on the probability or consequences of any other analyzed accidents, abnormal events, or special events. Continuing to operate the plant with reactor power greater than or equal to 90 percent power after one of the two independent pressure controllers has failed does not increase the probability that the other controller will fail. Failure of one or both DEH pressure controllers does not increase the probability that any other equipment important to safety will fail. This change does not create any new initiating events or failure modes that could degrade the barriers to fission product release. No equipment additions or modifications are being made. There are no Technical Specifications applicable to the DEH pressure controllers. The bounding analysis has not been changed and this assures that the margin of safety has not been reduced.

Procedure 15.AR.301, Revision 0
(USQE 1999-0023)

TITLE: Condenser Air Removal Isolation Valves Functional Test

DESCRIPTION: This new procedure was developed to provide guidance for functional testing of the condenser air removal valves. Design Change 91-088 had previously moved the condenser isolation safety function from the air removal isolation valves to the mechanical vacuum pump suction and discharge valves. Surveillance testing of the condenser air removal valves had been discontinued at that time. However, due to concerns that at least one of the valves may currently be closed rather than open, this procedure was developed to individually stroke the valves to demonstrate that the valve actuator and butterfly orientation is correct. The procedure has no set frequency and may be performed anytime at the request of Operations or Engineering when affirmation of orientation is needed.

SAFETY
EVALUATION: This activity could initiate a loss of condenser vacuum, which is a transient described in the USAR. However, this procedure does not increase the possibility of a loss of condenser vacuum as appropriate controls are put in place by the procedure. Momentary cycling of the condenser air removal isolation valves will maintain the system within normal design and operation. The off-gas isolation function will remain unchanged as a result of this activity, thus any mitigation activities associated with accidents and transients will remain unchanged. This activity will not jeopardize the operation of any components in the Steam Jet Air Ejector (SJAE) system. If this activity is successful in ultimately obtaining better SJAE and condenser performance, this may result in higher off-gas radiation in the SJAE room during the winter. However, any changes will be within limits already used for summer operation, which is the time of highest radiation levels for the off-gas. The normal gamma activity rate assumed in the development of the setpoint gross gamma activity rate level for high off-gas activity remains unchanged. With the 30-minute hold-up line and Augmented Off-Gas (AOG) charcoal bed hold-up, sufficient delay time occurs and no change in dose released to the public will occur. No permanent hardware changes are being performed and no new equipment failure mechanisms are being introduced. Individual valve closures will be conducted in a slow, controlled fashion with provisions for restoring system operation to a stable state should undesirable responses be observed. The condenser air removal isolation valves are no longer relied upon to mitigate the consequences of a malfunction of equipment important to safety. Since operation of the system is within normal design and because no new equipment is being added, the possibility of a new or different type of accident, transient, or equipment malfunction is not introduced. The margin of safety as defined in the Technical Specifications is not reduced as a result of procedure performance. None of the activity rates for AOG are being altered and the highest expected operating rate level is not being altered.

MISCELLANEOUS CHANGES

USQE 1998-0073

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

DESCRIPTION: In 1981 the 161kV offsite AC power supply was modified independent from but concurrent with Minor Design Change (MDC) 81-53, which modified control cables from the plant to the switchyards. A CNS Special Test Procedure (STP) 81-6 was performed in cooperation with NPPD substation crews to complete acceptance testing of the substation modification. At the time, these activities were evaluated as a "change to the facility" and a "test" respectively in accordance with 10CFR50.59. These 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. Specifically, these evaluations did not recognize that the new routing of the 161kV line placed them both beneath the CNS - Booneville 345kV line with one transmission tower between them. This aspect of the modification was identified during 1998 as part of a review for unauthorized modifications. As a result, this USQE was performed.

