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Omaha, Nebraska 68102-2247
402/636-2000

October 12, 1995
LIC-95-0179

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), 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). Pursuant to 10 CFR 50.71(e) and 10 CFR 50.4(b)(6), Attachment C provides one original set of inserts and 10 copies of the USAR update 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 1, 1994 through April 30, 1995.

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)
L. J. Callan, NRC Regional Administrator, Region IV (Copy #13)
S. D. Bloom, NRC Project Manager (Copy #12)
W. C. Walker, 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 (Fort Calhoun Station Unit No. 1)	)))))	Docket No. 50-285

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 Oct 12, day of October, 1995.

Steve R. Dixon

Notary Public

GENERAL NOTARY-State of Nebraska
STEVE R. DIXON
My Comm. Exp. April 4, 1999

Attachment A
LIC-95-0179

10 CFR 50.59 REPORT
JANUARY 1, 1994 THROUGH APRIL 30, 1995

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-90-054	Description: This modification changed the gearing, spring pack and limiter plate in the operators of motor operated feedwater isolation valves HCV-1103, HCV-1104, HCV-1385 and HCV-1386 to allow increasing the valve stroke time from 10 seconds to 40 seconds. On all four valves, the limit switch circuits were modified to limit out on torque rather than position and the motor brakes were removed since they are not required when shutting on torque at the reduced speed. Motors on Valves HCV-1103 and HCV-1104 were replaced with motors with different torque-current curves to assure sufficient torque under degraded voltage conditions. Safety Analysis: Although increasing the feedwater isolation valve stroke time results in increased blowdown to containment (increasing containment pressure slightly), containment pressure is still within the acceptance criteria for a main steam line break event. Therefore, the modification did not increase the probability of occurrence or consequences of an accident previously evaluated in the USAR. As only the four valves listed above are affected by this modification and they are placed in a more reliable configuration, neither the possibility of a malfunction of equipment important to safety is created nor are the consequences of a malfunction of equipment important to safety increased. The modification did not alter the feedwater systems interaction with other systems and therefore, does not increase the probability of occurrence or consequences of an accident of a different type than any previously evaluated in the USAR. The peak containment pressure is below the acceptance limit, therefore, the change did not reduce a margin of safety.	None
MR-FC-89-032 FDCR-93-0421	Description: This modification switched the DC power supplies on the Shutdown Cooling Heat Exchangers (SDHX) Component Cooling Water (CCW) inlet valves to ensure that only one CCW isolation valve on each SDHX will fail open due to a failure of one DC bus. Two backup nitrogen supply systems were also provided to the four CCW valves to prevent any of the four from failing open due to a loss of instrument air pressure. This modification was completed to meet a commitment made in LER-90-025. Safety Analysis: This modification was performed to maintain adequate CCW flow to other loads if Instrument Air or DC power is lost pre-Recirculation Actuation Signal (RAS). The RAS will continue to function as designed, therefore the change did not increase the probability or consequences of a accident previously evaluated. This modification did not reduce the seismic requirements, single failure design, electrical separation, or environmental qualification of the SDHX CCW valves. No new failure modes or different response to automatic actions were produced as a result of this change. Therefore, the change did not create the possibility of an different type of accident than previously evaluated. The response to a RAS is as originally designed, therefore the margin of safety is not reduced.	Section 9.12
Procedure Change (PC) 36668	Description: This procedure change disables the auto closure interlocks associated with shutdown cooling system isolation valves HCV-347 and HCV-348 after the pressurizer manway has been removed in preparation for refueling. Safety Analysis: The interlock is a single purpose feature that addresses the concern of the RCS pressure exceeding the design limits of the shutdown cooling system. With the pressurizer manway removed the RCS cannot be pressurized, therefore the change will not increase the probability or consequences of an accident previously evaluated nor is a situation created to create a new or different kind of accident. Disabling the auto closure interlocks does not reduce the margin of safety because a pressure greater than 225-250 psig (well within design limits) cannot be established in the reactor coolant system.	Section 9.3.2

