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Omaha, Nebraska 68102-2247
402/636-2000

July 1, 1994
LIC-94-0142

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

Reference: Docket No. 50-285

Gentlemen:

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

As required by 10 CFR 50.59(b)(2), please find attached as Attachment A, 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 provides a summary of Updated Safety Analysis Report (USAR) changes other than those resulting from 10 CFR 50.59. Pursuant to 10 CFR 50.71(e) and 10 CFR 50.4(b)(6), please find attached as Attachment C, one original set of inserts and 10 copies of the USAR changes for the Fort Calhoun Station. The original set is designated as Copy Number 1 and the 10 copies as Copy Number 2 through 11. This information is for the period of January 23, 1993 through December 31, 1993.

If you should have any questions, please contact me.

Sincerely,

W. G. Gates

W. G. Gates
Vice President

Attachments

WGG/mle

c: LeBoeuf, Lamb, Greene & MacRae (w/o Attachment C)
L. J. Callan, NRC Regional Administrator, Region IV (Copy #13)
S. D. Bloom, NRC Project Manager (Copy #12)
R. P. Mullikin, NRC Senior Resident Inspector (Copy #15)

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UNITED STATES 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 1 day of July, 1994.

Melva L. Champion
Notary Public

GENERAL NOTARY-State of Nebraska
MELVA L. CHAMPION
My Comm. Exp. 2-11-96

Attachment A

10 CFR 50.59 REPORT
JANUARY 23, 1993 THROUGH DECEMBER 31, 1993

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S),SECTION(S), OR FIGURE(S) REVISED
OI-MS-2 PC 35155	Description: The procedure change (PC) changed the position of Valves MS-338, MS-339, MS-341, and MS-342 (main steam isolation valves (MSIV) packing leakoff valves) from closed to open. The manufacturer of the MSIVs recommended that the packing leakoff valves remain open to preclude the use of special packing material and avoid possible blowby of steam and potential injury to maintenance personnel while replacing packing material. Safety Analysis: The safety evaluation analyzed the effect of changing the normal position of the MSIV packing leakoff valves from closed to open. The evaluation determined that opening the valves not only benefits personnel safety, but that the configuration potentially reduces the consequences of a steam generator tube rupture (SGTR). The margin of safety as defined in the Technical Specifications (TS) bases is not decreased. The probability or consequences of any previously analyzed or unanalyzed accident is not increased since opening the valves does not adversely affect any safety related equipment.	Figure 5.9-13, Sheets 62 and 63
PC 40622 CH-AD-0003	Description: This procedure was changed to revise the reactor coolant system (RCS) oxygen specification temperature requirement from 150°F to 250°F. Safety Analysis: The oxygen specification applies during power operation (per TS 2.1.5(1)). The procedure change did not alter the specification. Therefore, the probability or consequences of any previously analyzed or unanalyzed accident is not increased. The Electric Power Research Institute (EPRI) and Asea Brown Boveri/Combustion Engineering (ABB/CE) have both determined that there are no detrimental effects from oxygen in the RCS while RCS temperature is less than 250°F.	Section 4.3.13

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CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), OR FIGURE(S) REVISED
PC 40297 (OI-WDG-1) PC 40296 (OI-WDG-3) ECN-91-271	Description: The procedure changes (March 1993) deleted references to AI-100 and AI-110 alarms which alert operators when flammable concentrations of H_2 and O_2 are accumulating in the waste gas decay tanks because the alarms were inoperable. The concentrations were monitored by the operators during waste gas transfers. In August 1993, Engineering Change Notice (ECN) 91-271 added two Action-Pak limit alarm relays (YIA-628, YIA-627) to the AI-110 H_2 and O_2 analyzer systems, installed some wiring changes to the existing circuitry and corrected some minor drawing discrepancies. The ECN provided for annunciation at AI-100 when the concentration of each gas (H_2 & O_2) reaches 3% (flammable concentrations occur at 5.4% O_2 and 4% H_2). Alarm lights are provided on AI-110 when either gas reaches (or exceeds) 3%. Subsequent to the ECN, procedures OI-WDG-1 and 3 were revised to again reference the alarms. Safety Analysis: These alarms function to alert the operators when flammable concentrations of H_2 and O_2 accumulate in the waste gas decay tanks. Prior to the completion of ECN 91-271, the operators monitored H_2 and O_2 concentrations during the transfer of waste gas. Technical Specification 2.9(2) allows transfer of waste gases with the monitors inoperable provided grab samples are taken and analyzed. The ECN changes do not reduce the margins of safety delineated by the TS basis, nor do they impact accident mitigation or probability of bounded accidents evaluated in the USAR.	Section 11.1.3.5
MR-FC-85-093	Description: This modification involved the addition of a 6-cylinder halon bottle rack; revised halon piping and nozzles in the east switchgear room, west switchgear room, cable spreading room and the control room walk-in cabinets; removed the auto-actuation abort switches for the same aforementioned areas; added or modified logic for the master pull stations, and modified various aspects of the logic which activates the switchgear room halon zones. Safety Analysis: This modification was an enhancement to the existing non-critical quality element (CQE) system and did not involve failure modes or system interactions that have not been analyzed. The non-CQE components that comprise the halc suppression system are not identified as accident initiators. There is no adverse impact on procedures, structures, systems or components which contribute to nuclear safety. The margin of safety is not reduced through these design enhancements, i.e., increased halon capacity and improvement of the pipe configuration.	None