SAFETY EVALUATION: Due to existing design margins in the CNS - Booneville tower #2, it was concluded that the reliability of the AC sources was not degraded by the modification. Therefore, the probability of an accident previously evaluated in the SAR has not been increased. The change was determined to have no adverse effect on the ability of the critical service portions of the auxiliary AC distribution system and the emergency diesel generators (EDGs) to satisfy the requirements of General Design Criteria 17 and IEEE-308. As a result, the consequences of an accident previously evaluated in the SAR have not been increased. As previously noted, the reliability of the AC sources was not degraded by the modifications and, therefore, the probability of a malfunction of equipment important to safety has not been increased. Since the loss of the AC power sources is bounded by the analysis of a loss of offsite power with one EDG also failed, the consequences of a malfunction were not increased by the change. For this same reason, the possibility for an accident or malfunction of a different type has not been created. This modification did not change the operating parameters of systems that provide safety functions to safely shut down the reactor, maintain the safe shutdown condition, and operate all auxiliaries necessary for station safety. Accordingly, there was no reduction in the margin of safety as defined in the basis for any Technical Specification.

USQE 1999-0027

TITLE: Evaluation of the Continued Use of Caution Tag Order (CTO) 94-113 on the Controls of the Radwaste Building Clean Area Exhaust Fans

DESCRIPTION: The Radwaste Building Ventilation System is designed to route air flow from areas of lesser to progressively greater areas of radioactive contamination. This objective is accomplished in part through the use of the Radwaste Building Clean Area Exhaust Fans HV-FAN-(EF-RW-1C) and HV-FAN-(EF-RW-lD). These fans are a matched redundant set. By design, one fan is to be maintained in the RUN position and the other in the STANDBY position, ready to automatically start on a low flow condition. In 1994 the control system for the Radwaste Building Ventilation System was modified. This modification inadvertently introduced a yet unresolved conflict in the control logic for these fans. As a result, as soon as the second fan is selected to the STANDBY position, it starts. To keep from running the two fans in parallel, the control selector switch for the non-running fan has been placed in the OFF position. A Caution Tag (CTO 94-113) has been placed on the switch directing Control Room operators to place the fan in the RUN position should an actual low flow condition occur. Due to the extended duration of this condition, it has been evaluated under the requirements of 10CFR50.59 as a defacto change to the facility.

SAFETY
EVALUATION: The clean area exhaust fans are not considered to be accident initiators. Therefore, maintaining the standby fan in the OFF position has no effect on the probability of occurrence of an accident previously evaluated in the SAR. Since the clean area exhaust fans have no interaction or impact on the Control Room envelope, the consequences of an accident (i.e., Control Room dose) have not been increased. The fans do not provide cooling to any safety-related equipment, therefore, the interruption in flow (as a result of having to recognize a low flow condition and starting the second fan) would not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the SAR. Similarly, the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR has not been created. The clean area exhaust fans have no input to the margin of safety as defined in the basis for any Technical Specification and, therefore, placing the standby fan in the OFF position has no impact on these margins.

Problem Identification Report 4-05838
(USQE 2000-0008)

TITLE: Dry Tube/LPRM Bending Hoist on Refuel Bridge Monorail

DESCRIPTION: Temporary Design Change (TDC) 93-018 installed a second hoist on the monorail of the refueling bridge to be used during removal and bending (for storage and disposal) of Source Range Monitor/Intermediate Range Monitor dry tubes and Local Power Range Monitor (LPRM) assemblies. The subject Problem Identification Report was initiated because it was felt that the TDC did not adequately address the following issues:

- The hoist motors do not have heaters to prevent condensation damage. (The other hoist motors on the refuel bridge do have heaters.)
- The monorail mounted hoist does not have the interlocks required by the USAR and the Technical Specifications.
- The hoist is not controlled through the operator's console selector switch.
- The hoist is not fused other than for the fuse at the motor control center (MCC). A failure would shut down the entire refuel bridge and there is no disconnect switch to isolate the hoist.
- The installation on the monorail has reduced the movement range of the other monorail hoist. There is no documentation to indicate that this is acceptable.
- The MCC fuse does not meet the National Electric Code (NEC) unless only one hoist is allowed to operate at a time. There is no interlock on the LPRM bending hoist to prevent more than one hoist from operating at a time.
- No safety evaluation was performed regarding the bending of the LPRMs over the reactor core with the aid of the bender and LPRM hoist. This seems to increase the probability of a dropped object into the core. Previously, the LPRM assemblies were cut for storage and disposal, and this activity was done in the fuel pool and not over the core.
- Procedure 7.3.11, Refueling Platform Hoists and JIB Crane Load Cells Calibration, only calibrates one of the monorail hoists.