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SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
Memo PED-FC-94-0423	Description: The USAR is being revised to include a description of the limited conditions under which raw water (RW) direct cooling may be used for the containment air cooling coils; i.e. if containment atmosphere is less than 150°F. Safety Analysis: The nuclear safety function of the containment air cooling coils is unaffected by this change, since the containment atmospheric temperature limitation on the use of RW direct cooling effectively excludes its use after a major design basis accident such as a loss of coolant accident (LOCA) or main steam line break (MSLB).	Sections 6.4 and 9.8
PC 43071	Description: This procedure change specifies that although it must be operable, control room ventilation does not need to be operating in filtered makeup mode before irradiated fuel movement commences in containment or in the spent fuel pool area. Safety Analysis: Engineering Analysis EA-FC-90-094 determined that the dose consequences associated with a fuel handling accident in containment or the spent fuel pool area are within 10 CFR 100.11 requirements. Therefore, control room ventilation is not required to be in filtered makeup mode prior to irradiated fuel handling operations in containment or the spent fuel pool area.	Section 9.5.1.5
MR-FC-88-046	Description: The HCV-400 series A/B/D valves that provide component cooling water (CCW) isolation to the containment air coolers were equipped with backup nitrogen (not air) accumulators to enable these air-operated valves to be remotely repositioned in a post-accident situation without instrument air available. This will help Operators to optimize CCW system flow distribution in the long term after an accident. Safety Analysis: The post-accident performance of the containment air coolers is not adversely affected by this modification. The addition of N_2 backup only allows the valves to be remotely operated without instrument air; therefore, the change did not increase the probability or consequences of an accident previously evaluated. The addition did not introduce any new accident initiators and therefore did not create the possibility of a new or different kind of accident. The post accident performance of the containment air coolers will not be adversely affected by this change, therefore the change did not reduce a margin of safety.	Section 9.12 Figure 5.9-13 Sheet 46

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SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
MR-FC-91-036 EA-FC-92-81	Description: Offsite power low signal setpoints were changed, selected non-safety related loads will be tripped by safety injection actuation signal (SIAS) and containment spray actuation signal (CSAS), 480 V loads are removed or added to the SIAS or CSAS load shed schemes and 480 V loads are redistributed to eliminate bus or transformer overloading to accommodate the operation of main feedwater and condensate pumps after SIAS or operation of a condensate pump after CSAS. Safety Analysis: The consequences of tripping these loads are bounded by the current USAR safety analysis. The design of the OPLS and safety margin are unchanged. The 5 kV and 480 V loads tripped are non-safety related and do not contribute to any Technical Specification margin of safety. The probability of previously evaluated accidents occurring is not increased nor are any new accident scenarios created. Also, failure of the affected equipment does not affect safety equipment in an adverse manner.	Figure 8.1-1
MR-FC-93-021	Description: This modification added ground detection to the control element drive mechanism (CEDM) power supply and to the rod drive position mimic power supply. Indication and annunciation is provided to alert operators of a ground on either system. Safety Analysis: This modification does not change the operation of the CEDM system or rod drive position mimic power supply. Therefore, neither the probability of occurrence of an accident previously evaluated in the USAR nor its consequences are increased by this modification. Similarly, neither the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR nor its consequences are increased by this modification. No new failure modes were postulated as a result of this modification, therefore the change did not create the possibility of a new or different kind of accident. The reactor protective system (RPS) will trip the reactor for any previously analyzed event and is unaffected by this modification, therefore no margin of safety was reduced.	None
MR-FC-93-018	Description: This modification provided for resolution of items found during the SQUG/IPEEE walkdown which had the potential of damaging safety related equipment during a seismic event. Two work tables, a cabinet, a line printer, and a cubicle wall were anchored and a "pulled out anchor" on a rod hanger support were replaced under this modification. Safety Analysis: The anchoring of these items prevents adverse seismic II/I impact/pipe loading concerns. Therefore, the probability or consequences of previously evaluated accidents occurring is not increased, nor was any margin of safety reduced. No new failure modes were introduced due to this change, therefore no new or different kind of accident was created.	None

