OPPD
Omaha Public Power District
444 South 16th Street Mall
Omaha NE 68102-2247
May 23, 1997
LIC-97-0070

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docket No. 50-285

SUBJECT: Safety Analysis Report Update and 10 CFR 50.59 Report for Fort Calhoun Station

As required by 10 CFR 50.59(b)(2), Attachment A is provided as Omaha Public Power District's (OPPD) report of changes, tests and experiments performed pursuant to 10 CFR 50.59 for the Fort Calhoun Station. Attachment B is provided to describe revisions to the Updated Safety Analysis Report (USAR) other than those resulting from 10 CFR 50.59 changes (i.e., revisions resulting from 10 CFR 50.54, 10 CFR 50.90, and administrative changes). This information is for the period of May 1, 1995 through December 31, 1996.

In this submittal, Omaha Public Power District is reissuing the entire Fort Calhoun Station USAR. All previous USAR copies should be discarded. Pursuant to 10 CFR 50.71(e) and 10 CFR 50.4(b)(6), Attachment C provides one original set and 10 copies of the Fort Calhoun Station USAR. The original set is designated as Copy Number 1 and the 10 copies as Copy Numbers 2 through 11.

If you should have any questions, please contact me.

Sincerely,

W. G. Gates

W. G. Gates
Vice President

Attachments

WGG/mle

c: Winston & Strawn (w/o Attachments B & C)
E. W. Merschoff, NRC Regional Administrator, Region IV (Copy #13)
L. R. Wharton, NRC Project Manager (Copy #12)
W. C. Walker, NRC Senior Resident Inspector (Copy #15)

A053 1/12

UNITED STA~ , OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Omaha Public Power District	)	Docket No. 50-285
(Fort Calhoun Station	)	
Unit No. 1)	)	

AFFIDAVIT

W. G. Gates, being duly sworn, hereby deposes and says that he is the Vice President in charge of all nuclear activities of the Omaha Public Power District; that as such he is duly authorized to sign and file with the Nuclear Regulatory Commission the attached information concerning the Safety Analysis Report Update and 10 CFR 50.59 Report for Fort Calhoun Station; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information, and belief.

W. G. Gates

W. G. Gates
Vice President

STATE OF NEBRASKA)
) ss
COUNTY OF DOUGLAS)

Subscribed and sworn to before me, a Notary Public in and for the State of Nebraska on this 23Rd day of May, 1997.

Theresa Peterson
Notary Public

GENERAL NOTARY-State of Nebraska
THERESA PETERSEN
My Comm. Exp. Feb. 17, 2001

Attachment A
LIC-97-0070

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
MR-FC-94-005	Description: Modification Request (MR) FC-94-005 removed supervisory relays from the lockout relay circuits in the engineered safety features, diverse scram system, reactor coolant pumps, and certain electrical protection circuits. The supervisory relays were removed to eliminate the possibility of a transient to the plant caused by the actuation of a lockout relay due to the failure of a supervisory relay (i.e. short in its coil). Safety Analysis: The supervisory relays perform only an indication/alarm function and have no safety function. Removing the supervisory relays reduces the possibility of a transient or trip caused by the inadvertent actuation of a lockout relay due to the failure of a supervisory relay. The removal of the supervisory relays does not affect the function or capability of the lockouts associated with them. The indication/alarms provided by the supervisory relays are not used to mitigate any accident scenario. Therefore, neither the probability of a previously evaluated accident is increased nor are the consequences of such an accident increased. The modification does not increase the probability of a safety equipment malfunction nor the consequences of such a malfunction. The modification does not create the possibility of a new type of accident nor the possibility of a new type of safety equipment malfunction.	Sections 7.1 & 7.3.1, Figure 7.3-3
MR-FC-92-020	Description: This modification added three (3) new fire protection systems to the turbine building. These are (1) a pre-action system for the turbine/generator bearings, (2) a total flooding CO_2 system for the exciter, and (3) a wet pipe system for the turbine building mezzanine level for coverage of the lube oil hazards in that area. The systems were added in order to minimize potential financial losses associated with turbine/generator fires. Safety Analysis: These systems are not credited for mitigation of any of the accidents evaluated in the USAR. Therefore, neither the probability of a previously evaluated accident is increased nor are the consequences of such an accident increased. These systems do not provide service to or interface with any equipment credited for accident mitigation; therefore, the modification does not increase the probability of a safety equipment malfunction nor the consequences of such a malfunction. Since these systems do not provide service to or interface with any equipment credited for accident mitigation, the modification does not create the possibility of a new type of accident nor the possibility of a new type of safety equipment malfunction.	Section 9.11

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
MR-FC-92-044	Description: The logic configuration for the steam generator isolation signal (SGIS) was changed. Relays 86X-A-B1/CPHS and 86-B-A1/CPHS for SGIS channels A and B respectively were eliminated. Two new relays were added to each channel to support a new logic configuration. The modified circuit requires that two (2) SGIS relays go to their deenergized position before a normally open isolation valve will reposition. Each SGIS relay was individually fused. The previous logic configuration for SGIS would allow the inadvertent isolation of a steam generator if a SGIS relay deenergized due to a coil fault. An additional relay was added to each SGIS channel and wired so that two (2) relays must deenergize to reposition an isolation valve. This will prevent a relay failure from isolating flow to or from a steam generator. If a relay should fail to its deenergized position the channel will still respond properly to an accident signal. Individually fusing each relay will cause the same circuit response for an open or shorted relay coil. Safety Analysis: This modification changed the logic configuration within the SGIS circuit. SGIS isolates the steam generators in response to a main steam line break or a loss of coolant accident. The circuit will continue to produce the same response to the same input signals. This modification decreases the possibility of a loss of feedwater due to a SGIS component failure and does not increase the consequences of any previously evaluated accident. The modification reduces the possibility that a single component failure within the SGIS circuit logic would cause inadvertent feedwater isolation to or steam flow from the steam generators. The effect of a malfunction of a component within the SGIS logic circuit is reduced by this modification. Thus, the modification does not increase the probability of a safety equipment malfunction nor the consequences of such a malfunction. This modification only made changes in the SGIS logic. The SGIS logic performs the same function and operates with the same input signals. No other systems were affected. Therefore, the modification does not create the possibility of a new type of accident nor the possibility of a new type of safety equipment malfunction.	Sections 7.3.6 & 7.6.4
MR-FC-94-018	Description: A connection to the raw water header in Room 81 was made to allow a hose to be connected to it to supply the emergency feedwater storage tank (EFWST) with makeup water during an event such as Fire Safe Shutdown or Seismic Safe Shutdown where the reactor could be at hot standby or hot shutdown for up to 72 hours. The preferred tank makeup will remain demineralized water, then condensate and finally raw water (RW). A dedicated hose on a rack will be connected between the two tapped locations when needed under AOP-30 for EFWST makeup. Safety Analysis: The installed configuration of the new valves and their associated components meet Code requirements for stresses, materials and surveillance. The heat removal capability of the RW system during the first few hours of an accident is not affected. The RW tie in will be used after significant heat loads have subsided and the minor loss of RW flow is insignificant. Therefore, neither the probability of a previously evaluated accident is increased nor are the consequences of such an accident increased. The valve being added is manually operated locally and is controlled by AOP-30. It is normally capped off. It interfaces with no other components or systems until it is connected to the EFWST as directed by a plant procedure. A failure of the hose connected to the EFWST while connected to the RW system is possible. However, a check valve protects AFW inventory. The loss of 100 to 200 GPM of RW is of no consequence after the first few hours of an event and is bounded by existing USAR Appendix M high energy line break (HELB) and flooding analyses. Thus, the modification does not increase the probability of a safety equipment malfunction nor the consequences of such a malfunction. The loss of a significant amount of RW is not credible due to the valve size and the driving head available and flooding is bounded by previous analysis as stated above. The heat removal capability of the RW system at the early hours of an event when the greatest heat load occurs is not affected since the RW connection will not be in use at that time. Therefore, the modification does not create the possibility of a new type of accident nor the possibility of a new type of safety equipment malfunction.	Sections 2.5, 9.4, 9.8, Table N-1