This USQE justified the continued temporary use of the second bending hoist for Refueling Outage 19 (RFO-19). As discussed in the USQE, the lack of motor heaters is a long term reliability issue, and not a safety concern. The interlocks required by the USAR and Technical Specifications are designed to provide reactivity controls during refueling operations. Since the bending activities are not done during refueling operations, the lack of interlocks on the second hoist is not a violation of USAR and Technical Specification requirements. The console selector switch is used to select which control station provides refueling bridge and trolley traversing power. Since the hoist control pendant has no control for bridge and trolley movement, it is not required to be controlled through the operator's console. Although the hoist is on a common fuse with the refuel bridge, the circuit does not provide

essential power and a fault or trip would not present a safety concern. While the addition of the second hoist may have decreased the range of travel for the primary hoist, the primary hoist can still reach any part of the core or storage pool as required. Therefore, the functionality of the primary hoist has not been impacted and there is no safety issue. The current configuration does not meet the NEC, however, since the hoists are not fully loaded, both can operate concurrently without tripping the power circuit. While this configuration would not be acceptable for a permanent installation, it is acceptable for the TDC because, as noted above, the circuit does not provide essential power and a fault or trip would not present a safety concern. The consequences of dropped objects into the core during bending activities are bounded by the dropped fuel assembly during refueling accident. Lastly, since the second hoist is not used for refueling activities, it does not fall under the scope of Procedure 7.3.11.

Based on the USQE results it was concluded that there were no safety issues associated with using the hoist installed under TDC 93-018 during RFO-19.

SAFETY EVALUATION: This hoist installed under TDC 93-018 does not affect any initiator to transients and special events described in the SAR nor is it used during refueling activities. Therefore, the probability of an accident previously evaluated in the SAR has not increased. The tube bending hoist does not interface or impact with accident mitigation systems or assumptions. The fuel handling accident bounds the consequences of dropping an LPRM assembly. Therefore, the consequences of an accident previously evaluated in the SAR have not increased. The tube bending hoist is not used with the fuel handling hoist and does not perform any function important to safety. Further, the added hoist does not affect plant operating modes, operate equipment outside of design, or create any new equipment interface that affects equipment important to safety. Therefore, the probability of occurrence and the consequences of a malfunction of equipment important to safety previously evaluated in the SAR have not increased. Further, since the added hoist does not create a new mode of plant operation, operate equipment outside of design, or create any new equipment interface that affects equipment important to safety, the possibility of an accident or malfunction of a different type has not been created. There are no Technical Specifications governing the tube bending hoist. Therefore, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

<u>USQE 2000-0016</u>

TITLE: RHR Sample Valves De-energized Closed for Failing to Meet IST Requirements

DESCRIPTION: Valves RHR-SOV-SSV96 and RHR-SOV-SSV60 are the Residual Heat Removal (RHR) heat exchanger sample point isolation valves for the RHR heat exchangers A and B, respectively. These valves are part of the post-accident sampling system (PASS) as discussed in Section X-15 of the SAR. As these valves had a history of failing to meet inservice testing (IST) requirements and operability could not be assured, they were declared inoperable and isolated. The valves are not considered to be primary containment isolation valves and do not have a safety function in the open position. They have a safety function in the closed position in that they prevent diversion of Low Pressure Coolant Injection (LPCI) flow to the reactor.