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CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S),SECTION(S), OR FIGURE(S) REVISED
MR-FC-83-173	Description: This modification provided for the installation of 8 new primary sample coolers and sample piping modifications for the secondary plant sampling system, and for a bypass around the secondary sample coolers to Radiation Monitors RM-054 A/B in the steam generator blowdown sampling system. Safety Analysis: This modification to the secondary plant sampling system and the steam generator blowdown sampling system increases the accuracy of water chemistry monitoring in the secondary plant and steam generators. This permits tighter control of water chemistry in the secondary plant and steam generators and reduces the probability of a steam generator tube rupture (SGTR). No nuclear safety systems or critical quality element (CQE) components were affected by this modification. The chemistry sampling systems are not safety related nor are they required in the event of an accident. Therefore, the probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	None
MR-FC-92-025	Description: During the 1992 Refueling Outage (RO), Valve CA-555 (containment isolation valve for service air) and the downstream piping were removed and a blank flange was installed. Modification MR-FC-92-025 installed a qualified safety class 2 replacement valve for Valve CA-555 during the 1993 RO. Piping downstream of Valve CA-555 connecting to the service air header in containment was also reinstalled. Safety Analysis: This modification restored the system to its original design by replacing the previous valve (replaced with a blank flange in 1992 because it was unqualified) with a qualified safety class 2 replacement. As a result, no new failure modes were created and operation of the system is identical to the original licensing basis. During power operations, the service air piping is passive and unpressurized, isolated inside containment by a locked closed manual valve (CA-555) and outside of containment by a spring to close, normally closed valve (HCV-1749). During refueling operations when containment integrity is required, the system is pressurized greater than 60 psig. In addition, Valve HCV-1749 will fail closed upon receipt of a containment isolation actuation signal (CIAS).	Figure 5.9-13, Sheet 45

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S),SECTION(S), OR FIGURE(S) REVISED
MR-FC-92-031 CWO-92-0048	Description: This modification installed two new constant voltage regulating transformers (one on Inverter '1' and one on Inverter '2'), which can be switched to carry the inverter loads when the inverters are out of service for maintenance or testing. Temporary power was required to install the transformers. Therefore, CWO-92-0048 was initiated to install a temporary jumper to supply power between Buses AI-42A and AI-42B using two spare breakers. Safety Analysis: Operation of the non-safety instrument buses powered with the new inverter testing transformers is equivalent to operation of the inverter with a bypass transformer. Therefore, the probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	Section 8.3.5.2
MR-FC-89-081	Description: This modification added two valves in each warm up line bypassing Valves YCV-1045A (main steam line "A" to auxiliary feedwater (AFW) pump FW-10) and YCV-1045B (main steam line "B" to AFW pump FW-10). Valves MS-336 and MS-337 were removed as the new valves duplicate their function. One of the two valves is used for throttling and the other is used as a manual isolation valve. A pressure switch was added downstream of Valve MS-295 to provide an alarm in the control room if pressure in the steam line drops below 400 psi. This indication signals operations to close the warm-up lines if there is a steam line break in the associated piping. Safety Analysis: The functional configuration of the system is unchanged except for the addition of a pressure switch which does not introduce adverse interactions. The slightly lower initial steam line pressures and temperatures do not impact safety. The Technical Specification margins of safety are unchanged. The probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	Figure 5.9-13, Sheets 62 & 63

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S),SECTION(S), OR FIGURE(S) REVISED
MR-FC-93-021	Description: Ground detection was added to the control element drive mechanism (CEDM) power supply and to the rod drive position mimic power supply. Indication and annunciation were provided to alert operators of a ground on either system. Safety Analysis: The modification does not change the operation of the rod drive system. However, it does enhance the ability to detect system grounds and prevent a malfunction. Since the operation of the rod drive system is not changed, there is no increase in the probability/consequences of a previously analyzed accident. Similarly, the probability/consequences of a malfunction of equipment important to safety is not increased, nor are any new failure modes created by this modification.	None
MR-FC-93-015	Description: This modification installed a valve and tubing on each power operated relief valve (PORV) loop seal piping so that no loop seal water forms. This maintains the PORV internal temperature at close to pressurizer conditions which enhances PORV operation and reliability. Safety Analysis: The new loop seal components are equal in pressure integrity to the previous components and the PORVs were tested under actual steam and water conditions prior to unit criticality. Since the loop seal drain system will ensure PORV temperature uniformity with pressurizer conditions, the PORVs will become more reliable as far as opening upon demand to maintain pressure safety limits for the RCS. The performance of the PORVs and piping will not change with regard to flow, actuation point, response time, etc. Therefore, the modification does not increase the probability/consequences of a previously analyzed accident. The increase in probability of a PORV tubing rupture is small considering the number of existing RCS instrument taps for level, flow and pressure. A loop seal drain failure caused by plugging is mitigated by an upper drain which will return loop seal volume to the pre-modification state.	None