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S), OR FIGURE(S) REVISED
MR-FC-81-183	Description: This modification added a system to monitor vibrations on the four reactor coolant motors/pumps RC-3A through RC-3D. The system replaced the existing source of the "Vibration High" annunciators (CB-1/2/3, Box A-6, Windows A-4, B-4, C-4 and D-4). The system also replaced the existing source of the 0% and 90% reactor coolant pump speed signals. The system is located in a cabinet in Room 57 and has one channel for each motor/pump combination. The system is connected to a dedicated personal computer capable of providing detailed analysis of the vibration data. This system was added to provide early detection of cracking/failure of the reactor coolant motor/pump shafts and to upgrade the 0% and 90% speed signal sources. Safety Analysis: This modification replaced the existing vibration detector with an enhanced vibration monitoring system which generates the same control signals as the previous detector. The vibration detector (monitoring system) is not required to prevent or mitigate an accident. Therefore, neither the probability of a previously evaluated accident is increased nor are the consequences of such an accident increased. Vibration monitoring and alarms are still provided. The vibration monitor has no effect on any equipment important to safety. Therefore, this modification does not increase the probability of a safety equipment malfunction nor the consequences of such a malfunction. The new vibration monitoring system does not generate, transmit, receive, process or modify any safe shutdown control signals. Therefore, failure of the vibration monitoring system has no effect on safe shutdown components. The modification does not create the possibility of a new type of accident nor the possibility of a new type of safety equipment malfunction.	Table 4.3-5
MR-FC-87-044	Description: The vibration and loose parts monitoring system (VLPMS) was upgraded. The new system is equipped with a digital data acquisition and analysis subsystem and has additional vibration and loose parts sensors. The modification was done to upgrade the VLPMS and ensure its reliability. The VLPMS is used to monitor the thermal shield, core barrel motion and steam generator internals. Safety Analysis: The VLPMS is a passive system, which is not credited in any accident evaluated in the USAR. Therefore, neither the probability of a previously evaluated accident is increased nor are the consequences of such an accident increased by this modification. The new VLPMS equipment interfaces with existing plant equipment in the same manner as the old equipment. Therefore, this modification does not increase the probability of a safety equipment malfunction nor the consequences of such a malfunction. The modification does not create the possibility of a new type of accident nor the possibility of a new type of safety equipment malfunction.	Figure 7.6-1

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
MR-FC-94-021	Description: The actuation circuitry for Raw Water Isolation Valves HCV-2880A/B, HCV-2881A/B, HCV-2882A/B and HCV-2883A/B was reconfigured to provide actuation (opening) upon initiation of a safety injection actuation signal (SIAS). This provides the ability to operate with less than three (3) RW/component cooling water (CCW) heat exchangers in service and still provide assurance that no less than three (3) heat exchangers will have RW flow during post-accident conditions. This provides better control of CCW temperature and increases RW flow which helps reduce sanding of the RW side of the heat exchangers. Safety Analysis: The probability of a LOCA or MSLB is not increased because the modification doesn't adversely affect any components associated with the pressure boundaries. System/component failure modes are not affected. Therefore, neither the probability of a previously evaluated accident is increased nor are the consequences of such an accident increased by this modification. The increased cooling of components served by the CCW system is not detrimental to any component. Leakage due to failure of the reactor coolant pump seals is bounded by the existing LOCA analyses and the consequences of this failure are not increased. Therefore, this modification does not increase the probability of a safety equipment malfunction nor the consequences of such a malfunction. This modification does not affect the design conditions or design limits for any equipment. Control of the affected valves is available during all modes of operation and the modified configuration is essentially unchanged in that all system safety functions are maintained. No new equipment failure modes are created by this modification since equipment independence and single failure criteria are maintained. Thus, the modification does not create the possibility of a new type of accident nor the possibility of a new type of safety equipment malfunction.	9.8
MR-FC-92-040	Description: This modification replaced the diesel generator starting air compressors with booster compressors. The new booster compressors draw dry air from the instrument air (IA) system allowing the air dryers to be removed. Safety Analysis: This modification did not affect the availability of the starting air receivers or the starting air (SA) system. The impact on the IA and electrical system is negligible because the booster requirement is intermittent and only 10% of the capacity of the main compresser which is only loaded to 70%. The compressors and dryers are not credited in any USAR analysis. Thus, neither the probability of a previously evaluated accident is increased nor are the consequences of such an accident increased by this modification. This modification increased the reliability of the SA receivers by assuring that moisture in the tanks is held to a minimum. This also increases the reliability of the air start motors since system corrosion is minimized. The modification maintains the redundancy and separation of the primary and secondary SA systems. Thus, this modification does not increase the probability of a safety equipment malfunction nor the consequences of such a malfunction. Since only the SA system is impacted by this modification and since the SA system is not credited in any USAR accident analysis, the modification does not create the possibility of a new type of accident nor the possibility of a new type of safety equipment malfunction.	8.4

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S). SECTION(S). TABLE(S) OR FIGURE(S) REVISED
MR-FC-95-013	Description: This modification removed the existing orifice plates attached to each steam generator tube sheet (hot leg plenum side) and installed new replaceable orifice plates. This was done to permit eddy current testing (with the latest probe technology) of those tubes located under the existing orifice plates. The new orifice plates were installed to maintain an equivalent reactor coolant system (RCS) flow rate and eliminate the potential for fuel related failures resulting from increased vibration (flow induced) of the reactor internals. Safety Analysis: The replacement orifice plates are a completely passive device designed to restrict RCS flow to approximately the same rate as the previous orifice plates. The plates are designed to withstand all structural, thermal and seismic loading as specified by the original design specification for the FCS steam generators. Although the replacement orifice plates are not designed to withstand all accident conditions, they will not increase the consequences of an accident. A loss of coolant accident (LOCA) - hot leg break has the potential (due to reverse RCS flow) to cause the plate fasteners to fail and dislodge the plate. The potential for the orifice plates to travel down the hot leg in the direction of the break location was investigated. It is not considered credible that any of the orifice plates could enter the reactor vessel since positive flow out of the core through the hot leg break is always assumed. The impact of the replacement orifice plates on the main steam line break (MSLB) and LOCA - cold leg accidents were also reviewed. The structural analysis bounds the stresses that would occur in the replacement plates during a MSLB. Westinghouse performed a LOCA safety evaluation to address the impact of up to 20% effective steam generator tube plugging resulting from the presence of the orifice plates. Results showed that the 20% tube plugging margin is acceptable for the affected LOCA analysis. Thus, the replacement orifice plates do not increase the probability of occurrence nor the consequences of any previously evaluated accidents. The replacement orifice plates are located upstream of the reactor coolant pumps on the hotleg side of the steam generator and are separated from the RC pumps by the steam generator tube sheet. The probability of occurrence of a safety equipment malfunction is not increased since the plates are passive components. In the event the mounting fasteners fail and the plate becomes a loose part, its location on the hot leg side of the tube sheet would prevent it from being carried into the reactor vessel. the plates will not adversely affect any equipment important to safety. Since the replacement orifice plates will remain in place (or in the case of a hot leg break LOCA, the effects are bounded by the cold leg break LOCA),and affect only RCS flow, no other systems or components other than those previously evaluated are affected. Hence, an accident of a different type than previously evaluated isn't credible nor is the possibility of a malfunction of equipment important to safety. The replacement orifice plates maintain RCS flow at or above previous levels meeting the minimum flow requirements of Technical Specification 2.7, thus the replacement orifice plates do not reduce the margin of safety as defined in the Technical Specifications.	Sections 4.3, 14.9, 14.10, 14.15

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
Temporary Modification (TM) 95-014 and TM-95-011	Description: The interlock on the FH-12 trolley was disabled to allow it to travel west in the transfer canal past the upender zone to allow the removal of a spent fuel bundle from the spent fuel pool and bring it to the new fuel elevator for reconstitution. The upper limit on FH-14 was lowered to prevent fuel assembly in the new fuel elevator from coming near to water surface (to maintain adequate shielding). Safety Analysis: Disabling the interlock has no effect on the lifting capability of FH-12 to transport a spent fuel assembly to the new fuel elevator. Administrative controls ensured that all other interlocks and engineered barriers to ensure safe handling were functional. Disabling the interlock did not increase the consequences of a dropped fuel assembly. The disabling of the interlock had no impact on the capability of FH-12 to safely move fuel assmblies within the spent fuel pool area and transfer canal. Therefore, the change did not increase the probability or consequences of a malfunction of equipment important to safety. Since all other interlocks and engineered safety barriers were functional, the margin of safety was not reduced.	None
MR-FC-89-057	Description: This modification replaced the existing meteorological monitoring system with a microprocessor based data acquisition system that performs all the signal processing and analog to digital conversions in a self contained unit located at the weather tower shelter. The unit is linked to the emergency response facility via fiber optic line and the ERF is programmed to display the parameters in a manner similar to the original display. This modification was necessary because the pervious data collection and recovery system was obsolete and spare parts inventory could not be maintained. Safety Analysis: The meteorological monitoring system is not related to safety. The only interface to the plant is through the ERF computer which is non-safety related. This design change upgrades the equipment while maintaining the requirements for meteorological monitoring system at FCS. Therefore, neither the probability of a previously evaluated accident is increased nor are the consequences of such an accident increased by this modification. The new equipment interfaces with existing plant equipment in a similar manner as the old equipment. Therefore, this modification does not increase the probability of a safety equipment malfunction nor the consequences of such a malfunction. The modification does not create the possibility of a new type of accident nor the possibility of a new type of safety equipment malfunction.	Section 2.5.3.2 Table 2.5-21 Figure 7.6-1