SAFETY EVALUATION: Under this activity, the subject valves were declared inoperable and isolated in the safe (i.e., closed) position. Accordingly, the probability of occurrence of an accident or equipment malfunction previously evaluated in the SAR was not increased. By maintaining the valves in the closed position, there is no potential for the valves failing to close when required, thus preventing the diversion of LPCI flow from the reactor. Additionally, as there are other sources within the PASS for obtaining samples, adequate information regarding degraded core conditions would still be available in the event of an accident. Therefore, the consequences of an accident or equipment malfunction previously evaluated in the SAR were not increased. Further, since the valves were isolated in the safe position, the possibility for an accident or equipment malfunction of a different type was not created. Since the

valves are not primary containment isolation valves and the PASS function could still be achieved, the margin of safety as defined in the basis for any Technical Specification was not reduced.

PIR 4-08273
(USQE 2000-0017)

TITLE: Extended Operation with Primary Containment Penetration X33d Isolated

DESCRIPTION: Under this activity, a manual valve was closed isolating primary containment penetration X33d due to the failure of excess flow check valve RR-CV-28CV to pass a surveillance test and the inability to safely test the replacement valve. Isolation of this penetration renders the "B" Reactor Recirculation Pump (RRP) differential pressure instrumentation inoperable. As a result, the penetration will remain isolated until the next available opportunity to perform the post maintenance test during the Refueling Outage 20 (RFO-20) operational hydrostatic test. The "B" RRP differential pressure instruments are nonessential and provide indication only to operators in the main control room. They perform no safety functions. As there are other control room parameters that can be used to assess pump performance, operation without the "B" RRP differential pressure instrumentation will have no effect on safety. Additionally, isolation of the penetration X33d will not adversely affect the safety function of primary containment for this penetration. Temporary changes to the associated operating procedures have been made to reflect operation with penetration X33d isolated.

SAFETY EVALUATION: Since the inoperable containment penetration was isolated in accordance with the Technical Specifications and the "B" RRP differential pressure instrumentation is not relied upon to mitigate an accident, probability of occurrence and the consequences of an accident previously evaluated in the SAR have not been increased. Since containment penetration X33d will be positively isolated and the "B" RRP differential pressure instrumentation only provides the function of nonessential indication and is not relied upon for operation of equipment important to safety, the probability of occurrence and the consequences of an equipment malfunction previously evaluated in the SAR have not been increased. Since there are numerous alternate indications to compensate for the inoperable "B" RRP differential pressure instrumentation and the penetration is positively isolated, the possibility of an accident or malfunction of a different type has not been created. Finally, isolating the penetration in accordance with the Technical Specifications also ensures that the margin of safety as defined in the basis for any Technical Specification has not been reduced.

SSAR Change Package 00-01
(USQE 2000-0029)

TITLE: SSAR Changes Due to 8-Hour Limit of Diesel Fire Pump Fuel

DESCRIPTION: Under this activity, the Safe Shutdown Analysis Report (SSAR) was revised to document the design basis of the 8-hour limit for the diesel fire pump fuel supply and to establish said limit as a functional requirement for Service Water (SW) system operation. The diesel fire pump is used to supply gland seal water from the Fire Protection system to the SW pumps during fire scenarios concurrent with a loss of offsite power. In addition, manual actions necessary to maintain the operability of the Division II SW pumps after the 8-hour supply of diesel fuel is exhausted were defined. The SSAR demonstrates the capability of CNS to mitigate the consequences of fire induced equipment malfunctions within the operational capabilities of remaining plant equipment. The proposed changes do not affect the current compliance strategy for various CNS Appendix R fire areas or add any system, structure, or component that may be utilized to achieve post-fire safe shutdown. The aforementioned changes only apply to the special event of a fire where a station shutdown from outside the Control Room is required. The affected Emergency Procedure was also revised to include the manual actions described above.

