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Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

October 5, 2001
LIC-01-0076

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Reference: 1. Docket No. 50-285
2. NEI 99-04, "Guidelines for Managing NRC Commitments"

SUBJECT: 10 CFR 50.59 Report and Updated Safety Analysis Report (USAR) Revision for Fort Calhoun Station

As required by 10 CFR 50.59(d)(2), Attachment A is provided as Omaha Public Power District's report of changes, tests and experiments performed pursuant to 10 CFR 50.59 for the Fort Calhoun Station. Attachment B is provided to describe Quality Assurance (QA) Program (USAR Appendix A) changes, as required by 10 CFR 50.54(a)(4)(ii), and regulatory commitments revised that require Commission notification in accordance with NEI 99-04. As required by 10 CFR 50.71(e), Attachment C (Revised USAR Sections and Figures) provides the changes made since the previous submittal to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirements and includes changes made under the provisions of 10 CFR 50.59 but not previously submitted to the Commission. The attached information covers the period of January 1, 2000 through June 26, 2001.

In this submittal, Omaha Public Power District is reissuing the sections of the USAR that contain revisions. All USAR copies should be updated to reflect the enclosed revisions. 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 affected sections of the USAR. The Senior Resident Inspector copy of the USAR has been updated through the plant's distribution process.

I declare under penalty of perjury that the foregoing is true and correct. (Executed on October 5, 2001)

Sincerely,

S. K. Gambhir
Division Manager
Nuclear Operations

A053

Attachments

SKG/DLS/dls

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

Attachment A
LIC-01-0076

Changes, Tests, and Experiments Performed Pursuant to 10 CFR 50.59

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
DCN 10123, 99-062	Replacement of waste disposal system check valve WD-714 with a two-inch ball valve.	The evaluation has determined that the changes to the affected equipment and associated system do not affect nuclear safety. The change does not affect Technical Specifications or the plant's ability to comply with Technical Specifications. There is no impact to accidents already evaluated in the USAR, and the change does not result in the probability of new accidents not evaluated in the USAR. This activity is not an unreviewed safety question.
DCN 5097, 99-074	Addition of a pressure reducing valve to steam generator (SG) blowdown sampling points.	The evaluation has determined that the changes to the affected equipment and associated system do not affect nuclear safety. The change does not affect Technical Specifications or the plant's ability to comply with Technical Specifications. There is no impact to accidents already evaluated in the USAR, and the change does not result in the probability of new accidents not evaluated in the USAR. This activity is not an unreviewed safety question.
DCN 5515, 00-001	Installation of isolation valves and quick disconnect stems in the tubing downstream of steam generator blowdown sample coolers SL-8A & B, and check valves SL-140 and SL-141 as described in P&ID	The evaluation has determined that the changes to the affected equipment and associated system do not affect nuclear safety. The change does not affect Technical Specifications or the plant's

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
	11405-M-12 SH 1.	ability to comply with Technical Specifications. There is no impact to accidents already evaluated in the USAR, and the change does not result in the probability of new accidents not evaluated in the USAR. This activity is not an unreviewed safety question.
DCR 10516, 00-002	Configuration change to steam generator blowdown transfer pump recirculation line isolation valve FW-1023 as described on P&ID 11405-M-253 Cover Sheet & SH 2, relating to a change in valve type.	The subject valve, FW-1023, is a manually operated valve. It has no safety related or important-to-safety design functions. The evaluation concluded that no adverse interactions, which could affect nuclear safety functions, are credible.
DCR 11964	Clarification of description in USAR describing the method of controlling reactor coolant temperature in USAR Section 4.3.7.	This evaluation has concluded that this activity does not increase the probability or consequences of previously analyzed accidents or malfunctions and does not create a new failure mode that could create or cause an accident or malfunction of a different type. Margins of safety are not affected.
DCN 10235	Component cooling water (CCW) system carbon steel supply piping to reactor coolant pump RC-3A lube oil cooler will be replaced with stainless steel piping affecting P&ID 11405-M-40 (USAR 9.7.2 &	This change does not affect the functional performance of the CCW system or the operability requirements of the CCW system. This change does not reduce the margin of