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ECN-93-481	Description: ECN-93-481 provides for disconnecting Cable ED3003 during normal operations. Cable ED3003, which is a test signal cable that provides predetermined power level calibration signals to NT-004 will be connected only for testing purposes. Safety Analysis: Disconnecting Cable ED3003, ensures that an induced signal will not be received at the Alternate Shutdown Panel if there is a fire in the Control Room. Therefore the change did not increase the probability or consequences of an accident previously evaluated. With the cable disconnected it will not provide any function except during testing, therefore the change did not create the possibility of a new or different kind of accident nor did it reduce any margin of safety	Figure 7.2-7
TM-94-014 TM-94-015	Description: These temporary modifications (TM) isolated the supervisory relay from the control matrix for 86A/STLS and 86B/STLS. This eliminated the supervisory alarm while maintaining circuit integrity indication. Safety Analysis: The Supervisory circuit provides alarm and indication only. Therefore the temporary removal of the circuit could not induce a failure into any plant system which could cause an event or prevent the STLS from actuating. Therefore the change did not increase the probability or consequences of an accident previously evaluated. The STLS circuits do not require alarmed supervision to ensure its design function and has no other interaction with plant equipment; therefore the change did not create the possibility of a new or different type of accident. Supervisory circuits for the STLS are not required for operability of the STLS and therefore the change did not reduce any margin of safety.	None
TM-94-033	Description: A temporary blank spacer was installed in place of HCV-400E to allow VA-1A to be returned to service while the valve was rebuilt. (A blank spacer is essentially a blind flange but the same thickness as the existing valve body. Safety Analysis: The blank spacer did not affect operation of either the raw water (RW) or component cooling water (CCW) systems or any related component from fulfilling its required safety function. The consequences of a failure of a blank spacer is the same as a failure of the locked closed valve, which will result in the loss of CCW inventory. The loss of CCW has been evaluated in the USAR and procedures exist to address its loss. Therefore, the change did not increase the probability or consequences of an accident previously evaluated. The change did not affect systems other than RW and CCW and therefore did not create the possibility of a new or different kind of accident nor did the change 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
ECN-95-0130	Description:	This ECN lowered the setpoints for Component Cooling Water (CCW) relief valves PCV-2839, AC-341 and AC-364. These valves provide overpressure protection for CCW surge tank AC-2. It was postulated that during a design basis accident (DBA), the resultant thermal expansion of CCW would increase the surge tank Nitrogen pressure to the point that CCW pump discharge pressure exceeds the setpoints of several downstream relief valves when three CCW pumps are operating. Lowering the setpoints on the AC-2 relief valves will minimize the amount of CCW inventory lost during a DBA by preventing the thermal relief valves on the downstream CCW components from lifting.	Section 9.7
	Safety Analysis:	The ECN did not change overall system performance or design. Material and construction standards are also unchanged. Therefore, the probability of occurrence of an accident previously evaluated in the USAR is not increased. The ECN provides greater assurance of preventing CCW inventory loss without affecting CCW system pump requirements during a DBA. Thus, the consequences of an accident previously evaluated in the USAR are not increased. Redundancy and independence are maintained and no increase in operating parameters is proposed. Thus neither the probability of occurrence of an equipment malfunction nor the consequences of an equipment malfunction are increased. The equipment still functions in the same manner as before; therefore, the ECN did not create the possibility of a different type of accident, nor did it reduce any margin of safety.	
SAO-95-01 EA-FC-95-012	Description:	Safety Analysis for Operability (SAO) No. 95-01 and Engineering Analysis (EA) No. 95-012 document the operability of the CCW system and safety-related equipment cooled by CCW with regard to safety functions after a large break LOCA or main steam line break (MSLB) inside containment. The SAO contains the necessary operational provisions to ensure this operability.	To be incorporated after receipt of License Amendment.
	Safety Analysis:	SAO-95-01/EA-95-012 ensure that credited nuclear safety functions related to the RW/CCW systems will be met after a large break LOCA or MSLB inside containment. The operational conditions in the SAO/EA do not place the CCW or RW systems in configurations for which they were not designed. The provisions in the SAO/EA are more restrictive that current Technical Specifications to cover the interim period until the Technical Specifications are amended. The SAO/EA provisions do not increase accident likelihood or consequences, equipment malfunction likelihood or consequences; nor do they create an initiator for a new type of accident or equipment malfunction. The additional restrictions on the systems ensures that current margins of safety are maintained.	