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
SAO-96-02 TM-96-042 Amendment 172	Description: SAO-96-02 documents an evaluation of the operability of the CCW system and containment air cooling coils for a loss of offsite power (LOOP) coincident with a LOCA or MSLB. SAO-96-02 documents that a LOOP coincident with these accidents will not cause a consequential failure of the containment air coolers or CCW system. SAO 96-02's compensatory action addressing this is a requirement to maintain a minimum nitrogen overpressure in the CCW surge tank. TM 96-042 increased the CCW surge tank relief valve setpoints to accommodate a higher operating pressure in the tank in accordance with SAO-96-02. New relief valve setpoints are still below the surge tank's nameplate design pressure. Safety Analysis: The provisions of SAO-96-02 do not increase the probability of occurrence of an accident previously evaluated in the USAR. Maintaining the r...red pressure in the CCW surge tank ensures that vaporization of CCW in the containment air cooling coils does not occur for a LOO. coincident with a LOCA or MSLB. The CCW system will therefore operate as intended for those accidents and no containment integrity concern is created. The radiological consequences of a LOCA or MSLB are therefore still bounded by the dose assessment currently in the USAR for those events. Thus, the consequences of an accident previously evaluated in the USAR are not increased. The pressure boundary integrity of the CCW system is not threatened by the higher CCW surge tank operating pressure nor does the CCW piping become a high energy line as a result of the pressure increase. The SAO does not increase the probability of a safety equipment malfunction nor the consequences of such a malfunction. The modification does not create the possibility of a new type of accident nor the possibility of a new type of safety equipment malfunction. Technical Specification 2.4 pertains to the plant's capability to limit the containment pressure below the design value (60 psig) following a design basis accident (DBA). SAO-96-02 documents that the containment pressure will not exceed 60 psig for the accident scenarios of consequence (i.e., LOOP coincident with LOCA or MSLB). Since the capability to limit the post-DBA containment pressure is not compromised, the Technical Specification margin of safety is not reduced. TM-96-042 did not affect the ability of the CCW system to fulfill its nuclear safety function. Resulting CCW system operating pressures do not create the potential for a loss of pressure boundary integrity. No functional changes were made to the CCW system and installation and testing of the TM were done while the plant was in cold shutdown. Therefore, this TM did not involve an Unreviewed Safety Question.	Sections 9.7 and 9.8

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
MR-FC-92-039	Description: Remote control features such as solenoids, limit switches, light indications, and alarms associated with 32 RW/CCW interface valves were removed or abandoned in place. A local manual instrument air (IA) control valve was installed on each valve actuator for process valve control. Three-way control switches that operate both RW and CCW interface valves were replaced by a two-way switch for CCW valve control only. The auto open and close signal to control room (CR) air cooler RW interface valves was removed and air accumulators for HCV-2898 C/D and HCV-2899C/D were abandoned. The RW/CCW interface valves previously required both local manual and remote manual actions to open them. The remote manual operational features were removed to eliminate various preventive maintenance tasks and costs. Only local manual actions are now necessary to open these valves. A detailed analysis (EAR-91-037) concluded that remote actuation of the RW/CCW interface valves is not required. Safety Analysis: The configuration changes to the RW/CCW valves did not alter RW backup capability on a loss of CCW to the components they serve. This modification did not alter RW/CCW system pressure retaining components, containment isolation components, or radiological release components. Therefore, neither the probability of a previously evaluated accident is increased nor are the consequences of such an accident increased by this modification. The performance of safety related components cooled by RW/CCW was not changed since this modification did not alter RW/CCW direct cooling capability or flow. The manually operated IA control valves perform the same function as the solenoids which they replaced and are highly reliable due to the simplicity of their design. Therefore, this modification does not increase the probability of a safety equipment malfunction nor the consequences of such a malfunction. This modification did not introduce any new failure modes or components likely to initiate an accident. The capability and capacity of RW direct cooling has not changed. Thus, the modification does not create the possibility of a new type of accident nor the possibility of a new type of safety equipment malfunction.	Section 5.9, Figure 5.9-13 (numerous sheets)
MR-FC-94-020	Description: The CCW solenoid valves which isolate the control room (CR) HVAC economizers were modified to fail closed upon VIAS with reopening capability under administrative control. This will prevent the creation of an additional heat load in the CR in the event that a temperature control valve serving a water cooled economizer in a running VA unit fails during a MSLB or LOCA. Safety Analysis: The changes conform to design, material, construction and testing standards for this equipment. The modification does not cause the equipment to be operated beyond design limits or create new operating conditions. The change has no direct affect on the radiological consequences of an accident. The CR HVAC system will function to maintain personnel and electronic equipment cool to ensure the correct performance of devices which mitigate accidents. Thus, neither the probability of a previously evaluated accident is increased nor are the consequences of such an accident increased by this modification. The modification maintains seismic qualifications, ensures separation criteria are not violated and maintains environmental qualification of components necessary to ensure correct functioning of the CR air conditioning units. Redundant devices and manual action will mitigate any failure of the modified system. Thus, the modification does not increase the probability of a safety equipment malfunction nor the consequences of such a malfunction. No credible accident of a different type than previously evaluated in the USAR is created by this modification. Applicable failure modes were evaluated and the modification does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the USAR. No margin of safety as described in the TS is reduced. Operability of CCW components and the CR air conditioning units is enhanced by the control and logic changes.	Sections 7.3, 9.7, 9.8, 9.10 and Table 9.12-1

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
MR-FC-95-026	Description: An above ground, vault type, fuel storage tank was installed in the northeast portion of the protected area east of the warehouse to replace two fuel storage tanks determined not to meet fire code requirements. The tank has two separate 500 gallon compartments which hold gasoline and diesel fuel. A fire extinguisher and free standing storage box are installed near the tank. The tank is equipped with electric pumps for fuel dispensing. Safety Analysis: The applicable accident is a toxic gas release which could threaten control room habitability. Plant operation is not affected by any phase of this modification (the tank is outside the fire areas evaluated in the Updated Fire Hazards Analysis). The tank is a double-wall design with concrete fill between the inner and outer tanks. It is located in a yard area far from the fresh air intake of the control room. In the event of a fuel leak, the fuel capacity of the new tank is insufficient to create a threat to control room habitability. Thus, the likelihood of a toxic gas release which could threaten habitability of the control room is not increased by this modification. The toxic gas consequences of a fuel release from the new tank was evaluated and it was determined that the vapor concentration is well below established toxicity limits. Therefore, control room habitability is not threatened by a fuel tank accident. Since the incapacitation of the operators or forced evacuation of the control room is not a concern, the consequences of an accident previously evaluated in the USAR is not increased. The fuel tank does not directly or indirectly interface with any safety related equipment and is located in a yard area which is not in close proximity to any safety related equipment. Thus, the modification does not increase the probability of a safety equipment malfunction nor the consequences of such a malfunction. A gasoline or diesel fuel release is the only accident which is associated with the new fuel tank. The toxic vapor release from a gasoline spill is already listed in Table 14.23-3. Diesel vapors are classified as an asphyxiant rather than a toxic gas source. Calculations show that a fuel release from the new tank does not threaten control room habitability. Thus, the modification does not create the possibility of a new type of accident nor the possibility of a new type of safety equipment malfunction. Since the tank does not interface with any safety-related equipment, the modification does not impact any Technical Specification margin of safety either.	Section 14.23