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
	Appendix X).	safety. This change does not alter the operating parameters of the CCW system nor does it adversely affect the functional performance of the CCW system post DBA. The use of stainless steel piping does not affect accident initiating mechanisms, accident consequences, or the margin of safety. This activity therefore does not constitute an unreviewed safety question.
DCN 10236	Component cooling water (CCW) system carbon steel supply piping to reactor coolant pump RC-3B lube oil cooler will be replaced with stainless steel piping affecting P&ID 11405-M-40 (USAR 9.7.2 & Appendix X).	This change does not affect the functional performance of the CCW system or the operability requirements of the CCW system. This change does not reduce the margin of safety. This change does not alter the operating parameters of the CCW system nor does it adversely affect the functional performance of the CCW system post DBA. The use of stainless steel piping does not affect accident initiating mechanisms, accident consequences, or the margin of safety. This activity therefore does not constitute an unreviewed safety question.
DCN 10237	Component cooling water (CCW) system carbon steel supply piping to reactor coolant pump RC-3C lube oil cooler will be replaced with stainless steel piping affecting P&ID 11405-M-40 (USAR 9.7.2 &	This change does not affect the functional performance of the CCW system or the operability requirements of the CCW system. This change does not reduce the margin of

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
	Appendix X).	safety. This change does not alter the operating parameters of the CCW system nor does it adversely affect the functional performance of the CCW system post DBA. The use of stainless steel piping does not affect accident initiating mechanisms, accident consequences, or the margin of safety. This activity therefore does not constitute an unreviewed safety question.
DCN 10238	Component cooling water (CCW) system carbon steel supply piping to reactor coolant pump RC-3D lube oil cooler will be replaced with stainless steel piping affecting P&ID 11405-M-40 (USAR 9.7.2 & Appendix X).	This change does not affect the functional performance of the CCW system or the operability requirements of the CCW system. This change does not reduce the margin of safety. This change does not alter the operating parameters of the CCW system nor does it adversely affect the functional performance of the CCW system post DBA. The use of stainless steel piping does not affect accident initiating mechanisms, accident consequences, or the margin of safety. This activity therefore does not constitute an unreviewed safety question.
DCN 10171	Installation of double door (wafer style) check valves and vacuum breakers in the fire protection system piping to mitigate the effects of water hammer (DCP 10183 to drawing 11405-M-266).	This change has no affect upon design functions described in the USAR. The design basis and assumption of the Fire Hazards Analysis (FHA) are maintained. This evaluation has determined

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
		that this activity does not increase the probability or consequences of previously analyzed accidents or malfunctions nor does it create a new failure mode that could create or cause an accident or malfunction of a different type not previously analyzed.
DCN 10335	Installation of a new air line oil filter/moisture trap for trash rack sluice hoist HE-6 (MCC per GEI-35, DCN 10335, P&ID 11405-M-263).	This change has no affect upon design functions described in the USAR. This evaluation has determined this activity does not increase the probability or consequences of previously analyzed accidents or malfunctions nor does it create a new failure mode that could create or cause an accident or malfunction of a different type not previously analyzed.
DCR 12340	Incorporate administrative changes to the USAR (EC 21283, USAR Section 7.6.4 & App M Section 3.5.1).	Administrative and clarifying changes are being made to the USAR discussion on the alternate shutdown panel correcting confusing terminology. This activity does not increase the probability or consequences of previously analyzed accidents or malfunctions nor does it create a new failure mode that could create or cause an accident or malfunction of a different type not previously analyzed.