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S),SECTION(S), OR FIGURE(S) REVISED
MR-FC-90-054 EA-FC-93-022	Description. This modification replaced the gearing and modified the circuit logic to de-energize Valves HCV-1103, HCV-1104, HCV-1385, HCV-1386 on torque rather than position. This reduces the inertial forces affecting the open and closed travel of the valves. HCV-1103, HCV-1104, HCV-1385 and HCV-1386 valve motors were replaced. Motor brakes were removed from all four valves. Safety Analysis This modification does not reduce the margin of safety established in the Technical Specifications. Generic Letter 89-10 dynamic MOV test results provide assurance that these valves will operate as designed. There is no impact on any existing radioactive release path and no new paths were added. The probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	Section 14.16
MR-FC-86-061	Description: This modification provided additional communication capabilities for Fort Calhoun Station including an 800 MHz radio system which utilizes OPPD's corporate 800 MHz trunking system and 158 MHz paging system. This system consists of hand-held portable radios, an interstation RF amplifier, antennas, and "leaky" coax system for signal distribution throughout highly shielded areas of the plant. Safety Analysis: The new radio and paging system does not affect the margin of safety as defined in the Technical Specifications. No credible event can be postulated such that the modification would increase the probability of occurrence or consequences of any analyzed or unanalyzed accident. Walkdowns performed in the design phase provide a very high degree of assurance that electromagnetic interference (EMI) is not a concern. Additionally, the use of radios is prohibited in sensitive electrical areas.	Sections 7.6.6, 7.6.7

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), OR FIGURE(S) REVISED
MR-FC-87-044	Description: The vibration and loose parts monitoring system (VLPMS) was upgraded with a digital data acquisition and analysis subsystem and now has additional vibration and loose parts sensors. Safety Analysis: This modification does not constitute an unreviewed safety question as the modification enhanced the performance of the existing system by replacing outdated equipment with both new and additional equipment. The basic function of the new system remains the same as the old system. Therefore, the modification does not increase the probability of occurrence or consequences of any analyzed or unanalyzed accident.	Sections 3.7.4, 3.9, 4.5.9, Figure 7.6-1
MR-FC-88-028	Description: This modification increased the container volume for hydrazine from 55 to 365 gallons. This action reduces both the frequency of material handling and the probability of a spill. Also, the consequences of a spill are not increased because the control room remains habitable (below TLV levels) even with a larger spill. The toxic gas and control room heating, ventilation & air conditioning (HVAC) systems are unaffected by the modification. Safety Analysis: A spill of a 365 gallon container filled with a 35% Hydrazine solution will not result in exceeding TLV levels in the control room and there are no direct affects on any other safety related activities. The probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	None

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S),SECTION(S), OR FIGURE(S) REVISED
MR-FC-88-052	Description: This modification changed the water supply to the water plant pretreatment system to utilize water from the Blair Municipal Water System. Previously, the water plant pretreatment system utilized river water from either the raw water (RW) or circulating water (CW) systems. The water plant pretreatment system supplies water to the demineralized water (DW) system and is a backup supply to the potable water system. The RW and CW systems are now isolated from the water plant pretreatment system by locking closed the DW inlet isolation valve. Safety Analysis: This modification has no direct or indirect effect on the failure of equipment important to safety. Additionally, the new water supply system meets or exceeds all original design requirements and therefore, nuclear safety is not affected. The probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	None
MR-FC-88-121	Description: This modification provided system interconnections (auxiliary steam, demineralized water etc.) for the radioactive waste processing building (RPB) and the chemical and radiation protection (CARP) building with the existing plant systems in the auxiliary building. Tie-ins to the existing plant systems were necessary for the operation and service of the new RPB and CARP buildings. Safety Analysis: These systems are all non-CQE. No new interfaces or failure modes which can challenge CQE systems were created by the installation of the tie-ins. The probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	Section 11, Appendix F