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
Procedure Change (PC) 48546	Description: This procedure change eliminated the CEA coupling check test in Procedure RE-CPT-RX-0001, "Post Refueling Core Physics Testing and Power Ascension." This determination already exists in Procedure MM-RR-CEDM-1201, "CEDM Coupling" via measurements of the coupled and uncoupled weight of the letdown tool. Safety Analysis: Elimination of the CEA coupling check test in RE-CPT-RX-0001 has no effect on the probability of an accident occurrence since CEA coupling is verified prior to entering RE-CPT-RX-0001. The coupling check performed in MM-RR-CEDM-1201, as well as other tests performed as part of RE-CPT-RX-0001 will identify an uncoupled rod prior to beginning power ascension. A flux symmetry check at 25% power using CECOR snapshots will identify an uncoupled CEA via excessive flux peaking across the core from the depressed flux area which is the location of the uncoupled CEA. Thus, neither the probability of a previously evaluated accident is increased nor are the consequences of such an accident increased by this PC. Ensuring that the CEA coupling check is performed prior to startup has no effect on the malfunction of any safety related equipment as the methods/procedures of operation remain unchanged. Thus, the modification does not increase the probability of a safety equipment malfunction nor the consequences of such a malfunction. Elimination of the coupling check test from RE-CPT-RX-0001 does not change any design to the facility, which ensures that an accident of a different type cannot be created. Thus, the modification does not create the possibility of a new type of accident nor the possibility of a new type of safety equipment malfunction. The margin of safety as defined in the TS will be maintained since the coupling checks will continue to be performed despite the elimination of the coupling check test from RE-CPT-RX-0001.	Section 3.4.5.1
PC 48543	Description: The Core Operating Limits Report (COLR) was revised to provide updated figures and information specifically pertaining to operation of the Cycle 17 core. Safety Analysis: The COLR provides cycle specific limits on operation based upon results from the reload analyses performed in accordance with NRC approved reload methodology. Since the Cycle 17 COLR update does not involve changing any previously evaluated accident in the USAR, there is no increase in the probability of an accident, nor is there an increase in the consequences of an accident. The Cycle 17 COLR update does not alter equipment important to safety. Thus, there is no increase in the probability of occurrence of a malfunction of equipment important to safety nor the consequences of a malfunction of equipment important to safety. The Cycle 17 COLR update does not involve an accident condition not previously evaluated in the USAR. Thus, the Cycle 17 COLR update does not create the possibility of a new type of accident nor the possibility of a new type of safety equipment malfunction. The Cycle 17 COLR update was prepared using NRC approved reload methodology and thus there is no reduction in the margin of safety as defined in the TS.	Section 7.2.3

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
PC 44780	Description: Procedure Change 44780 (Maintenance Work Document 960446) allowed leak testing of the waste gas vent header using P-10 Gas (90% Argon-10% Methane). MSA combustible gas detectors were used to locate leaks. Safety Analysis: Although Methane, which is a flammable gas, is part of the tracer gas used, it is considered non-flammable in a 10% concentration. Argon is a noble gas and as such, inert. Methane has no known adverse interactions with any materials of construction and is non-corrosive. The use of P-10 gas did not increase the probability of a waste gas disposal tank rupture, which is the only applicable accident. The test was performed at approximately 1.5 psig and the waste header is rated at 150 psig. The change did not increase the consequences of any previously analyzed accidents as the containment isolation valves for the header were shut during the procedure. Since there are no known adverse interactions with the materials, the change did not increase the probability or consequences of a malfunction of equipment important to safety. Maintaining allowable concentrations of oxygen and hydrogen below flammable limits in the waste gas vent header was the applicable margin of safety. Since P-10 does not contain these gases, and the limits were maintained, no margin of safety was reduced as a result of this change.	11.1
PC 47782	Description: River level/temperature limits were incorporated into the Technical Data Book to provide limitations applicable to certain off-normal raw water system alignments to ensure acceptable conditions exist in the RW and CCW systems. Safety Analysis: Analyses completed indicated that river level and temperature limits need to be observed to ensure acceptable conditions exist in the RW and CCW systems if a high heat load accident occurs while in some of the RW system alignments allowed by TS. The river condition limits incorporated are operational constraints only, and therefore did not increase the probability of any accident previously evaluated. The limits ensure acceptable conditions exist in the RW and CCW systems if a high heat load accident (large LOCA or MSLB inside containment) occurs while the RW system is in an off-normal alignment. This ensures that the consequences of an accident previously evaluated will not be increased. Observing the limits also ensures that the probability and consequences of a malfunction of equipment important to safety is not increased. The river limits incorporated are more restrictive than presently allowed by TS 2.16. Issuance of the limits therefore did not reduce the margin of safety as defined in the basis for TS 2.3, 2.4, or 2.16.	9.8

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
MR-FC-95-022	Description: The General Electric (GE) motor on Reactor Coolant Pump (RCP) RC-3B was replaced with a motor manufactured by ABB Industries. Testing indicated that the original GE RC-3B motor had the greatest potential for winding insulation failure which would have caused significant plant downtime if the motor had to be replaced. Safety Analysis: The new motor does not affect overall RCS or RCP performance since it is equivalent to the previously installed GE motor. Motor design, material, fabrication, and construction standards insure that the new motor has equivalent or better reliability than the old motor. The new motor requires less total oil, both the upper and lower reservoirs combined, than the existing motor. The new motor does not affect the Fire Hazards Analysis or fire detection/suppression systems. The possibility of a flywheel failure that would generate missiles in containment is not a credible event for this motor and therefore, a shroud around the flywheel is not included in the motor design. The loss of power and seized rotor events are not impacted by this modification. On a loss of power, the new motor flywheel acts like the old motor flywheel to reduce the rate of flow decay as credited in the USAR. The new motor has mechanical and electrical controls that are equivalent to the old motor. Thus, the probability of occurrence of an accident previously evaluated in the USAR is not increased nor are the consequences of such an accident. The new motor has no adverse effects on the RCP or RCS piping and supports. Both the old and the new motors weigh approximately the same. The new motor is designed to maintain structural integrity during a safe shutdown earthquake. The new motor does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR nor does it increase the consequences of such a malfunction. Since the new motor does not change the operational characteristics of the RCS or RCP, no new accident scenarios or safety equipment malfunctions are created. No TS safety margins are reduced since RCS/RCP performance, operations, or Limiting Conditions for Operation are affected by the new motor.	Section 4.3, Figures 4.3-12, 4.3-13, 4.3-14, 4.3-6A, 4.3-6B
MR-FC-94-004	Description: Four (4) control room wide range neutron flux monitors were replaced because electrical parts for them are no longer available. The assemblies which support monitor operation and their control room indicators were modified to support replacement of the monitors. The "D" channel electronic assemblies were reconfigured to allow the elimination of a one of a kind isolation circuit. Safety Analysis: The replacement monitors were designed to meet the operating specifications of the previous monitors. The replacement equipment duplicates the function, response time, and accuracy of the previous equipment. Therefore, the new monitors do not increase the probability of an accident previously evaluated in the USAR, nor do the new monitors increase the consequences of such an accident. All of the replacement equipment was designed and configured in accordance with USAR requirements. The replacement monitors provide the same safety related output, meet the same seismic and separation criteria as the existing monitors. Thus, these monitors do not increase the probability of occurrence of a malfunction of equipment important to safety nor the consequences of such a malfunction. The replacement monitors do not create the possibility of an accident of a different type than any previously evaluated in the USAR nor do the replacement monitors create the possibility of a malfunction of equipment of a different type than any previously evaluated in the USAR. The margin of safety in the TS is not affected by this modification.	7.2, 7.5

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
USAR Clarification	Description: USAR Section 9.4 was revised to delete discussion on the 8 hour capacity of the day tank of Diesel Engine FW-56 and clarify the operation of its fuel transfer pump. Safety Analysis: No credit is taken for operation of FW-56/FW-54 in any accident previously evaluated. The failure of FW-56/FW-54 therefore is not an initiator of any accident previously evaluated and reducing the amount of fuel in the day tank to FW-56 did not increase the consequences of any accident previously evaluated. Since no credit is taken for FW-56/FW-54, the malfunction of the fuel oil transfer pump has no effect on previous evaluations. EA-FC-92-047 documents the fuel consumption of the DG LOCA loads to ensure IEEE-308 requirements to maintain a 7 day supply of on-site fuel are met. Therefore, no margin of safety of the TS basis was reduced.	9.4
ECN-93-394	Description: This ECN permanently removed the seismic peak acceleration recorders. Safety Analysis: The recorders are not a licensing commitment and are not justified for safety or technical reasons. The input was supplied by the peak acceleration at one steam generator and one reactor coolant pump. This is of little value since ground spectra for the event would not be available to estimate system response. The strong motion triaxial accelerographs are a licensing commitment and are still available. The removal does not increase the probability or consequences of a previously evaluated accident or create the possibility of a different type of accident since the equipment only records peak acceleration data following a seismic event. The strong motion accelerographs still will record the data necessary to determine if plant shutdown is required. These recorders are not involved in and therefore do not reduce any margin of safety in the TS.	F-15
MR-FC-94-019	Description: Raw Water Pump Seal Water Supply Upgrade Installation replaced piping, valves, filters, gages, pressure regulators, flow indicators, annunciator window, and new supports on pumps AC-10A/B/C/D. (The design and testing of this modification did not require a 10 CFR 50.59 Safety Analysis to be performed). Safety Analysis: The installation activities were performed with the plant at power with the applicable pump removed from service. During installation only the applicable pump was removed from service, other operating plant systems such as CCW, Instrument Air, or electrical power supply were not required to be position or manipulated in a manner different than as they are designed. Therefore, the change did not increase the probability or consequences of an accident previously evaluated, nor did it create the possibility of a malfunction different than previously evaluated. During installation, the availability and redundancy of the RW system met the requirements of TS, therefore the installation did not reduce any margin of safety.	None