SAFETY
EVALUATION: The Fire Protection and SW systems are not initiators of any accidents, abnormal transients, or other events evaluated in the USAR. As these systems will continue to operate as designed, the probability of an accident previously evaluated in the SAR has not increased. The changes to the SSAR serve to demonstrate capability of the CNS design to effectively respond to a fire event and achieve safe shutdown as defined by 10CFR50 Appendix R. Accordingly, the consequences of an accident previously evaluated in the SAR remain unaffected. Under this activity, the design basis limitations of the diesel fire pump are documented and compensatory actions are established to mitigate the consequences of running out of diesel fuel during a fire event. As the bounding assumptions of the Appendix R analysis remain valid, the probability of a malfunction of equipment important to safety as previously evaluated in the SAR has not increased. Further, the consequences of fire induced equipment malfunction remain within the previously analyzed results for an Appendix R scenario. Accordingly, the consequences of an equipment malfunction previously evaluated in the SAR remain unaffected. Since this activity serves only to document the existing SSAR design basis, and the specified manual actions do not result in equipment operation outside analyzed system design basis limitations, the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created. Similarly, as Technical Specification equipment operation in response to a fire event remains within analyzed design basis parameters, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

ENCLOSURE 2

SUMMARY OF REGULATORY COMMITMENT CHANGES

REGULATORY COMMITMENT CHANGES

The following regulatory commitments were revised based upon evaluation performed in accordance with Nuclear Energy Institute (NEI) 98-05 Revision 2, December 1995, "Guideline for Managing NRC Commitments." This summary does not include those commitment changes where separate NRC notification was provided.

Source of Commitment: Letter NLS880312 from G. A. Trevors (NPPD) to USNRC Document Control Desk dated July 8, 1988 - Reply to a Notice of Violation, NRC Inspection Report (IR) No. 50-298/88-05

Revision of Commitment: The response to IR 88-05 stated that the ignition systems on designated station vehicles would be modified with a warning system that will alert the driver if the keys are left in the ignition. Since 1988, vehicle changes and upgrades have resulted in ignition systems that cannot be modified with a key warning system without causing damage to the vehicle or invalidating the vehicle warranty. Accordingly, the commitment was changed to allow keys to be issued with an attached wrist or waist band device that attaches to the driver such that a warning or reminder is provided if there is an attempt to exit the vehicle without first removing the ignition key.

Source of Commitment: Letter from T. P. Gwynn (USNRC) to G. R. Horn (NPPD) dated March 17, 1995 - NRC Inspection Report (IR) 50-298/95-01 and Notice of Violation

Revision of Commitment: The cover letter for IR 95-01 documents a verbal commitment made by CNS management during the exit meeting. Specifically, a commitment was made to interpret Section 3.7.A.5.d of the Technical Specifications such that the plant would be placed in cold shutdown within 24 hours if instrument air is directed to the primary containment pneumatic header. During implementation of the Improved Technical Specifications (ITS) under Amendment 178, Section 3.7.A.5.d was superseded by ITS Section 3.6.3.1. Accordingly, the commitment was revised to state that oxygen content will be assumed to not be within limits if instrument air is directed to the primary containment pneumatic header. This being the case, Limiting Condition for Operation (LCO) 3.6.3.1 actions will be followed provided plant conditions meet those specified in the Applicability section of the LCO.

Source of Commitment: Letter NLS970215 from G. R. Horn (NPPD) to Director, Office of Enforcement dated December 31, 1997 - Reply to a Notice of Violation, NRC Inspection Report (IR) Nos. 50-298/97-07 and 97-12

Revision of Commitment: The combined response to IR 97-07 and IR 97-12 stated that Senior Management will implement the individual action plans as outlined in the performance improvement strategy (both for the Corrective Action Program and for Engineering). This commitment was further clarified in letter NLS980016 from P. D. Graham to USNRC Document Control Desk dated January 28, 1998 - Clarification of Commitments. Under this clarification, it was stated that back-end review of root cause evaluations is being conducted by the Condition Review Group (CRG). Under this clarification, it was further explained that the committed actions stated therein were intended to be short term, until sustained Corrective Action Program improvement was demonstrated. As a result of the implementation of formalized procedures for the back-end review of all root cause evaluations and improving trends in quality and acceptance rate, sustained Corrective Action Program improvement has been demonstrated. Consequently, the commitment for CRG to conduct back-end reviews of root cause evaluations is being discontinued.