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), OR FIGURE(S) REVISED
MR-FC-91-034	Description: This modification removed unnecessary valves and tubing from the AFW system thereby reducing the potential for leakage. Valves FW-1006 and FW-1470 on the AFW piping going to Steam Generator RC-2A were removed and replaced with a weldlet and plug. Valves FW-1007 and FW-1471 for Steam Generator RC-2B were removed similarly. The interconnecting tubing between Valves FW-1470 and FW-1471 was removed also. Safety Analysis: This modification involved the replacement of one passive minor component with another. The replacement of FW-1006/FW-1470 and FW-1007/FW-1471 with equivalent (or better) passive pressure containing components does not adversely affect any function of the AFW system. The probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	Figure 5.9-13, Sheets 60 & 65
MR-FC-92-007	Description: This modification added a bypass line and normally closed isolation valve in parallel with the VA-82 hydrogen purge filter train. The bypass was added to ensure purging capability if the filter were to become plugged. Safety Analysis: The design does not constitute an unreviewed safety question. The new VA-82 bypass line does not impact the ability of the hydrogen purge system to perform its post-LOCA combustible gas control function. VA-82 filtration is not required for combustible gas control and the filter is not credited in the dose analysis for any accident or anticipated operational occurrence.	Sections 9 & 11 Figure 9.10-1

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S),SECTION(S), OR FIGURE(S) REVISED
MR-FC-89-013	Description: This modification refurbished the 480V load centers switchgear circuit breakers. The refurbishment included: breaker disassembly for inspection, cleaning and lubrication, the replacement of the existing electro-magnetic trip units with solid state trip units, replacement of the trip shaft bearings and the trip latch roller assembly followed by post refurbishment mechanical and electrical testing. Safety Analysis: The design function and method of operation of the 480V switchgear is unaffected by the replacement of the trip units. Breaker coordination is enhanced and breaker reliability is improved by replacement of the electro-mechanical trip devices with solid state trip units. The modification does not increase the probability of occurrence or consequences of previously analyzed accident or unanalyzed accidents since the solid state trip units are more accurate and reliable than the electromechanical trip units currently installed. The potential for breaker failure and subsequent load failure has been evaluated previously and remains unchanged.	None
TM-93-050	Description: Temporary changes to the emergency diesel generator (EDG) auto-start circuitry inhibited twelve of sixteen auto-start initiating signals to prevent unnecessary challenges to safety systems during planned maintenance. The temporary modification was installed for the 1993 Refueling Outage only. Only one diesel was affected at a time. The twelve signals affected are considered to be anticipatory starts. The genuine low bus voltage auto-start signal, pressurizer pressure low signal (PPLS), containment pressure high signal (CPHS) and manual pushbuttons were not inhibited. Safety Analysis: A Technical Specification bases review determined that no credit is taken for anticipatory EDG auto-starts. System redundancy is maintained since each EDG contains two redundant channels of circuitry, either one of which is able to auto start the EDG following a low bus voltage signal. The probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	None

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S),SECTION(S), OR FIGURE(S) REVISED
TM-93-063	Description: This temporary modification removed Valves AC-341 (CCW Surge Tank N_2 Relief Valve) and PCV-2839 (CCW Surge Tank N_2 Vent Header Pressure Relief) from the component cooling water (CCW) surge tank for setpoint and seat leakage testing. The plant was in Mode 5 with the core fully off-loaded. Blank flanges were installed to permit continued operation of the system with minimal CCW system outage time. After completion of setpoint and seat leakage testing, the valves were returned to service. Safety Analysis: With the plant in Mode 5, the installation of blank flanges in place of Valves AC-341 & PCV-2839 did not create or increase the likelihood of an accident. No credit is taken for this relief valve to prevent overpressure. Thermal expansion in the surge tank is considerably less during Mode 5, therefore, the temporary modification did not affect the operational capability of the CCW system.	None
TM-93-049	Description: A temporary blind flange was installed immediately downstream of Valve HCV-400F (Containment VA-1A Cooling Coil RW Backup Outlet Valve) to isolate leak-by occurring across the valve seat. The temporary modification was removed prior to startup from the 1993 Refueling Outage. Safety Analysis: The temporary blind flange restored the pressure retaining boundary of the CCW system. The flange did not affect the normal operation of either the RW or CCW systems and did not prevent any related component from fulfilling its required safety function. The blind flange did not put the system in an unanalyzed condition or increase the potential or consequences of an accident.	None

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CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S),SECTION(S), OR FIGURE(S) REVISED
PC 41787	Description: This procedure change updated TDB-VI (Core Operating Limits Report (COLR)) to reflect Cycle 15 safety and setpoint analyses results. Safety Analysis: Changing TDB-VI to reflect the Cycle 15 analyses ensures that the margin of safety is not reduced. These changes do not impact any system(s) which could increase the probability or consequences of any previously analyzed accident or create the possibility of a new type of accident.	Section 3.6.5, Figures 3.6-1, 3.6-2
PC 40606	Description: USAR Section 11.3 was revised to reflect changes in the Offsite Dose Calculation Manual (ODCM) and take credit for the addition of Regulatory Guide 1.109. Safety Analysis: These changes to the USAR are administrative in nature. All modifications to equipment or systems important to nuclear safety were reviewed and approved by the NRC in the safety evaluation report (SER) for Amendment No. 152.	Section 11.3.3