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SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
MR-FC-93-002	Description: Two time-on delay relays were added to valves HCV-438A/C and HCV-438 B/D. Each relay is connected in series with the respective CCW pumps discharge header pressure switch to override and eliminate the momentary CCW system pressure fluctuation below the setpoint of 60 psia due to CIAS actuation. Safety Analysis: The containment isolation valves for supply and return of CCW to the reactor coolant pump seal coolers and CEDM seal coolers remain open upon receipt of a CIAS signal. The only time they automatically close is upon receipt of a CIAS in conjunction with CCW pressure low signal present for greater than 30 seconds. The modification added the 30 second delay so CCW is not lost to RCP seal cooling for pressure fluctuations. The 30 seconds plus 40 second stroke time of the valves is well below the safety analysis assumption of 120 seconds, therefore the change did not increase the probability or consequences of a previously evaluated accident. CCW flow to the RCP seal coolers will be enhanced and therefore the change did not increase the probability of a malfunction to equipment important to safety. The change did not affect any margin of safety in the TS.	None
MR-FC-93-022	Description: The modification replaced valves HCV-746A/B with Anchor Darling valves and included new Bettis air operators. Safety Analysis: The design change met the design, material, and construction standards equal to the original globe valves. There were no seismic issues since the valves are equal or lighter weight, therefore, the change did not increase the probability or consequences of an accident previously evaluated. The valves are a different type and manufacturer only, the controls remain the same, therefore the change did not increase the probability of a malfunction to equipment important to safety. The new valves provide a tighter shutoff and therefore have less leakage and as such did not decrease the margin of safety in the TS on containment integrity and leak rate.	None
MR-FC-92-036	Description: A non-CQE manual isolation valve was installed on the Demineralized Water Fill/Makeup Line to the Pressurizer Quench Tank. The isolation valve was installed to establish a pressure boundary on the containment side of isolation valve HCV-1560A/B. This allows the containment isolation valves to be pressurized during local leak rate testing from the direction in which the valves are required to perform their safety function. Safety Analysis: Installation of a manual isolation valve has no effect on the probability or consequences of any accident previously evaluated, or the malfunction of equipment important to safety. The supply of DW to the quench tank is not credited in any accident and does not effect the basis of any TS.	None

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SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
MR-FC-93-008	Description: A bypass switch was added to the Spent Fuel Handling Machine to allow it to travel beyond its stop limits to support reracking of the spent fuel pool. Safety Analysis: The modification provided the operator with visual indication and automatic speed reduction when a travel limit is exceeded. These safety features will allow the operation of the SFHM outside its normal limits without an increase in the probability of an accident, and did not effect the consequences of a fuel related accident. The change did not affect any safety related equipment, nor did it affect the margin of safety in the basis of any TS.	None
MR FC-94-002	Description: Obsolete ITE molded case circuit breaker on DC buses were replaced and new substitute equivalent Westinghouse switchboards were installed. Safety Analysis: The design, material, construction and functional/performance requirement of the obsolete circuit breakers were duplicated by the new boards. As such, no new failure modes or system interactions were introduced. Therefore, the probability and consequences of accidents previously evaluated were not increased, nor did the change increase the probability or consequences of a malfunction of equipment important to safety. The TS margin of safety were not effected by this change.	None
TM-95-035	Description: A temporary modification was installed to provide capability of switching RTD TE-121H to input to either D/TT-122H (a RPS temperature loop) or TT-121H (a RRS temperature loop) which indicates at AI-185 due to failure of RTD D/TE-122H. Safety Analysis: The temporary switches provided the same level of RPS performance as the originally specified RTD. The consequences of failure were not increased since the failure mode remained the same. Since operation and operability of the RPS did not change, there was not the possibility of a different type of accident. Development of the temp mod included evaluations that ensure that design criteria for RPS inputs such as qualification, separation, time response and accuracy were maintained. Therefore, the change did not affect the margin of safety as defined in the TS.	None
TM-96-023	Description: Temporary carbon filters were installed in ventilation housing for VA-33A/B upstream of the HEPA filters. This was done to provide additional limited iodine removal capability during a containment purge. Safety Analysis: The installation of the filters did not alter any equipment/system interface in a way that could increase the probability or consequences of an accident. They reduced the amount of radioactive iodine released as a part of normal purging of the containment. The installation did not introduce any new failure modes or accidents than previously evaluated, and did not reduce the margin of safety of any TS.	9.10

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SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
TM-96-024	Description: Temporary carbon filters were installed in place of each prefilter (VA-186A/B/C) in ventilation housing VA-18. This was done to provide limited iodine removal capability for the Auxiliary Building Controlled Access ventilation system. Safety Analysis: The installation of the filters did not alter any equipment/system interface in a way that could increase the probability or consequences of an accident. They reduced the amount of radioactive iodine released as a part of normal purging of the containment. The installation did not introduce any new failure modes or accidents than previously evaluated, and did not reduce the margin of safety of any TS.	9.10
EA-FC-94-029 Rev 0	Description: USAR Section 9.5 was revised to include discussion on the new fuel storage enrichment limits and add references. Safety Analysis: Engineering Analysis 94-029 concluded that the enrichment limit of the new fuel storage racks could be 5 weight percent nominal U-235. The change to the USAR documents the calculated design basis for the new fuel storage racks. The analysis documents that fuel storage with the analyzed enrichment limit will remain subcritical, thus the probability and consequences of previously evaluated accidents is not increased. The proposed change does not alter any setpoints, or physical components of equipment, and no modifications were necessary, therefore the change did not increase the probability or consequences of a malfunction of equipment important to safety. The margin of safety as defined by the original plant regulatory basis specifies controls for remaining subcritical and specifies k-eff limits. The change did not alter the k-eff limits and thus the margin of safety to prevent criticality was maintained.	9.5
USAR Clarification (CR 199500061)	Description: USAR Sections 11 and 14 were updated to reflect analyses performed for radiological consequences at 4.5 weight percent U-235. Safety Analysis: The effects from the 4.5 weight percent source term isotopics had previously been evaluated and updated in the USAR, but were listed as applying to 4.0 weight percent. This change only corrects documentation errors and includes additional information from previously performed analyses. Therefore, the change did not increase the probability of consequences of an accident previously evaluated nor did it increase the probability or consequences of a malfunction of equipment important to safety. The margin of safety was unaffected by this administrative change.	11, 14.12, 14.15, 14.18, 14.24

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SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
USAR Clarification (IR 950544)	Description: USAR Section 6.2 was clarified by deleting the Available Net Positive Suction Head values for the HPSI, LPSI, and Containment Spray pumps. Available NPSH is not a pump attribute, but rather an operating characteristic of the system in which the pumps is installed. USAR Section 6.2.1 already contains detailed discussion of NPSH during injection and recirculation phases. Safety Analysis: The changes made to the USAR tables have no impact on the performance of the safety injection and containment spray systems. The changes are administrative in nature and do not have any effect on hardware. No changes were made to the pumps themselves. Therefore, the change did not increase the probability or consequences of any previously analyzed accident, nor did the change increase the probability or consequences of a malfunction of equipment important to safety. The administrative change did not impact the margin of safety as defined in the TS for the LPSI, HPSI or CS systems.	6
USAR Clarification (CR 199601365)	Description: USAR Section 7.3.6 and 8.3.2 were revised to remove the portion of the description of shutdown outside the Control Room that refers to manual initiation of Safeguards in response to a DBA requiring automatic initiation, and provide description of the interlock preventing the 480V island buses from being energized from both electrical trains, and justify removal of an inadvertent mechanical interlock. Safety Analysis: The manual interlock was removed as it would prevent local manual closure in the event the control room was evacuated. As evaluated in EA-FC-96-054, the electrical interlocks remain functional and administrative controls are in place to ensure that mechanical closing is controlled. Therefore the change did not increase the probability or consequences of any accident previously evaluated. The safe shutdown power supplies meet single failure criteria where applicable and no new failure modes were determined. Therefore the change did not increase the probability or consequences of a malfunction of equipment important to safety. This change had no effect on any margin of safety.	7, 8
ECN-89-142 ECN-96-066	Description: ECN-89-142 upgraded the Raw Water Pump AC-10A/B/C/D bowls and cases to a more durable material. Safety Analysis: The original material for the pump casing was AISI 4330 alloy. ECN-89-142 approved the use of ASTM A743 CA6NM material and ECN-96-066 provided justification for the use of ASTM A487 CA6NM material. The USAR was revised to return the material specification to the original (AISI 4330) and indicate that approved equivalent material is acceptable. Equivalency will be evaluated and approved through a proceduralized configuration change process. The upgraded material is more resistant to wear due to abrasive material in the flow stream. No new failure mechanisms or increased failure frequency is introduced by use of the upgraded material. Therefore, the change did not increase the probability or consequences of any accident previously evaluated, nor did the change increase the probability or consequences of a malfunction of equipment important to safety. The margin of safety as it relates to the RW pumps is a function of system and component redundance and reliability. This change did not affect redundancy and or reliability but will increase the maintenance life of the pumps; therefore, the margin of safety as defined in the basis of the TS were not affected.	9.8