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
EC 14951	Replacement of raw water (RW) system return check valve CW-188 with a stainless steel check valve.	There are no safety functions associated with this valve, and RW direct cooling is not credited in USAR Section 14 accident analysis. No actions taken by this modification will result in increasing the consequences or the probability of occurrence of an accident or the probability of a malfunction of equipment important to safety.
EC 14983	Installation of inside containment manual valve for testing of containment penetration M-22 per 10 CFR 50, Appendix J. The thermal relief valve will provide protection against over-pressurization of the piping.	Installation of the manual valve will ensure that proper testing of containment penetration M-22 is performed in accordance with 10 CFR 50, Appendix J. No actions taken by this modification will result in increasing the consequences or the probability of occurrence of an accident or the probability of a malfunction of equipment important to safety.
EC 14964	Modification of the chemical volume and control system (CVCS) letdown strainer CH-24 to allow it to be cleaned without mechanical agitation.	This modification allows CVCS letdown strainer CH-24 to be cleaned without mechanical agitation. No plant safety functions are affected. This modification will not affect the method by which the CVCS system is operated during normal, transient, and accident conditions. The strainer does not perform a safety related function. This change does not result in an

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
		unreviewed safety question.
EC 14882	Modification to minimize the infiltration of dust into the control cabinets associated with the ESF lockout relays.	There are no adverse affects upon design functions. This activity does not increase the probability or consequences of previously analyzed accidents or malfunctions nor does it create a new failure mode that could create or cause an accident or malfunction of a different type not previously analyzed.
EC 14994	Emergency feedwater pump FW-10 reliability enhancements.	Design function of FW-10 is to supply auxiliary feedwater to the steam generators in an event where the main feedwater pumps are unavailable and decay heat removal is required. The evaluation discussed the changes to FW-10's lube oil system and the related minor electrical changes to FW-10's control circuitry. The evaluation determined that there was no adverse affect on FW-10's performance or reliability. There is no decrease in safety because of this change or due to the modification changes in general.
DCR 11151	Procedure change to add operability requirements	Design function affected is fire protection and detection. Operability criteria as credited in the

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
	of Appendix R (Emergency Lighting) to SO-G-103.	UFHA are transferred to this procedure. This will ensure that the features credited for fire protection in the USAR are correctly interpreted. This is an administrative change and has no affect upon design functions or the related accidents or malfunctions.
EC 25494	Clarification and correction of statements in the USAR description of the auxiliary feedwater system.	This change involves corrections and clarifications to USAR Section 9.4 and 14.10 discussions on the auxiliary feedwater system. There are no changes to design functions. The evaluation concludes that there is no effect on component or system performance and no adverse system interactions are created by the change. The changes do not result in an unreviewed safety question.
EC 14959	This modification will provide a free-standing switchgear consisting of a new three pole, 4160 VAC vacuum contactor, two mechanically keyed interlocked 4160 VAC isolation switches, and low voltage compartment with key-operated control switch.	There are no safety functions related with this change. This activity will not increase the probability or consequences of any accident previously analyzed in the USAR nor create any unanalyzed accident. The change will enhance the performance of the electrical distribution system.
EC 25423	Temporary modification to the compressed air	There are no safety functions related with this

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
	supply for operation of circulating water pumps discharge check valves FCV-1904A, B, and C. The supply air is being changed from the normal compressed air system to an instrument air system.	change. This activity will not increase the probability or consequences of any accident or malfunction previously analyzed in the USAR nor create any unanalyzed accident.
EC 25532	Administrative and analysis changes to chemistry administrative procedure CH-AD-0003 (Plant Systems Chemical Limits and Corrective Actions) and USAR Table 9.2-2 to meet EPRI guidelines.	The change to USAR Table 9.2-2 necessitates the safety evaluation; however, the limit is being changed only to reflect the ability to more accurately measure pure water pH and the fact that small amounts of carbon dioxide absorption during the sampling process can cause the results to read slightly lower than actual. No unreviewed safety question is created.
EC 25696	Temporary modification to reactor coolant system (RCS) cold leg temperature instrument loops T-121 and T-122 to compensate for an unreliable resistance temperature detector (RTD) (T-121C).	This temporary modification will use the reactor protection system (RPS) safety-related RCS cold leg temperature loop B/T-122C temperature signal as the input to the reactor regulating non-safety related RCS cold leg temperature loop T-121C, including the computer point T0121C. Isolation between the two systems is provided to ensure that the RPS is not affected. Nuclear safety, the RPS functions, and operational characteristics are not being altered or impacted in any way. There are no reductions in margin of safety. There are no changes to RPS setpoints.