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

Revision of Commitment: The SALP response stated that a procedures "betterment" project would be implemented in 1999 to improve the quality of procedures governing abnormal operating conditions and emergency operations. While the intent of the commitment as stated in the body of the letter was to implement (but not necessarily complete) the project in 1999, the commitment summary sheet attached to the letter reworded the commitment slightly to say that the project would be completed in 1999. To eliminate possible confusion, a commitment change was processed to state that an Abnormal and Emergency Procedure betterment project will be developed and implemented, and that the affected abnormal and emergency procedures would be revised by December 31, 2000 in accordance with the plan schedule.

Source of Commitment: CNS Licensee Event Report (LER) 1997-014 dated December 12, 1997 - Station Modification Creates Potential Inability to Mitigate Accident Consequences

Revision of Commitment: The response to LER 1997-014 stated that a review would be conducted of past modifications to risk significant systems to identify and evaluate station modifications with the potential to impact the ability of safety-related systems to perform their design basis functions. The previously committed completion date of June 30, 1999, was extended under this change to December 31, 1999. Experience gained through completion of approximately two-thirds of the planned scope showed that additional time was warranted in order to maintain the review quality and to satisfactorily resolve the resulting design questions.

Source of Commitment: CNS Licensee Event Report (LER) 1998-001 dated February 26, 1998 - Technical Specification Violation Due to Failure to Address Equipment Inoperability During Surveillance Testing

Revision of Commitment: The response to LER 1998-001 stated that the Standby Gas Treatment (SGT) surveillance procedures would be revised such that a) testing is performed when the SGT system is not required to be operable, OR b) testing is conducted without closing the cross-tie valve. Option a) of this commitment was changed to state that the SGT surveillance procedures would be revised to ensure that the SGT system is not aligned such that both trains are inoperable when the cross-tie valve is placed in the closed position. While Option a) as originally worded would preclude recurrence, it did not accurately reflect the intent of the corrective actions developed as part of the underlying root cause evaluation (i.e., to provide acceptable testing configurations when SGT is required to be operable). Therefore, since Option b) is unaffected, the change allows for conducting SGT surveillance testing with the cross-tie valve either closed or not closed, provided that both SGT trains are not rendered inoperable by the test.

Source of Commitment: CNS Licensee Event Report (LER) 1999-001 dated March 22, 1999 - Failure to Recognize Setpoint Shift Causes Common Mode Failure of High Pressure Coolant Injection Pressure Switches

Revision of Commitment: The response to LER 1999-001 stated that the applicable setpoint calculation would be reviewed to determine whether setpoint adjustments could be made to allow additional margin to the Technical Specification values. The response further stated that this calculation

would be revised to incorporate the vendor information on setpoint drift. This review, and the subsequent revision, were to be completed by July 25, 1999. While the review was completed and a revision to the calculation developed by the stated commitment date, additional time was required to facilitate the formal review and approval process. Accordingly, the commitment date was revised to August 24, 1999.

Source of Commitment: CNS Licensee Event Report (LER) 1999-005 dated July 1, 1999 - Failure to Adequately Perform Logic System Functional Testing Places Plant in a Condition Prohibited by Technical Specifications

Revision of Commitment: The response to LER 1999-005 stated that guidance would be developed on how to satisfy logic system functional test requirements for different circuit configurations and functions. This guidance was to be captured in a controlled document and incorporated into the modification and procedure change process by October 7, 1999. As a result of higher priority plant issues, including a forced plant outage, the completion date was revised to December 14, 1999.

Source of Commitment: CNS Licensee Event Report (LER) 1999-005 dated July 1, 1999 - Failure to Adequately Perform Logic System Functional Testing Places Plant in a Condition Prohibited by Technical Specifications

Revision of Commitment: The response to LER 1999-005 stated that Emergency Core Cooling Systems logic circuits that contain bypass and override control switches in parallel with automatic functions would be identified. Once identified, the circuits would be reviewed to confirm that logic system functional test requirements are being acceptably met. These actions were to be completed by October 7, 1999. As a result of higher priority plant issues, including a forced plant outage and several unscheduled system outages, the completion date was revised to February 25, 2000 and again to April 26, 2000.