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SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
EA-FC-90-082	Description: USAR Section 5 is being revised to reflect the results of EA-FC-90-082. During the design basis reconstitution effort, an issue was identified related to post-DBA thermally-induced pressurization of liquid trapped between containment isolation valves. The EA concluded that pressure boundary integrity would not be compromised for penetrations susceptible to this phenomenon. Safety Analysis: No change in plant equipment or operational methods was performed. The EA concluded that pressure boundary integrity would not be compromised for penetrations susceptible to post DBA thermally-induced pressurization. Therefore, the change did not increase the probability or consequences of any accident previously evaluated. Since no change in plant operation or operational methods were performed the change did not increase the probability or consequences of a malfunction of equipment important to safety. Since boundary integrity/operability requirements were met, the margin of safety contained in TS 2.6 on containment was not reduced.	5.9, 5.12
PC 47733	Description: Procedure Change 47733 revised the operability requirements of the incore instrumentation in OI-NI-2 to allow continued operation with less than 75% operable incore instrumentation strings. Safety Analysis: An inadvertent loading of a fuel assembly into an improper location could not occur since the plant was at power and no fuel movements took place. For Cycle 6, ABB-CE analyzed a similar situation of failures. An explicit analyses of current and projected detector failure patterns was performed (CEN-150(O)-P. This analysis was formally submitted to the NRC and the NRC granted Amendment 55 to the TS. The changes made for Cycle 16 were the same as those approved in Cycle 6 except a TS change was not required and the minimum allowed operable strings was 28%. Since there have been no changes made to the design of the ICI system since Cycle 6, the analysis remains valid. An additional 1% uncertainty to the peaking factors and Peak Linear Heat Rate were added which was conservative since the Cycle 6 analysis determined that the increase in uncertainty up to 80% failed detectors was less than 1%. During the 1996 refueling outage, the inoperable detectors were replaced to ensure the USAR requirement to startup with at least 75% operable detectors was maintained. No changes were made to the configuration or operation of the system or plant, therefore the change did not increase the probability of an accident. Additional measurements and application of penalties on the values of measurements by the incore detectors prior to comparison with the limits of the TS and COLR ensured that the fuel design limits were maintained. Therefore, the change did not increase the consequences of an accident previously evaluated. Since no physical alteration of the ICI system occurred, the probability and consequences of a malfunction of equipment were not increased. The assumptions used in the safety analysis remained valid, therefore the change did not involve a reduction in any margin of safety as defined in the basis for any TS.	7.5, 7.7

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SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
TS Basis Clarification	Description: The Basis of TS 2.1.6 was clarified regarding the minimum number of Main Steam Safety Valves required to be operable on each main steam header as a result of EA-FC-97-004 and a Part 21 report issued by Combustion Engineering that identified that the CESEC computer code did not include line pressure losses in the calculation of peak secondary pressure. An application for amendment has subsequently been submitted to the NRC to incorporate this information. Safety Analysis: EA-FC-97-004 results showed that four of five MSSVs on each main steam header provide sufficient overpressure protection to keep peak steam generator secondary side pressure with 110% of the design for the transient associated with Loss of Load or a Loss of Feedwater event. No physical modification to the plant occurred. Therefore the change did not increase the probability of an accident previously evaluated. EA-FC-97-004 verified that the consequences of Loss of Load and Loss of Feedwater were not increased. Since no hardware or operational change to the plant was made, the change did not increase the probability or consequences of a malfunction of equipment important to safety. TS 2.1.6 requires 8 of 10 MSSVs be operable during power operation. Since EA-FC-97-004 shows that sufficient overpressure protection is provided based on 4 of 5 MSSVs on each main steam header, clarifying the Basis to state this did not reduce the margin of safety.	14.9, 14.10, 4.3
SAO-95-02	Description: SAO-95-02 specified that certain plant 480 V MCC loads must be deenergized when the plant was in Modes 1, 2, and 3 to address potentially defective RMS-9 trip breakers. The breakers were subsequently replaced under modification MR-95-05. Safety Analysis: The SAO opened certain load breakers to prevent grounds and required manual action to trip certain loads. The equipment tagouts, operator actions, and time available to restore equipment ensured adequate equipment was available to safely shutdown the plant with single equipment failures. The probability of an accident was not changed due to the RMS-9 issue. The actions taken to implement the SAO and the expected equipment performance ensured that the accident consequences were not increased. The SAO recommended actions and analyzed equipment performance allowed operation of equipment as designed such that the probability or consequences of a malfunction of equipment important to safety was not increased. Since adequate equipment was available, the SAO would have maintained the core within analyzed limits during a DBA and post DBA and therefore, no margin of safety was reduced.	None
SAO-96-01 Rev. 2	Description: SAO-96-01 Revision 2 was an operability evaluation of the tornado venting provisions for the Auxiliary Building Control Room and Cable Spreading Room. The reconstituted tornado venting calculations concluded that the Control Room and Cable Spreading Room need to be vented for the de-pressurization phase of a tornado event. Safety Analysis: The evaluation did not change the probability of occurrence of an accident previously evaluated. Actions assumed in the USAR were not degraded. Radiological consequences were not increased since there would be no failure of a safety-related structure. The de-pressurization phase of a tornado would meet the 1.5 psi differential pressure as stated in the USAR. Since no failure of safety-related structures would occur, the change did not increase the probability or consequences of a malfunction of equipment important to safety. Structural loads were determined to be acceptable for walls and floors as described in the USAR, therefore no margin of safety was reduced.	None

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SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
PC 48330	Description: Procedure Change 48330 revised CH-AD-0003 to incorporate the use of Carbonhydrazide as a secondary system oxygen scavenger. Safety Analysis: Research indicates the carbonhydrazide is more efficient at removing dissolved oxygen. It has also been shown to reduce iron transport rates into the steam generators which is an indication of reduction in the overall corrosion of plant systems. This enhances system integrity and reduces the likelihood of a tube rupture. Therefore, the change did not increase the probability of an accident previously evaluated and had no effect on the potential consequences of a previously evaluated accident. The chemical characteristics of carbonhydrazide are similar to hydrazine and will not adversely affect secondary plant components. Therefore, the change did not increase the probability or consequences of a malfunction of equipment important to safety and did not affect any margin of safety.	None
PC 46700	Description: Procedure Change 46700 revised OP-ST-RW-3003 to allow testing of HCV-482A/B and HCV-483A/B with nuclear fuel in the reactor. Safety Analysis: A dedicated operator was stationed at the CCW surge tank makeup valve to manually make up any lost CCW surge flow if automatic makeup should fail. Also RW backup to the shutdown cooling heat exchangers was available. The restrictions on RCS dilution, refueling operations and containment integrity specified in the TS are proceduralized. Therefore, the change did not increase the probability or consequences of any accident previously evaluated. The test required restoration of shutdown cooling at 25% of the calculated time to boil, and the ability to makeup any lost CCW surge volume remained available. Therefore, the change did not increase the probability or consequences of a malfunction of equipment important to safety. Since the test required restoration of shutdown cooling if shutdown cooling was secured for 25% of the calculated time to boil, the TS margin of safety was not reduced.	None
PC 46803	Description: Procedure Change 46803 revised procedure OI-CH-8 to allow use of an alternate chemical addition pump rig taking suction at the metering pump strainer flush line and discharging at the test tee on the metering pump discharge line for chemical injection into the RCS. This was completed due to the unavailability of chemical metering pump CH-3. Safety Analysis: The temporary chemical addition pump performed the same function as the existing pump. Equipment pressure protection was provided by the charging pump suction relief valves. Failure of the pump was not postulated to increase the probability or consequences of any accident previously evaluated. The temporary pump and CH-3 are normally isolated from the CVCS by a locked closed valve and are only used on an intermittant basis to add Lioh or N2H4 to the charging pump suction header. Directly affected equipment is non-safety related, therefore the change did not increase the probability or consequences of a malfunction of equipment important to safety. Providing an alternate chemical addition ability to the RCS did not reduce the margin of safety as defined in the TS.	None