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
		There is no impact on any equipment important to safety.
EC 25851	Pressurizer temperature nozzle leak repair.	Repairs will be made in accordance with ASME Section XI Code requirements. Evaluation by Westinghouse has concluded that the amount of corrosion that is likely to take place is insignificant and will not affect the design function of the pressurizer. No unreviewed safety question is created.
EC 25891	Temporary modification to cross-tie the variable leg of pressurizer level detector LT-106 to the variable leg of pressurizer level detector LT-101X.	The design function affected is pressurizer level indication. This temporary modification will not impact the accuracy of the affected instruments and there will be no affect on any setpoints associated with the instrumentation. This change has no affect on the margin of safety and does not affect the occurrence or consequences of any accidents either new or previously evaluated by the USAR.
EC 25898	Modification of pressurizer steam space temperature element TE-107 to install a Mechanical Nozzle Seal Assembly (MNSA).	A reinforcing structural attachment known as a Mechanical Nozzle Seal Assembly (MNSA) will be placed around the upper Pressurizer temperature nozzle for TE-107. This change has no adverse affects upon any design functions

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
		described in the USAR. This activity does not increase the probability or consequences of previously analyzed accidents or malfunctions and does not create a new failure mode that could create or cause an accident or malfunction of a different type.
EC 25873	Procedure revisions to abnormal operating procedure AOP-31 (161 kV Grid Malfunctions) and its associated technical basis document TBD-AOP-31 addressing 161 kV grid instability guidance.	The procedure changes add quantitative criteria to verify operability of the 161 kV offsite power and define the required compensatory actions. The safety evaluation determined that the changes do not affect reactor safe shutdown nor the equipment required to safely shutdown the plant. No unreviewed safety question is created.
EC 15006	Substitute replacement item (SRI) engineering change (EC) material replacement for raw water (RW) system spool pieces described in USAR Section 9.8 & P&ID 11405-M-100. This modification will change the pipe class and material specification.	This evaluation is being performed because the RW spool pieces, which are described in the USAR, are being changed from carbon steel to stainless steel (SS). Replacement with SS will significantly reduce an active wear mechanism in the RW system piping, and RW system availability will be enhanced. No new failure modes or mechanisms are introduced. There are no adverse affects upon RW system design functions.

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
		functions.
EC 14962	Modification to install debris filter for condensate cooler FW-3 and bearing water cooler CW-6A as described in USAR Fig 8.1-1 and Drawing 11405-M-257.	This evaluation considered the impact of the installed debris filter on the operation of the circulating water and related systems under normal and accident conditions. Neither the normal operation nor any failure of this equipment will adversely affect any safety related system required to mitigate the effects of postulated accidents. The presence of this equipment will not create a new failure mode for any safety related equipment.
EC 14974	Replacement of carbon steel components with alloy-steel for feedwater heaters FW-3A & B affecting drawing 11405-M-252, Sheet 3 (DCN 10262).	This change has no adverse affects upon design functions of the feedwater system and will improve the system's ability to resist flow accelerated corrosion (FAC). The evaluation has concluded that the change does not create any unreviewed safety concerns and has no negative affect upon the USAR Safety Analysis.
EC 26581	Procedure for component cooling water (CCW) system drain down and refill. This procedure will control activities during a CCW system outage.	This evaluation has determined that this activity does not increase the probability or consequences of a previously analyzed accident nor does it create a new failure mode that could create or cause an accident of a different type not

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
		previously analyzed. This temporary change does not have an adverse affect upon system operation, reliability, or structural integrity.
EC 26580	Procedure for raw water (RW) system drain down and refill. This procedure will control activities during the RW system outage.	This evaluation has determined that this activity does not increase the probability or consequences of a previously analyzed accident nor does it create a new failure mode that could create or cause an accident of a different type not previously analyzed.
EC 26641	Temporary modification to minimize leak-off from pressurizer spray valve PCV-103-1 affecting USAR Section 4.3.1 and P&ID E-23866-210-110.	This temporary change will not alter the performance or operation of PCV-103-1. Any potential consequences associated with this configuration are bounded by existing USAR analysis. This temporary modification will be removed during the 2001 refueling outage (spring 2001).
EC 26721	Temporary modification to provide an alternate steam generator (SG) blowdown sampling point for steam generator RC-2A affecting USAR Section 14.14.	This change will obtain the SG sample from the SG blowdown sampling system in lieu of the normal sample line direct from the SG. The use of an alternate sample path for obtaining SG blowdown samples does not adversely affect system design functions. This activity does not increase the probability or consequences of