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MR-FC-93-008	Description:	A travel limit bypass switch was added to the spent fuel handling machine (FH-12) to allow it to travel beyond its stop limits. This was necessary because the reracking of spent fuel in the spent fuel pool provides for fuel rack locations that are outside of the current travel limits of FH-12. When FH-12 is operated beyond its normal limits, the maximum travel speed is reduced to allow the operator more response time and prevent inadvertent contact of a fuel bundle with the spent fuel pool wall. The stop limits for south and east travel were changed to reduce the amount of time that FH-12 has to operate at reduced speed. The new stop limit minimizes the possibility of inadvertently resetting the bypass circuit by operating the switch as it nears the inner wall of the outer fuel cell location.	Page 9.5-12
	Safety Analysis:	This modification provides the operator with visual indication and automatic speed reduction when a FH-12 limit is exceeded. These safety features allow the operation of FH-12 outside its normal limits to reach additional spent fuel locations provided by reracking. The modification does not increase accident likelihood or consequences, equipment malfunction likelihood or consequences nor does it create an initiator for a new type of accident or equipment malfunction.	
EA-95-13 R1	Description:	USAR discussion of a 120 VAC Instrument Inverter design feature was revised. The discussion concerned the inverter reverting to an internal frequency reference if the synchronous source reference frequency deviates beyond a predetermined limit. The discussion was revised because the installed inverter configuration does not have the frequency based transfer feature. The inverters actually revert to an internal frequency reference on loss of synchronous source voltage.	Sections 7.3.5.4, 8.3.5.1, 8.3.5.3
	Safety Analysis:	The EA shows that the equipment will operate as designed. There is no increase in accident likelihood or consequences, nor in equipment malfunction likelihood or consequences. This change also does not create an initiator for a new type of accident or equipment malfunction.	
PC 43005	Description:	Procedure CH-AD-0003, "Plant Systems Chemical Limits and Corrective Actions," was changed to incorporate the use of Ethanolamine (ETA) as a secondary chemistry pH control additive. The FCS Chemistry Department determined that ETA is effective as a pH control additive in reducing the transport of corrosion products to the steam generators.	Page 9.13-1
	Safety Analysis:	ETA performs the same reduction in corrosion rates as Morpholine and does not increase the likelihood of a tube or pipe rupture, nor does it affect sample system equipment, valves, gaskets and seals or the turbine any differently than Morpholine. Thus, there is no increase in accident likelihood or consequences, nor any reduction in a margin of safety. This change also did not create an initiator for a new type of accident or equipment malfunction.	