Source of Commitment: CNS Licensee Event Report (LER) 1999-005 dated July 1, 1999 - Failure to Adequately Perform Logic System Functional Testing Places Plant in a Condition Prohibited by Technical Specifications

Revision of Commitment: The response to LER 1999-005 stated that a review of surveillance procedures would be performed to confirm that logic system functional test requirements are acceptably demonstrated for a selected set of Technical Specification functions. This action was to be completed by October 7, 1999. As a result of higher priority plant issues, including a forced plant outage and several unscheduled system outages, the completion date was revised to February 25, 2000 and again to May 23, 2000.

Source of Commitment: CNS Licensee Event Report (LER) 1999-007 dated October 18, 1999 - Sump Z Inoperability Results in Technical Specification Required Shutdown

Revision of Commitment: The response to LER 1999-007 stated that, prior to placing Augmented Off-Gas Train "B" in service, revisions would be made to the system operating procedure (Procedure 2.2.58.4) and the special procedure (SP 97-13) being used to return the "B" train to service following a previously implemented modification. The revisions were to incorporate the lessons learned from the subject event and were to be implemented by February 6, 2000. Due to the volume of changes required to SP 97-13, the commitment was revised to note that a new SP

(SP 00-002) would be developed. Also, as the SP would not be required until restart from Refueling Outage 19 (RFO-19), the implementation date was revised to March 28, 2000. Since the system operating procedure required operational input from the SP before being revised, a separate implementation date of May 16, 2000 was assigned to the procedure. As a result of delays to the restart from RFO-19 (due to problems unrelated to the subject LER), these implementation dates were subsequently revised to April 8, 2000 and August 15, 2000, respectively.

Source of Commitment: CNS Licensee Event Report (LER) 1999-007 dated October 18, 1999 - Sump Z Inoperability Results in Technical Specification Required Shutdown

Revision of Commitment: The response to LER 1999-007 stated that, prior to placing the Augmented Off-Gas (AOG) system in service, a formal briefing would be required in accordance with Conduct of Operations Procedure (COP) 2.0.1.1. It was further stated that this requirement would be added as a prerequisite action in the associated system operating procedures (Procedures 2.2.58.3 and 2.2.58.4) by December 6, 1999. As stated, the commitment was not technically accurate since COP 2.0.1.1 does not drive the actions necessary to conduct a formal briefing. Instead, COP 2.0.1.1 controls special procedures and special test procedures that meet the criteria in Significant Operating Experience Report (SOER) 91-01, "Conduct of Infrequently Performed Tests or Evolutions." All other procedures that meet SOER 91-01 criteria are controlled by the normal procedure change process. The requirements for a formal briefing are established by Procedure 2.0.1, Plant Operations Policy. Accordingly, the commitment was reworded to state that, prior to placing the AOG system in service, the Operations department will require a formal briefing consistent with the guidance contained in SOER 91-01, and that this requirement would be added as a prerequisite action in the associated system operating procedures (Procedures 2.2.58.3 and 2.2.58.4). As part of this commitment change, the implementation date was revised to February 4, 2000.

Source of Commitment: CNS Licensee Event Report (LER) 1999-007 dated October 18, 1999 - Sump Z Inoperability Results in Technical Specification Required Shutdown

Revision of Commitment: The response to LER 1999-007 stated that Conduct of Operations Procedure (COP) 2.0.1.1, Conduct of Infrequently Performed Tests or Evolutions, would be evaluated and revised, if necessary, to ensure that pre-evolution briefs meet management expectations. Further, a review was to be conducted to identify and document Operations department procedures and evolutions that warrant inclusion into COP 2.0.1.1. These actions were to be completed by December 6, 1999. As stated, the commitment was not technically accurate since COP 2.0.1.1 does not drive the actions necessary to conduct a formal briefing. Instead, COP 2.0.1.1 controls special procedures and special test procedures that meet the criteria in Significant Operating Experience Report (SOER) 91-01, "Conduct of Infrequently Performed Tests or Evolutions." All other procedures that meet SOER 91-01 criteria are controlled by the normal procedure change process. The requirements for a formal briefing are established by Procedure 2.0.1, Plant Operations Policy. Accordingly, the commitment was reworded to state that the briefing requirements associated with SOER 91-01 would be evaluated and revised, if necessary, to ensure that pre-evolution briefs meet management expectations. Additionally, the aforementioned review would identify and document Operations department procedures and evolutions that warrant inclusion into the subset of procedures identified with SOER 91-01. As part of this commitment change, the implementation date was revised to February 4, 2000.