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S),SECTION(S), OR FIGURE(S) REVISED
PC 40924 MR-FC-90-070	Description: The ODCM was revised to allow the containment building to be depressurized following completion of Type A Integrated Leak Rate Testing through a vent path that does not have automatic isolation capability. Manual isolation of the effluent release is administratively controlled. Exemption of the automatic isolation requirement is valid only for post-ILRT venting. Safety Analysis: This change to the ODCM does not reduce the effectiveness of the radiation effluent controls. The change does not affect the basis for any Technical Specification or adversely affect any safety related equipment. Containment venting following completion of the ILRT is conducted when the Technical Specifications do not require containment integrity. The changes described are in compliance with federal regulations and standard industry practices. Administrative guidance is provided to ensure that radiation effluent controls are not reduced. The probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	Sections 7.3.2.6 & 11.3.2.1
PC 41493 RE-CPT-NI-0001	Description: A new procedure was developed to determine the shape annealing factor (SAF). This allows the SAF's to be determined during a power increase without inducing a xenon oscillation. Safety Analysis: This change does not affect the margins of safety in the USAR. It collects axial shape index data during a power change. This change closely monitors core parameters for which Technical Specification limits apply during the power change and provides the operators with additional time to take the appropriate corrective actions if needed. The probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	Section 3.4.8.5

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S),SECTION(S), OR FIGURE(S) REVISED
SDBD-CONT-501	Description: Section 5.9.5 of the USAR was revised to clarify the requirements and descriptions of mechanical containment penetrations. The change also revised penetration diagrams (USAR Figure 5.9-13 sheets) to reflect the revisions to section 5.9.5. These clarifications address issues raised in three open items in the containment system design basis document SDBD-CONT-501. The clarifications will allow the open items to be closed. Safety Analysis: This USAR change does not constitute an unreviewed safety question (USQ) because the revised wording still ensures that containment penetration configurations satisfy the nuclear safety function of the containment, which is to prevent the release of radioactivity from the containment in the event of an accident. The probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	Section 5.9.5, Figure 5.9-13 (numerous sheets)
SAO 90-008, Rev 1	Description: SAO 90-008 Rev. 1 re-approved/reissued SAO-90-008. SAO-90-008 identified a potential containment bypass in the CCW system should concurrent LOCA and loss of DC power events occur. In response to SAO 90-008 procedure changes were implemented in 1990 to the following procedures: 1) CH-ST-RM-0052/0053 alert setpoints for Radiation Monitors RM-050 and RM-051, 2) OP-1 to require that at least one of the defined RCS leak detection systems is operable upon startup, 3) OI-RM-1 and 4) Form FC-75 "Control Room Log." Safety Analysis: SAO 90-008 Rev. 1 justifies continued operation since FCS is operating in a more conservative manner with respect to current Technical Specifications. SAO 90-008 Rev. 1 requires more stringent monitoring and equally accurate indication of RCS leaks so that action is taken prior to the potential for a primary loop throughwall circumferential pipe break. NRC approved leak before break (LBB) methodology will be implemented upon NRC approval of Facility License Change (FLC) 90-15, submitted August 20, 1993 (Revision submitted on June 6, 1994). Implementation of LBB methodology ensures a more conservative FCS position regarding the ability to detect a RCS primary loop throughwall circumferential pipe break.	USAR change to be initiated upon NRC approval of FLC 90-15

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S),SECTION(S), OR FIGURE(S) REVISED
TDB-III.26.A PC 38794 ECN-91-306	Description: Technical Data Book (TDB) curve TDB-III.26.A has been changed by data and analysis. The calculation indicates that the maximum KW rating of the EDGs has been reduced from 2654 KW to 2627 KW and the high temperature operability limit for EDG-1 was reduced to 104°F from 110°F. Safety Analysis: While the use of a glycol/water mixture causes a greater engine derating, there is no increase in the probability that the EDGs would be unable to perform their safety function. During a postulated accident and loss of offsite power on a day when the ambient temperature is 110°F, the EDGs would still be capable of performing their required functions. The operability limit for EDG-1 was reduced to 104°F because emergency loading of EDG-1 is higher than EDG-2. This does not mean that EDG-1 will not carry the emergency loads, it simply means that the maintenance inspection interval should be shortened to something less that the 2000 hour engine rating. The 2000 hour rating of the engine is the amount of power that the engine can routinely produce before a maintenance inspection should be performed. Therefore, the probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	None
PC 41160 RS-ST-MM-0002	Description: Procedure RS-ST-MM-0002 was revised to change reference to Technical Specifications, Section 3.11 to the ODCM, change responsibility from Manager-Radiological Services to the Supervisor-Radiochemistry. Flynn Dairy was removed as a milk sample collection site because it no longer has milk cows. Safety Analysis: Both the USAR and the ODCM allow for deviation from the monitoring program to occur due to participants ceasing participation in the program. No new milk locations were identified in the last land census. Therefore, the probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	Section 2.10 Figure 2.10-2 Figure 2.10-4