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MR-FC-94-009	Description: MR-FC-94-009 replaced the carbon steel portion of the packing leakoff lines from HCV-1041A and HCV-1042A with stainless steel. Check valves MS-372 and MS-373 were replaced with Class II valves, and check valves MS-374 and MS-375 were removed. This modification was completed to eliminate repeated replacement of the lines due to erosion. Safety Analysis: The modification met design, material and construction standards applicable to the system. The change did not change, degrade, prevent actions prescribed or assumed, or alter any assumptions used in the USAR. Therefore, the change did not increase the probability or consequences of any previously evaluated accident. The change met or exceeded the original design specifications for design, material and construction. Therefore, the change did not increase the probability or consequences of a malfunction of equipment important to safety. Parameters used in determining the margin of safety remained unchanged.	None
MR-FC-95-007	Description: MR-FC-95-007 gagged relief valves AC-1026, AC-1027, and AC-1059 on the CCW system because there was a potential for them to lift after a DBA, which would cause the loss of CCW system inventory. Safety Analysis: Gagging the thermal relief valves did not increase the probability of any USAR Chapter 14 accident because the relief valves are not directly associated with initiating mechanisms for any Chapter 14 accident. The change did not adversely affect the post-DBA performance of the shutdown cooling heat exchangers or the CCW system, therefore the consequences of previously analyzed accidents were not increased. Thermal overpressure protection is still available to the respective heat exchangers since the shell side liquid has a path available to expand to the CCW surge tank. The maximum pressure on the shell side will still be within the Code maximum of 110% of design. The gagged valves will function adequately as passive parts of the CCW system pressure boundary. Therefore, the change did not increase the probability or consequences of a malfunction of equipment important to safety. The change did not affect the operability status or performance of the respective heat exchangers, therefore TS margins of safety were not affected.	None
EA-96-045	Description: USAR Chapters 3, 7, and 14 were revised to reflect the results of Cycle 17 Reload Analysis. Safety Analysis: The results of this analysis were incorporated into reactor core design and the setpoint analysis for determining the DNB and LHR LCOs for the cycle specific plant operation. Therefore, the change did not increase the probability of an accident previously evaluated. The cycle specific reload analysis, performed per NRC approved methodology, is performed to ensure that the consequences of previously evaluated accidents are not increased. The calculated plant parameters were found to be within the limiting values allowed by TS or as documented in previously evaluated accidents in the USAR. Therefore the change did not increase the consequences of any accident previously evaluated. The change has no effect on the probability or consequences of a malfunction of equipment important to safety since no new modes of plant or reactor equipment operation were created. The cycle specific reload analysis was evaluated against the margin of safety as defined in the TS and the margin of safety was maintained.	3, 7, 14

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SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
USAR Clarification	Description: USAR Section 9.2 was clarified for the conditions under which the RCS boron concentration may be reduced if the shutdown rods are inserted in the core. This was done because the USAR description was ambiguous. Safety Analysis: Requiring that the shutdown margin (SDM) boron concentration be maintained ensures that the probability and consequences of a boron dilution transient as currently documented in the USAR remain bounding and that there is no increase in probability or consequences of such an event. The clarification had no effect on the probability or consequences of a malfunction of equipment important to safety as the methods and procedures of operation remained unchanged. The minimum value of SDM assumed in the analyses was maintained, therefore the margin of safety as stated in the TS was not reduced.	9.2
EA-FC-95-020	Description: EA-FC-95-020 was prepared to analytically qualify the existing New Fuel Storage Rack for Seismic Class I earthquake. Safety Analysis: The evaluation had no effect on the probability of an accident previously evaluated. Qualifying the new fuel storage rack precluded the postulated consequences of an overturned rack which would cause damage to the SIRWT and damage to new fuel; therefore the consequences of a previously evaluated accident was not increased. No changes were made to the passive function of the rack and therefore the probability and consequences of a malfunction of equipment important to safety were not increased. With the rack seismically qualified, the criticality margin for the stored unirradiated fuel remained unchanged.	9.5
EA-FC-96-049	Description: Engineering Analysis EA-FC-96-049 documented the core redesign to accommodate reconstituted nuclear fuel. Safety Analysis: Reconstitution of fuel did not increase the probability of an accident previously evaluated since the mechanical integrity of the fuel was maintained. Also the neutronic, thermal/hydraulic and system performance criteria were maintained. By removing already failed fuel and replacing with stainless steel rods, the radiological consequences of previously evaluated accidents were not increased. Installation of filler rods, as long as they do not exceed 16, does not create hot channels, voiding, or severe shifts in power production. Thus, the potential for an increase in failure rate of fuel rods was not increased. Reconstitution was evaluated against Fr, Fxy, and Fq limits, SAFDL, PLHR, DNBRs and ROPM criteria. None of these limits were exceeded through evaluation of the use of reconstituted fuel, therefore the margin of safety was maintained.	3.7, 14.1
EA-FC-90-04	Description: EA-FC-90-04 revised the USAR to reflect the minimum SIRWT inventory used in the LOCA analysis of record. Safety Analysis: EA-FC-90-04 documents the amount of inventory assumed for the SIRWT. The change to the USAR does not increase the probability or consequences of any previously analyzed accident as it only documents assumptions in the analysis. The change does not alter equipment setpoints, structures, or frequency of testing systems and therefore does not increase the probability or consequences of a malfunction of equipment important to safety. EA-FC-90-04 determined that the margin of safety in the Basis of TS 2.3 was not impacted, nor does the change result in a violation of the 10 CFR 50.46/Appendix K acceptance criteria.	14.15

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
USAR Clarification (CR 199601041)	Description: USAR Section 9.2 was revised to clarify the maximum boric acid concentration for certain components since the current USAR Tables state the maximum concentration for certain components is 1 weight percent. Safety Analysis: The components assessed are made of materials qualified for service with boric acid concentration greater than 2400 ppm and will withstand the corrosive effects of the boric acid. Therefore the probability or consequences of a previously evaluated accident were not increased. Nor was the probability or consequences of a malfunction of equipment important to safety increased. The margin of safety is maintained by using materials qualified for their service environment. All components assessed are qualified for service with boric acid concentrations greater than 2400 ppm, therefore the margin of safety was not reduced.	3.4, 9.2
EA-FC-95-010	Description: EA-FC-95-010 evaluated the post-DBA Component Cooling Water Temperature. Safety Analysis: This EA added information to the containment analysis chapter because the methodology used a containment analysis computer model similar to that used in USAR Section 14.16. No change to the existing containment pressure analysis results contained in the USAR is changed, therefore the probability and consequences of an accident previously evaluated was not increased. No change to plant equipment or operational methods was made, therefore the change did not increase the probability or consequences or a malfunction of equipment important to safety. No change to plant equipment or operational methods was made Therefore, the margin of safety with regard to systems which remove heat from the containment following a DBA was not reduced.	14.16
USAR Clarification (CID 970019/02)	Description: The USAR was clarified based on recommendation from the NSRG to document the maximum boron concentration for the SIRWT as included in analyses. Safety Analysis: The USAR was clarified to state that the maximum boron concentration for the SIRWT is 2400 ppm, which is the same as the SITs. This is the upper limit as assumed in the LOCA analysis. As this value is already assumed and maintained, this change did not increase the probability or consequences of any accident previously evaluated, nor did it increase the probability or consequences of a malfunction of equipment important to safety. Since the assumptions of the accident basis remains unchanged, the margin of safety was unchanged.	14.15

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
EA-FC-96-042	Description: EA-FC-96-042 evaluated and revised Appendix M of the USAR to reflect a previously undocumented design feature of the plant which is required to mitigate the consequences of a HELB in Room 81 of the plant. Specifically, it has been determined that existing fusible link actuated trap doors and fire dampers located in HVAC ducts are required to vent steam into the turbine building. Safety Analysis: The trap door and fire damper do not directly interact, except for seismically, with any other component and they are adequately mounted to prevent an adverse seismic interaction. Therefore, the probability of an previously analyzed accident was not increased. Analysis has shown that the fusible links will allow the trap doors to open during steam migration, thereby mitigating the consequences of the HELB accident previously analyzed. Therefore, the consequences are not increased as a result of this change. The trap doors and fire dampers preclude the formation of a harsh environment which could lead to a malfunction of equipment. Therefore, the change did not increase the probability or consequences of a malfunction of equipment important to safety. Analysis has shown that the fusible links will preclude the environmental qualification limits for safety related electrical equipment from being exceeded. Therefore, the margin of safety was not reduced.	Appendix M
USAR Clarification	Description: USAR Section 9.8.2 was revised to clarify raw water direct cooling capability. Safety Analysis: No change to plant equipment or operational methods was performed. Therefore the probability and consequences of accidents previously evaluated were not increased. The revision more clearly describes the raw water direct cooling function as it presently pertains to sets of components having this capability. Therefore, the change does not increase the probability or consequences of a malfunction of equipment important to safety and no margin of safety was reduced.	9.8