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
		could create or cause an accident or malfunction of a different type.
EC 26711	USAR change to Section 2.5 to clarify differences between general and local climate and clarifies differences in National Weather Service Stations.	These changes have no affect upon design functions for systems described in the USAR. These changes are purely clarifying in nature and are in compliance with the Fort Calhoun Station Offsite Dose Calculation Manual (ODCM). This change presents no safety questions because it only changes the description of weather conditions and locations where data is taken and has no effect upon equipment or methodology for determining off site dose.
EC 27029 and EC 27151	Revision to Technical Data Book (TDB) Section VI (i.e., TDB-VI) and the USAR to reflect the update for Cycle 20 core reload analysis.	The changes made to the core operating limits report (COLR) will update the boron concentration required for refueling based upon the cycle 20 core analysis. These changes ensure that the Technical Specifications's shutdown requirements are maintained and thus assure all safety margins are maintained. These changes have been developed within NRC approved reload analysis methodology.
EC 27057	Revision to Technical Data Book (TDB) Section VI (i.e., TDB-VI) and the USAR to correct changes made in the previous update for Cycle 20 core	This change returns the core operating limits report (COLR) to the state that it was for the cycle 19 analysis. These changes correct

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
	(i.e., TDB-VI) and the USAR to correct changes made in the previous update for Cycle 20 core reload analysis	report (COLR) to the state that it was for the cycle 19 analysis. These changes correct several pages of the COLR that were incorrectly revised to cycle 18 values. These changes do not constitute a reduction in the margin of safety and the analysis validates the changes. These changes have been developed within NRC approved reload analysis methodology.
EC 27049	Temporary modification to disconnect the source range input to the wide range nuclear instrument (WRNI) detector.	Although the proposed change will result in a reduction of source range signal to the WRNI channel C indication, the channel will remain capable of fulfilling its design function of providing indication of neutron flux from source range through full power level. The change will not impact the performance of the channel for the wide range power signal or indication as utilized by the reactor protection system.
EC 26886	Replacement of raw water (RW) system strainers AC-12A & B backwash piping floor penetration.	The design function affected is the flood boundary. The evaluation considered the changes to the flush piping and floor seal with respect to the licensing basis for normal operation and external events (flooding). The conclusion is that the modified configuration has better pressure integrity than the original design

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
		and its presence will not result in any new failure modes or compromise any safety features of the plant. No unreviewed safety question is created.
EC 14983	Field design change request #27130 for revision to DCN 10271 – installation of a locked closed manual isolation valve and test vent valve on containment penetration M-22.	This is a re-approval of the previous evaluation due to changes. There is no affect upon the previous evaluation conclusions.
EC 27174 and EC 27177	Modification to Siemens fuel assemblies to prevent interference/physical contact with six cap-screws on the reactor upper core support plate (plant procedures RE-RR-FE-0002 and OI-FH-1 and USAR Sections 9.5.1.2, 9.5.2 & 9.5.8).	The design function affected is criticality accident requirements. This evaluation is the result of changes to the USAR allowing the removal of unirriadiated fuel. Strict administrative controls will be established and will prevent a criticality accident. This activity does not increase the probability or consequences of previously analyzed accidents or malfunctions and does not create a new failure mode that could create or cause an accident or malfunction of a different type.
EC14983	Field design change request #27260 for revision to DCN 10271 and revision to the 10 CFR 50.59 safety evaluation approved February 26, 2001 for installation of a locked closed manual isolation valve and test vent valve on containment penetration M-22.	This is a re-approval of the previous evaluation due to changes. There is no affect upon the previous evaluation conclusions.