Source of Commitment: CNS Licensee Event Report (LER) 1999-008 dated November 17, 1999 - Troubleshooting Activities Causes Critical Bus Undervoltage and Engineered Safety Feature (ESF) Actuations

Revision of Commitment: The response to LER 1999-008 stated that CNS would perform a "needs analysis" for training electricians on the use of portable recorders, and implement the required training by January 6, 2000. The commitment was revised to state that CNS will incorporate training on portable recorders into initial electrical training for new employees and into continuing training for incumbent electricians by July 31, 2000. This change was to allow for development of the training as well as the scheduling of initial and continuing training classes.

Source of Commitment: CNS Licensee Event Report (LER) 1999-008 dated November 17, 1999 - Troubleshooting Activities Causes Critical Bus Undervoltage and Engineered Safety Feature (ESF) Actuations

Revision of Commitment: The response to LER 1999-008 stated that CNS would provide training that emphasizes that the boundary statement and the plant impact statement must be accurate and that troubleshooting steps confine work to within the stated boundaries for individuals that prepare and perform troubleshooting instructions by January 6, 2000. The commitment implementation date was revised to March 30, 2001 to allow for sufficient time to schedule and deliver training to Engineering personnel that prepare and perform troubleshooting instructions.

Source of Commitment: CNS Licensee Event Report (LER) 2000-004 dated February 28, 2000 - High System Flow During Restoration from Maintenance Causes Primary Containment Group 3 Isolation

Revision of Commitment: The response to LER 2000-004 stated that the appropriate station procedure(s) would be revised to: 1) define and state the applicability of the fill and vent section, 2) inform the reader of the inability to fill and vent the Reactor Water Cleanup (RWCU) system pump suction piping and add a requirement for Operations to evaluate the necessity of draining the RWCU suction piping with the plant on line, 3) add a caution statement prior to performing applicable steps stating that RWCU-MOV-MO18 must be slowly throttled open to preclude the possibility of a Group 3 isolation, and 4) add a step requiring the conduct and logging of a formal brief of the pressurization evolution including the potential of a Group 3 isolation and required actions. As stated in the LER, this action was to be completed by June 6, 2000. However, due to an extended refueling/forced outage, plant resources were redirected and the due date for this commitment was revised to August 3, 2000.

Source of Commitment: Letter from J. M. Pilant (NPPD) to E. M. Howard (NRC) dated October 6, 1976 - IE Circular No. 76-02, Relay Failures - Westinghouse BF and BFD

Revision of Commitment: The response to IE Circular No. 76-02 stated that BFD relay testing would be conducted once per operating cycle to assure that testing is conducted at least every two years. The BFD relays are normally de-energized and are used for alarm only. Since they have no failure history and the ground detection system is redundant, the commitment was changed such that the testing interval will be once every three years or as needed to ensure the relays operate properly when energized.

Source of Commitment: Letter from J. M Pilant (NPPD) to T. A. Ippolito (NRC/NRR) dated April 16, 1979 - Reload 4, Cycle 5 - Startup Physics Testing

Revision of Commitment: The subject letter described the startup physics testing to be performed at CNS following Reload 4 and subsequent refuelings. The startup physics test program components included: Core Verification, Control Rod Operability and Subcriticality Check, and the Traversing In-core Probe (TIP) Signal Uncertainty Test. Because of the decreased uncertainty associated with the gamma TIP system currently in service, and because of the implementation of 3D Monicore for core monitoring, the TIP Signal Uncertainty Test is no longer required and has been deleted from the startup physics test program.