Attachment B
LIC-97-0070

LIC-97-0070 Attachment B
USAR CHANGES OTHER THAN THOSE RESULTING FROM 10 CFR 50.59

DESCRIPTION	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
Engineering Assistance Request 94-058 revised the USAR to increase the low steam generator pressure pretrip setpoint to ≤ 600 psia in accordance with Amendment 153.	Table 7.2-1, Section 7.2.3.5
Document Change ECN 95-365 revised the USAR to reflect the addition of a new 161 kV line (1584) to Substation 1226.	Sections 8.2.1, 8.8.2, Figures 8.2-1, 8.2-2
The USAR was revised as a result of a discrepancy identified during closure of a design basis open item. The USAR incorrectly stated that a humidifier is installed in the computer room supply duct. The humidifier is actually wall mounted and discharges directly into the computer room and not through the supply duct.	Section 9.10.2.4
Procedure Change (PC) 47733 placed In-core Instrumentation (ICI) operability requirements formerly located in Technical Specification 2.10.3 into Procedure OI-NI-2. Amendment 167 removed Technical Specification 2.10.3 based on relocation of the ICI operability requirements to the USAR and station procedures.	Section 7.5.4.3
Section 12 of the USAR was revised to incorporate new management position titles and to revise the experience required for the positions of Manager - Station Engineering and Manager - Design Engineering.	Section 12
A reference to EA-FC-96-001"Criticality Safety Evaluation of the Fort Calhoun Spent Fuel Storage Rack for Maximum Enrichment Capability" was added. The EA was done to support Facility License Change No. 95-11, which was NRC approved as Amendment 174	Sections 9.5.3.4 & 9.5.8
A reference to the Safety Evaluation Report for Amendment 172 on the chemical and volume control system was added.	Section 9.2.9
A sentence that references the use of WCAP-13060-P-A, "Westinghouse Fuel Assembly Reconstitution Evaluation Methodology" was removed as this methodology is only applicable for Westinghouse fuel with mixing vane grid designs. The Westinghouse fuel used at FCS does not have mixing vane grids.	Section 3.7.3
In response to Information Notice 95-54, a revision to Section 5.0 was made to clarify that full core off-loads are not an abnormal condition. Section 9.6.2 was revised to clarify that the shutdown cooling system provides an emergency backup for the spent fuel pool cooling system when the plant is shutdown and the core has been fully offloaded.	Section 5.0 and Section 9.6.2
Tables 3.4-13 and 3.4-14 were revised to add results from EA-FC-96-040, "CPTP #41 End-of-Cycle 16 At-Power MTC Test."	Tables 3.4-13 & 3.4-14
A review of the USAR following receipt of Amendment 173 resulted in an administrative correction in Section 7.3 to replace an incomplete description of the acronym STLS (safety injection and refueling water (SIRW) tank low level signal).	Section 7.3
The USAR was revised to clarify that the safety analysis assumes a nominal flow rate of 120 gpm from three (3) charging pumps.	Section 14
USAR Section 9.5 was revised to increase the minimum boron concentration that must be maintained in the spent fuel pool from 100 ppm to 500 ppm based on receipt of Amendment 174.	Section 9.5
USAR Section 7.6 concerning Panel AI-179 was revised to make it consistent with the correct description of Panel AI-179 contained in Section 7.6.4(a) and in Appendix M.3.5.1.	Section 7.6.1
Drawing updates to Figures 7.3-1, 8.1-1, 8.5-8, 8.5-9, 8.5-10, and 8.5-11.	Sections 7 and 8

LIC-97-0070 Attachment B
USAR CHANGES OTHER THAN THOSE RESULTING FROM 10 CFR 50.59

DESCRIPTION	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
USAR Section 4.4 was revised in accordance with Amendment 179 to increase the amount of tri-sodium phosphate (TSP) required to be in the containment sump storage baskets.	Section 4.4
Sections 11.2.3.4 and 11.4 were revised to incorporate information from a NRC Safety Evaluation Report evaluating OPPD's updated response to Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an accident."	Sections 11.2.3.4 and 11.4
Table 14.1-1 was revised to correct a typographical error pertaining to the delay time for the containment pressure high trip discovered during an ABB/CE RPS Delay Time Design Basis Review.	Table 14.1-1
Amendment 181 revises Section 5.2 of the Technical Specifications to relocate controls for working hours to the USAR.	Section 12.1.5.1

REVISION TO QA PROGRAM	REASON WHY CHANGE IS NOT A REDUCTION IN A COMMITMENT
Transfer of personnel performing external auditing of suppliers, purchase order review, receipt package development and receipt inspection, to Production Engineering Division.	Certain activities previously performed by Quality Assurance and Quality Control personnel were transferred to the Production Engineering Division. These activities include performing external auditing of suppliers, purchase order review, receipt package development and receipt inspection. As part of this transfer, an evaluation was made to determine if there had been any reduction in commitment to our QA Plan or USAR requirements. Currently OPPD is committed to having a quality program that requires the OPPD organizations and companies under contract to supply technical services or products for the plant to comply with established requirements. These requirements ensure that personnel have the authority and freedom to identify quality problems and recommend solutions for conditions adverse to quality. Additionally, the plan ensures that individuals or groups assigned the responsibility for auditing an activity are not directly responsible for performing the specific activity being audited. USAR Appendix A outlines the regulatory guides and standards with which OPPD is committed. Standards such as 10 CFR Part 50 Appendix B, ANSI N45.2-1977, ANSI N45.2.13-1976 and N45.2.23-1978 are described within Appendix A of the USAR, and have been reviewed to ensure that no reduction in commitment was made due to the reorganization. Attributes such as inspection independence, production independence, personnel qualifications, and auditing were evaluated to ensure that activities necessary for maintenance of the QA Program would not be compromised. To ensure that the transferred activities continue to meet the requirements established within the USAR and QA Plan, the QA internal auditing program will verify that the attributes have been maintained. Audit #13, Procurement Control, will ensure items such as personnel qualifications, external auditing, receipt inspection and freedom from the pressures of production are maintained. Based upon this review, commitments established by OPPD have not been reduced.
Administrative changes	USAR Appendix A was revised to reflect organizational changes and to reflect new position titles resulting from a reorganization of the various nuclear divisions. A review of these changes determined that they do not reduce QA program commitment.

REGULATORY COMMITMENT CHANGES

The following Regulatory Commitments were revised based upon evaluations made in accordance with the NEI Guideline For Managing NRC Commitments. This NEI Guideline was accepted in SECY-95-300.

SOURCE OF COMMITMENT	REVISION TO COMMITMENT
IE Bulletin 79-21	In response to IE Bulletin 79-21 (LIC-79-0117, 9/12/79) OPPD stated "Emergency Procedures are presently being revised to include specific information relating corrections that should be applied to level measurements by plant operators during post-accident monitoring and to assure that all applicable curves, tables, etc relating level measuring correction factors are available." The results of the OPPD evaluation on the effect of post-accident ambient temperatures determined that the information required for Steam Generator level could be presented in four figures instead of the seven figures needed to satisfy the original commitment made in the response to Information Bulletin 79-21.
OPPD Safety Enhancement Program	In the OPPD Safety Enhancement Program (LIC-88-1094, 12/9/88) OPPD stated "The Senior Vice President has conducted and will continue to conduct special monthly (or more frequent) meetings with the three nuclear Division Managers and the department managers who report to them." This commitment was made while FCS was on the "watch list." At that time conditions were rapidly changing and these special meetings were required to assure rapid dissemination of information. Under present conditions meeting of this frequency are still planned but not required, thus the words "monthly (or more frequent) meetings" were changed to "regularly scheduled meetings, normally on a monthly basis," allowing some flexibility.
LER-89-018	In Fort Calhoun Station LER-89-018 (LIC-89-0777, 9/14/89) a long-term corrective action necessary to minimize recurrence of the adverse condition was made. This Regulatory Commitment included Specific requirements for "the signature of the fire protection engineer or the operations shift supervisor to clear a fire watch" in the hourly fire watch log (Form FC-1006), and also noted that the Fire Impairments Log (Form FC-1140) was revised to require that the fire protection engineer be contacted prior to terminating any hourly fire watch. As revised, verbal notification is now allowed. Further, compensatory measures can only be released at the direction of the Fire Protection System Engineer or the Operations Shift Supervisor, and this release is now documented on the FC-1142 Form.

Attachment C
LIC-97-0070