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SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S),SECTION(S), OR FIGURE(S) REVISED
USAR Appendix N	Description: USAR Appendix N, Section N.3.4 was revised to state that Valves HCV-2895A/B (Waste Evaporator CCW Inlet/Outlet Valves) provide adequate safety class boundary interfaces, which is an exception to Section N.3.1. Section N.3.5 was revised to show a safety class interface between non-nuclear safety (NNS) CL-1 & NNS CL-2. Table N-1 was revised to show the waste evaporator, vacuum deaerator, and primary sample cooler as NNS CL-1. Safety Analysis: The CCW closed loop piping although not CQE, was shown to be adequately designed for the temperatures and pressures involved in the event that HCV-2895 A or B fail to close upon demand. This piping can be relied upon to maintain the pressure boundary of the CCW System. The probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	Sections N.3.4, N.3.5, Table N-1
PC 45230 RE-RR-SFP-0700	Description: Based on the guidelines of NUREG-0612, procedure RE-RR-SFP-0700 was created to provide administrative controls over the transfer of a reactor vessel surveillance capsule from a spent fuel pool storage location to a shipment cask. The cask was then secured to a trailer for shipment offsite. Safety Analysis: Heavy load handling in the area of the spent fuel pool per procedure RE-RR-SFP-0700 meets the guidelines of NUREG 0612. The NUREG 0612 guidelines ensure that the potential for a load drop is extremely small and provides a defense-in-depth approach to control the handling of heavy loads. The probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	None

10 CFR 50.59 REPORT
JANUARY 23, 1993 THROUGH DECEMBER 31, 1993

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S),SECTION(S), OR FIGURE(S) REVISED
PC 41271 PC 41272 PC 41273 EA-FC-93-027	Description: Setpoint changes were implemented for the 480V load center bus undervoltage relays, the transformer secondary undervoltage relays, and the 4160V bus undervoltage relays (PCs 41271, 41272 and 41273, respectively) to better conform to the vendor's recommended calibration range. The relays' design basis functions continue to be met with the revised setpoints. This blanket procedure change revised the IAV Relay dropout time setpoint and the calibration procedure time tolerances in the following procedures: SP-CP-08-480-1B3A, -1B3A-4A, -1B3B, -1B3B-4B, -1B3C, -1B3C-4C, -1B4A, -1B4B and -1B4C; SP-CP-08-161-IAV, -345-IAV; SP-CP-08-1A1-IAV, -1A2-IAV. An additional dropout time check point and associated tolerance were added to the calibration procedures to check loss of voltage time delay. Safety Analysis: The new IAV setpoints meet the required design basis functions. The relay logic and design basis function are not changed. A change to the setpoints does not reduce the Technical Specification basis margins. Loss of voltage loadshed and degraded voltage protection are provided. The probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	Section 8.4.3.2
Turbine Generator Control System	Description: This revision removed unnecessarily detailed information concerning the accuracies, repeatabilities and linearities of the turbine generator Electohydraulic Control System. Safety Analysis: The turbine generator control system has no impact on any of the Technical Specifications bases. The loss of load accident and the turbine overspeed accident are the only credible accidents that the turbine generator control system could be involved in. Both of the design basis analyses for these accidents already assume that the turbine generator control system fails. Therefore, this USAR revision does not increase the probability or consequences of any new or previously analyzed accident.	Section 7.4.6.2

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT COMMISSION APPROVAL

SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), OR FIGURE(S) REVISED
MR-FC-89-074	Description: This change revised Table 14.24-1, "Load/Drop Table," of the USAR. Specifically, reference to the electrical supply to the packing cooling pump for Charging Pump CH-1A was removed because the packing cooling pump motor is no longer classified as CQE. Therefore, the need for consideration for a load drop on the power supply to the packing cooling pump motor no longer needs to be considered. Safety Analysis: The safety evaluation concludes that removal of the packing cooling pump motor power supply cables from Table 14.24-1 is appropriate since the CQE classification for the motors has been changed in accordance with Modification MR-FC-89-074. The probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	Table 14.24-1
ECN-93-074	Description: Information pertaining to the type of filter element and filtration rating for Letdown Purification Filters CH-17A and CH-17B was deleted from Table 9.2-7 of the USAR. Safety Analysis: This revision is administrative in nature in that it removes information that is not pertinent to the design or operation of the CH-17A/B filtration units. The probability/consequences of any previously analyzed accident is not increased, nor is the possibility for a previously unanalyzed accident created.	Section 9.2.3.5