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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 43743	Description: The receipt of Technical Specification Amendment No. 164 required changes in the Offsite Dose Calculation Manual (ODCM) and the USAR. Safety Analysis: The USAR changes 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 for Amendment 164	Section 11
PC 42534A	Description: The "Normal" valve position shown in tables on USAR Figure 5.9-13 was deleted since the normal valve position is dependent upon plant operating mode. The "normal" position is maintained by Plant Review Committee and/or qualified reviewer approved administrative controls. Safety Analysis: The valve positions are plant mode dependent and are controlled by PRC and/or qualified reviewer approved controls. The USAR Figures still state the failed and accident position of valves and what signals reposition the valves. This change did not affect the valves in any way and therefore, the margin of safety is not reduced, the probability and consequences of previously analyzed accidents are not increased and the possibility of any previously unanalyzed accident is not created.	Figure 5.9-13 (numerous sheets)
MEMO PED-FC-94-1172	Description: Sections 9.3.1 and 9.7.4.2 of the USAR are being revised to exclude statements about the ability to cool down the RCS from 300°F to refueling temperature in a specific time frame. This will avoid possible misinterpretations of these sections in the future. Safety Analysis: There are no nuclear safety functions requiring the RCS to be cooled to refueling temperature in a specific time frame, therefore this change did not impact the probability or consequences of an accident previously evaluated. No changes to equipment or operating procedures were made due to this change, therefore the change did not create the possibility of a new or different kind of accident, nor did the change reduce any margin of safety.	Pages 9.3-1 and 9.7-5

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
EA-FC-95-001	Description: EA-FC-95-001 revised an assumption made in Engineering Study 81-004 that was incorrect. The incorrect assumption indicated that a heavy load drop of a circulating pump motor over the raw water pump bays would result in a raw water header break that could be isolated by closing remote actuated valves in the header. Further investigation revealed that the load drop would also sever the instrument air lines supplying the isolation valves. Manual operation of the isolation valves could not be accomplished before the raw water pump bays are flooded due to raw water leakage into that area. Safety Analysis: The probability of a heavy load drop accident damaging safety related equipment in the intake structure is reduced because the allowable load limit is reduced. The consequences of a heavy load drop in the intake structure are reduced since the administrative limits will decrease damage to the structure and/or equipment. The analysis and administrative limits assure that the raw water system is maintained in a condition that will allow it to perform its safety function in the event of a heavy load drop. Therefore, the change decreases the probability and consequences of accidents previously evaluated. No additional failure modes were created by the limitations on handling heavy loads in the intake structure and therefore, the change did not create the possibility of a new or different kind of accident. These additional restrictions did not reduce any margin of safety.	Pages 14.24-3 and 14.24-9
MR-FC-93-022	Description: Valves HCV-746A and B are containment relief valves that were replaced to ensure a leak rate of less than 600 SCCM per ASME Section XI. Valves HCV-746A and B were originally Fisher Model 667-A globe valves that were replaced by Anchor/Darling Model 398-20-003 ball valves. Bettis air operators replaced the original Fisher model 667-40 actuators. Safety Analysis: This activity did not increase the probability of occurrence or consequences of an accident or equipment malfunction previously evaluated in the USAR. Valve control remains the same and the valve design, material and construction standards are equal to the originally installed globe valves. Seismic issues are not a concern since the replacement valves are of equal or lighter weight than the original valves. The new valves provide a tighter shut-off than the original valves, thereby meeting the Type C leakage requirements. Technical Specification and system operability requirements are still the same thus, this modification did not create the probability of a different type of accident or equipment malfunction than any previously evaluated in the USAR.	Figure 5.9-13 Sheet 34