CHANGES, TESTS, AND EXPERIMENTS PERFORMED PURSUANT TO 10 CFR 50.59

Change Number	Description	Summary
EC 27317	Modification to provide double isolation valves for safety injection tank sample valves as described on P&ID E-3866-210-130 Cover sheet 2B and sheet 2.	This modification will install a second isolation valve in the existing sample lines. This will provide zero leakage from the modified sample lines and minimize leakage from the existing isolation valves. The addition of these valves does not affect the design function of the associated valves or system.
EC 27729	Technical specification interpretation (TSI) 01-03-0 extended the containment integrated leak rate test (ILRT) interval as allowed by NEI 94-01, *Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J.*	The safety evaluation indicates that no safety concern is raised by extending the ILRT test interval by six months. This extension is solely due to the scheduling of outages which are driven by the Fort Calhoun Station established 18 month refueling interval. This six month extension is within the 25 percent allowance as described in NEI 94-01. There are no adverse affects upon design functions.

Abbreviations and Acronyms:

DCN - Design Change Notice
DCR - Design Change Request
ECN - Engineering Change Notice
P&ID - Process and Instrumentation Drawing
DCP - Design Change Package
EC - Engineering Change
MCC - Minor Configuration Change
USAR - Updated Safety Analysis Report

Attachment B
LIC-01-076

Quality Assurance Program Changes and Regulatory Commitments Revised in Accordance with NEI 99-04

QUALITY ASSURANCE PROGRAM CHANGES AND REGULATORY COMMITMENTS REVISED IN ACCORDANCE WITH NEI 99-04

QA Program Change Number	Description
EC 20734	In accordance with 10 CFR 50.54(a)(3), Fort Calhoun Station (FCS) revised the inspection qualifications of personnel performing quality control inspections of activities affecting the fire protection program. This revision was a reduction to previous commitments and, therefore, required Commission approval. The Commission concluded (refer to NRC letter dated April 12, 2000) that the reductions in commitments continued to satisfy the requirements of Appendix B to 10 CFR 50 and were acceptable.
EC 25335	A. QA Program specific organizational titles were changed to generic titles. The Nuclear Safety Review Group (NSRG) and QA/QC departments were combined under one manager. Several administrative FCS Technical Specifications related to audits, procedures, and QA records were duplicated to the QA Program. The changes described above were not a reduction of commitment and, therefore, did not require NRC review and approval in accordance with 10 CFR 50.54(a)(3). B. The QA Program was revised to change the commitments regarding applicability to communication equipment intended for the Fire Brigade and to remove Fire Brigade communication equipment from the listing of fire protection-related equipment. This revision was a reduction to previous commitments and, therefore, required Commission approval. The Commission concluded (refer to NRC letter dated July 3, 2000) that the reductions in commitments continued to satisfy the requirements of Appendix B to 10 CFR 50 and were acceptable.

QUALITY ASSURANCE PROGRAM CHANGES AND REGULATORY COMMITMENTS REVISED IN ACCORDANCE WITH NEI 99-04

QA Program Change Number	Description
EC 26027	The QA Program was revised to allow the use of optical storage of records in accordance with NRC Generic Letter 88-18. This change was not a reduction of commitment and, therefore, did not require NRC review and approval in accordance with 10 CFR 50.54(a)(3).
EC 27342	The QA Program was revised to correct typographical errors.

QUALITY ASSURANCE PROGRAM CHANGES AND REGULATORY COMMITMENTS REVISED IN ACCORDANCE WITH NEI 99-04

Commitment Number	Description
AR 4769	In response to NUREG-0758, Section 2.1.6.a, by letter dated October 25, 1979 (LIC-79-0131), OPPD stated, "OPPD will implement a series of surveillances to inspect auxiliary systems for leakage, quantify and evaluate any leaks that are found to the extent possible, and initiate appropriate corrective actions... this Surveillance program will be generally applied to... Portions of the waste disposal system including containment sump piping, spent regenerative tanks and piping, neutralization tank and piping, SI and CVCS valve leakoff piping, auxiliary building sump tank and piping, monitor tanks inlet piping, waste filters, gas stripper, waste evaporator and piping, monitor tanks and recirculation piping, waste evaporator to drumming piping, concentrate tanks, waste holdup tanks, spent resin storage and associated piping and pumps, and the gaseous waste disposal system." The waste disposal system (WDS) components were classified as ASME Safety Class 2 or 3, and were required to be tested and VT-2 inspected in accordance with ASME Section XI, Article IWA-5000. Subsequently, the Fort Calhoun Station Updated Safety Analysis Report (USAR) was revised which reclassified the majority of the WDS components/piping as Non-Nuclear Safety (Class 4). Because of this reclassification, the ASME Section XI testing no longer applies to these WDS components. Alternative methods of determining WDS integrity will be used in lieu of the above stated commitment. In addition, several WDS components have been or will be "abandoned in place" and, therefore, these portions of the WDS will no longer have the potential to contain highly radioactive material.