Attachment B

JANUARY 23, 1993 THROUGH DECEMBER 31, 1993

USAR CHANGES OTHER THAN THOSE RESULTING FROM 10 CFR 50.59

DESCRIPTION	USAR PAGES AFFECTED
This update incorporates reference to the alternate seismic criteria and methodologies (ASCM) as an alternate seismic design basis into Appendix F based on the NRC's Safety Evaluation Report (SER).	Appendix F
Drawing change to reflect roof replacement on intake structure.	Figure 1.2-9
Drawing change to reflect MR-FC-92-031 which installed dedicated test transformers for Inverters 1 and 2.	Figure 8.1-1
Drawing change to add valves SI-9 and SI-10 to drawing to reflect current configuration.	Figure 5.9-13, Sheets 1&2
The USAR change resulting from ECN 93-506 corrects a typographical error in Table 9.8-1 and eliminates the need for additional USAR updates should future RW pump impeller material upgrades be approved through OPPD's internal procedures.	Page 9.8-2
[illegible] what is a "normal" load, the USAR was changed to delete a statement concerning variations from normal loads which [illegible]ally stop the motion of the refueling machine hoist winch mechanism. Although reported on the 1993 50.59 report, the USAR [illegible] as not identified at that time.	Section 9.5.4.3
EC[illegible]3-383 (drawing change) incorporates the addition of a 1/2" test tee between Isolation Valve HCV-1559A and the containment penetration. The configuration was verified by a system walkdown during the 1993 Refueling Outage.	Figure 5.9-13, Sheet 51
EA-FC-93-044 revised Section 11, "Radioactive Waste and Radiation Protection and Monitoring," of the USAR to reflect changes due to updating the source term, annual average relative concentration and new 10 CFR Part 20 criteria.	Section 11
EA-FC-93-049 revised USAR Tables 3.4-13 and 3.4-14, "Comparison of Predicted and Measured Power Coefficients," to add the measured values taken at the end of Cycle 14 and it corrected a typographical error.	Tables 3.4-13 & 3.4-14
ABB/CE Study O-MECH-92-089 recommended clarifications to Section 4 of the USAR relating to transient cycle counting.	Sections 4.2.2, 4.2.4
The USAR was revised to clarify that the CCW operating temperature range is a description of the normal operating range of the system and not a definition of design limits.	Sections 9.7.1, 9.7.3, 9.7.4.1, 9.7.5
Table 9.2-10 of the USAR was changed to indicate that ambient is the nominal operating temperature for Tank CH-15.	Table 9.2-10
USAR Section 8 was revised to state that Fort Calhoun Station is a four hour DC dependent plant reflecting a NRC SER concerning 10 CFR 50.63 (Station Blackout Rule).	Section 8.1.1
ECN 93-634 revised Figure 8.2-1 to reflect modifications made to Substation 1251 during the Fall of 1993.	Figure 8.2-1
The Quality Assurance (QA) Program contained in Appendix A was revised to reflect organizational and other administrative changes.	Appendix A

JANUARY 23, 1993 THROUGH DECEMBER 31, 1993

USAR CHANGES OTHER THAN THOSE RESULTING FROM 10 CFR 50.59

DESCRIPTION	USAR PAGES AFFECTED
EA-FC-93-011, 014, 016, 018, 019 and 020 incorporate Cycle 15 changes resulting from Amendment 157 to the Technical Specifications and NRC SERs on OPPD Topical Reports.	Sections 3, 4, 14
USAR Section 9.11 was revised to include former Technical Specifications 2.19, 3.15 and associated bases and interpretations. The USAR markup was submitted to the NRC as part of Facility License Change (FLC) 93-006 as required by Generic Letter 88-12. In addition, OPPD's USAR Verification Project is adding a list of references.	Section 9.11
Drawing change ECN-94-020 was initiated to correct a typographical error.	Figure 5.9-13, Sheet 44
Drawing change ECN-93-359 was issued to incorporate the addition of a test tee between Valve CH-198 and the containment penetration.	Figure 5.9-13, Sheet 04
USAR Section 12 was revised to reflect the current organizational charts, current fuel vendor and to correct typographical errors and an ambiguous statement in Section 12.3.2.	Section 12
USAR sections were revised to correct inaccurate information and to incorporate additional information due to the USAR Verification Project.	Sections 3, 4, 9 and 14
EA-FC-91-008 revised USAR Section 14.2 to show that the control element assembly withdrawal incident was reanalyzed for Cycle 14, reference the TORC computer code, reference COLR Figure 4, and revise the variable high power trip and moderator temperature coefficient assumptions.	Section 14.2

Attachment C

REMOVE	INSERT
PAGE STATUS	PAGE STATUS
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Fig 1.2-9	Fig 1.2-9
2.10-1	2.10-1
2.10-5	2.10-5
2.10-5a	2.10-5a
Fig 2.10-2	Fig 2.10-2
Fig 2.10-4 Sh 1	Fig 2.10-4 Sh 1
Fig 2.10-4 Sh 2	Fig 2.10-4 Sh 2
Fig 2.10-4 Sh 3	----------------
Fig 2.10-4 Sh 4	----------------
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3.4-11	3.4-11
3.4-12	3.4-12
3.4-14	3.4-14
3.4-15	3.4-15

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3.5-6	3.5-6
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3.6-8	3.6-8
3.7-2	3.7-2
3.7-3	3.7-3
3.7-4	3.7-4
3.7-10	3.7-10
3.7-12	3.7-12
3.7-13	3.7-13