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-93-002	Description: To override and eliminate momentary CCW system pressure fluctuation below the setpoint of 60 psi due to CIAS actuation, two time on-delay relays were added. One was placed in the HCV-438A/C control circuit logic and the other was added in the HCV-438B/D control circuit logic. Each relay was connected in series with the respective CCW pumps discharge header pressure switch (PCS-412 for HCV-438A/C & PCS-413 fro HCV-438B/D). The normally open output pressure contact of each relay was connected in series with the respective CIAS relay (742A or 742B) normally closed contact. Safety Analysis: This modification only added time delay on closure of containment isolation valves HCV-438A,B,C,D. The fail safe position of these valves is open. Thus this modification does not increase the probability of accident occurrence. Valve functioning is not affected by this modification and the additional time delay is still within the stroke time evaluated in the safety analysis. Thus, the consequences of an accident are not increased nor is the possibility of an unanalyzed accident created. The addition of the time delay to the valves control circuits eliminates valve closure due to momentary CCW pressure fluctuation below the setpoint. CCW flow to the RCP seal coolers is enhanced which increases RC pump reliability. Thus, the modification does not increase the probability of occurrence nor the consequences of a malfunction of equipment important to safety.	Page 5.9-8
EA-FC-94-016 through EA-FC-94-026 and EA-FC-95-019	Description: USAR Sections 3, 4, and 14 were updated to incorporate the neutronic, transient, non-transient and setpoint analyses performed for the Cycle 16 reload. The changes reflect NRC approved Cycle 16 methodology. Safety Analysis: Changing the USAR to incorporate Cycle 16 values does not affect any systems that could increase the probability of occurrence or consequences of a previously evaluated accident. Nor do the USAR changes increase the probability of occurrence or consequences of a malfunction of equipment important to safety. No new modes of operation are proposed. Therefore, the USAR revisions do not create the possibility of a different type of accident or malfunction of equipment important to safety.	Sections 3, 4, and 14
USAR Section 9.11	Description: USAR Sections 9.11 was revised to account for strainer pressure drop which was previously unaccounted for. The fire water supply pump minimum operability requirements at 1800 gpm was 260 ft of head and must be raised to 280 ft of head. Safety Analysis: The function of the fire suppression system is not an initiator of any accident, therefore this changes did not increase the probability of an accident previously analyzed. The change did not change the operation of the system, the pumps will still perform within the normal design capabilites, therefore the change did not increase the consequences of an accident previously analyzed. The operation of the fire protection system will not change as a result of this change, therefore no new or different kind of accident was created and no margin of safety was reduced.	Section 9.11

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SOURCE	DESCRIPTION/SUMMARY OF SAFETY ANALYSIS	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
MR-FC-91-009 Amendment 155	Description: The spent fuel pool storage racks were replaced by maximum density storage racks to increase the onsite spent fuel storage capacity from 729 locations to 1083 locations. The rerack extends the onsite full core discharge capability through the year 2007 based on current refueling projections. A field design change revised the rack-to-wall clearances of the original modification. Safety Analysis: Potential accident scenarios including heavy load handling accidents, seismic events, and loss of spent fuel pool cooling were evaluated for this modification. Neither the probability of occurrence nor consequences of these accidents were increased by this modification nor was the possibility of a new type of accident created. Similarly, neither the probability of occurrence nor consequences of an equipment malfunction were increased by this modification nor was the possibility of a new type of equipment malfunction created. The modification was approved by the NRC by Amendment 155. Changes in rack-to-wall clearances were submitted to the NRC in a letter dated March 20, 1995 (LIC-95-0065).	Sections 1, 5, 9, and 14

Attachment B
LIC-95-0179

USAR CHANGES OTHER THAN THOSE RESULTING FROM 10 CFR 50.59

DESCRIPTION	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
USAR Appendix A changes. The OPPD QA Program was revised to reflect NRC approval of the reduction of SARC audits as documented in a letter from the NRC (T. P. Gwynn) to OPPD (T. L. Patterson) dated August 16, 1995, and administrative/organizational changes.	Appendix A
Amendment 162 demonstrates that the current fuel oil system configuration meets the fuel oil storage capacity requirements of IEEE-308 and is capable of providing fuel oil for 7 days of continuous emergency diesel generator (EDG) operation following the most limiting accident.	Section 8.4.1
Amendment 167 relocated the requirements of TS 2.10.3 and the peak linear heat rate uncertainty factors stated in TS 2.10.4 to the USAR.	Section 7.5.4.3
Amendment 166 clarified what equipment requires raw water backup in the event of a LOCA and loss of CCW and also what equipment is credited with RW backup in the fire safe shutdown analysis.	Pages 9.7-6 & 9.8-1
The USAR was revised following receipt of a NRC Safety Evaluation Report (SER) dated February 17, 1994. The SER approved revisions and clarifications to the licensing bases for the post accident sampling system.	Section 9.13
Amendment 169 relocated the requirements for the Containment Building and Auxiliary Building overhead cranes to the USAR.	Sections 9.5, 14.18, and 14.24
Amendment 168 revised PRC and SARC composition. Figures were revised to reflect changes in titles and/or reporting responsibilities.	Section 12
A reference was added to page 4.7-2 to reflect receipt of the SER on the Inservice Testing Program Third Ten-Year Interval for inservice testing of pumps and valves.	Page 4.7-2
USAR Section 14.24 was revised to address containment closure requirements for heavy load lifts over irradiated fuel with the reactor vessel head removed. This information is being added in response to LER-95-002.	Section 14.24
Engineering Analysis EA-FC-93-089, CPTP #38, Beginning of Cycle 15 At-Power Moderator Temperature Coefficient Test was completed.	Table 3.4-13, Table 3.4-14
Engineering Analysis EA-FC-94-043, CPTP #39, Beginning of Cycle 16 At-Power Moderator Temperature Coefficient Test was completed.	Table 3.4-13, Table 3.4-14
Memorandum PED-FC-94-0883 USAR clarifications pertaining to the different requirements for light steel deflector plates and cable tray covers.	Page 8.5-2, Appendix M, Figure 8.5-1 & Figure 8.5-3
Engineering Analysis EA-FC-93-082 performed a bounding analysis for the steam generator tube rupture incident for Fort Calhoun Station.	Section 14.14.2, Section 14.14.4
MR-FC-92-015 made the final control tie-ins for the new 161 KV line and it's associated breakers and provided for the alarm of the sub-station gate.	Figure 8.1-1
ECN-93-214 removed and capped a 10" exhaust duct which served the steam generator blowdown processing system in Room 20 of the Auxiliary Building.	Section 9.10.4.1
Incident Report 940223 resulted in a correction to USAR Section 7.3.4.3 concerning when safety related motor operated valves give an alarm condition in the control room. The modification (MR-FC-86-091) that physically made these changes was reported in the 1991 50.59 report.	Section 7.3.4.3
PC 43153 to OI-RC-8 modified figures to more accurately reflect the RCS level control program that is installed.	Figures 4.3-10 and 7.4-4

USAR CHANGES OTHER THAN THOSE RESULTING FROM 10 CFR 50.59

DESCRIPTION	USAR PAGE(S), SECTION(S), TABLE(S) OR FIGURE(S) REVISED
ECN-94-242 determined that the spent fuel pool cooling emergency cross tie piping should be NNS CL-1. EAR-94-005 determined that a limited portion of the fire protection system that serves the control room charcoal adsorbers is safety class 3.	Pages N.5-3 & N.5-7
USAR revised to show that bodies of main steam safety valves are constructed from A-105 Grade II carbon steel.	Page 4.3-8
ECN-92-372 changed USAP Figure 7.6-1, Control Room Par	Figure 7.6-1
USAR Sections 6.4.2 & 6.4.4 and Tables 6.4-1 & 6.4-5 were revised to incorporate clarifications and remove erroneous or unsupported statements.	Sections 6.4.2 & 6.4.4 Tables 6.4-1 & 6.4-5
Drawing change resulting from ECN-92-197	Figure 5.9-4
ECN 94-521 is superseding Drawing File #36546 with File #1582	Figure 7.2-2
USAR Section 9.2.3.7 was revised to strike the cooling water requirement for the charging pump's lube oil coolers. The requirement is for extending the life of the pump and is not necessary to maintain the pump during a design basis accident.	Section 9.2.3.7
USAR Section 9.4 was revised to clarify the use of demineralized water and condensate as makeup sources to the emergency feedwater storage tank.	Section 9.4
USAR Section 9.4-1 was revised as a result of calculation FC-06148, Rev 1 to make the USAR consistent with the methodology used in the LOCA analysis approved by the NRC.	Section 9.4-1

Attachment C
LIC-95-0179