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3.7-16	3.7-16
3.7-18	3.7-18
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Fig 3.4-1	Fig 3.4-1
Fig 3.4-2	Fig 3.4-2
Fig 3.4-4	Fig 3.4-4
Fig 3.4-5	Fig 3.4-5
Fig 3.4-6	Fig 3.4-6
Fig 3.4-7	Fig 3.4-7
Fig 3.4-8	---------
Fig 3.6-1	---------
Fig 3.6-2	---------
Fig 3.7-3	---------
Fig 3.7-4	---------
Fig 3.7-5	---------
Fig 3.7-6	---------
Fig 3.7-8	---------
4-i	4-i

REMOVE	INSERT
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4-iv	4-iv
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4.7-1	4.7-1
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Fig 4.3-1	Fig 4.3-1
Fig 4.3-3	Fig 4.3-3
Fig 4.3-6	Fig 4.3-6
Fig 4.3-9	Fig 4.3-9
Fig 4.5-5	---------
Fig [illegible]	Fig 4.5-6
Fig 4.6-[illegible]	---------
5.9-4a	5.9-4a
5.9-5	5.9-5
5.9-6	5.9-6
5.9-6a	5.9-6a
5.9-7	5.9-7
Fig 5.9-13 Sh 01	Fig 5.9-13 Sh 01
Fig 5.9-13 Sh 02	Fig 5.9-13 Sh 02
Fig 5.9-13 Sh 03	Fig 5.9-13 Sh 03
Fig 5.9-13 Sh 04	Fig 5.9-13 Sh 04
Fig 5.9-13 Sh 05	Fig 5.9-13 Sh 05
Fig 5.9-13 Sh 06	Fig 5.9-13 Sh 06
Fig 5.9-13 Sh 15	Fig 5.9-13 Sh 15
Fig 5.9-13 Sh 16	Fig 5.9-13 Sh 16
Fig 5.9-13 Sh 20	Fig 5.9-13 Sh 20
Fig 5.9-13 Sh 32	Fig 5.9-13 Sh 32
Fig 5.9-13 Sh 33	Fig 5.9-13 Sh 33
Fig 5.9-13 Sh 34	Fig 5.9-13 Sh 34
Fig 5.9-13 Sh 44	Fig 5.9-13 Sh 44
Fig 5.9-13 Sh 45	Fig 5.9-13 Sh 45
Fig 5.9-13 Sh 47	Fig 5.9-13 Sh 47
Fig 5.9-13 Sh 48	Fig 5.9-13 Sh 48
Fig 5.9-13 Sh 49	Fig 5.9-13 Sh 49
Fig 5.9-13 Sh 51	Fig 5.9-13 Sh 51
Fig 5.9-13 Sh 52	Fig 5.9-13 Sh 52
Fig 5.9-13 Sh 53	Fig 5.9-13 Sh 53
Fig 5.9-13 Sh 54	Fig 5.9-13 Sh 54
Fig 5.9-13 Sh 55	Fig 5.9-13 Sh 55
Fig 5.9-13 Sh 56	Fig 5.9-13 Sh 56
Fig 5.9-13 Sh 57	Fig 5.9-13 Sh 57
Fig 5.9-13 Sh 58	Fig 5.9-13 Sh 58
Fig 5.9-13 Sh 59	Fig 5.9-13 Sh 59
Fig 5.9-13 Sh 60	Fig 5.9-13 Sh 60
Fig 5.9-13 Sh 62	Fig 5.9-13 Sh 62
Fig 5.9-13 Sh 63	Fig 5.9-13 Sh 63
Fig 5.9-13 Sh 65	Fig 5.9-13 Sh 65
Fig 5.9-13 Sh 66	Fig 5.9-13 Sh 66
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7.6-9	7.6-9
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8.3-8	8.3-8
8.4-8	8.4-8
Fig 8.1-1	Fig 8.1-1
Fig 8.2-1	Fig 8.2-1
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Fig 9.2-2	----------
Fig 9.3-1	Fig 9.3-1
Fig 9.13-1	Fig 9.13-1
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Fig 12.1-1	Fig 12.1-1
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Fig 12.1-3	Fig 12.1-3
Fig 12.1-4	Fig 12.1-4
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NOTE: SOME PAGES IN SECTION 14 HAVE NOT CHANGED BUT ARE INCLUDED TO SIMPLIFY PAGE REPLACEMENT.

REMOVE	INSERT
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Section 14.2 ALL PAGES	Section 14.2 ALL PAGES
Section 14.3 ALL PAGES	Section 14.3 ALL PAGES
Section 14.4 ALL PAGES	Section 14.4 ALL PAGES
Section 14.6 ALL PAGES	Section 14.6 ALL PAGES
Section 14.9 ALL PAGES	Section 14.9 ALL PAGES
Section 14.10 ALL PAGES	Section 14.10 ALL PAGES
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