QUALITY ASSURANCE PROGRAM CHANGES AND REGULATORY COMMITMENTS REVISED IN ACCORDANCE WITH NEI 99-04

Commitment Number	Description
AR 9368	In OPPD correspondence dated October 25, 1979 (LIC-79-0131), OPPD stated "The backup power supply will be the emergency diesel generator normally associated with [the low pressure safety injection] LPSI pump required to be operable or in operation." At the time this commitment was made, the emergency diesel generator was the only available source of backup power for the LPSI pump. Since then, extensive upgrades have been made to the Fort Calhoun Station switchyard, providing several alternate success paths. Therefore, in order to minimize operability restrictions, OPPD no longer considers this a regulatory commitment.
AR-8515 & AR-9357	In response to NRC Generic Letter 88-17, by letter dated January 1, 1989 (LIC-88-1106), OPPD stated "upon replacement of the equipment hatch as specified in checklist OI-CO-4-CL-B, that the seal be pressure tested." This commitment is being revised to "upon replacement, the equipment hatch door will be closed and held in place by a sufficient number of bolts such that there is reasonable assurance that no gaps exist in the seating surface." The hazards analysis associated with this revision has determined that even with the equipment hatch not installed, the radiological consequences are less than the values in the USAR.

Attachment C
LIC-01-076

Revised USAR Sections and Figures

Fort Calhoun Station
September 2001, USAR Update
Replacement Instuctions

Remove	**Insert**
USAR Revision Status	USAR Revision Status (9/25/01)
Section 1, Table of Contents	Section 1, Table of Contents (7/23/01)
Section 1.2	Section 1.2 Rev 8
Section 1.3	Section 1.3 Rev 2
Section 1.4	Section 1.4 Rev 2
Section 1.6	Section 1.6 Rev 2
Section 1.11	Section 1.11 Rev 1
Section 2, Table of Contents	Section 2, Table of Contents (5/16/01)
Section 2.5	Section 2.5 Rev 4
Section 2.8	Section 2.8 Rev 2
Section 2.10	Section 2.10 Rev 3
Section 3, Table of Contents	Section 3, Table of Contents (7/23/01)
Section 3.1	Section 3.1 Rev 3
Section 3.2	Section 3.2 Rev 3
Section 3.4	Section 3.4 Rev 11
Section 3.5	Section 3.5 Rev 1
Section 3.6	Section 3.6 Rev 4
Section 3.7	Section 3.7 Rev 6
Section 3.9	Section 3.9 Rev 4
Section 4, Table of Contents	Section 4, Table of Contents (6/1/01)
Section 4.3	Section 4.3 Rev 10
Section 4.5	Section 4.5 Rev 1
Section 4.6	Section 4.6 Rev 2
Section 5, Table of Contents	Section 5, Table of Contents (10/6/00)
Section 5.4	Section 5.4 Rev 2
Section 6, Table of Contents	Section 6, Table of Contents (6/1/01)
Section 6.1	Section 6.1 Rev 2
Section 6.3	Section 6.3 Rev 7
Section 6.4	Section 6.4 Rev 2
Section 7, Table of Contents	Section 7, Table of Contents (7/24/01)
Section 7.2	Section 7.2 Rev 3
Section 7.3	Section 7.3 Rev 3
Section 7.4	Section 7.4 Rev 1
Section 7.5	Section 7.5 Rev 9
Section 7.6	Section 7.6 Rev 6

Fort Calhoun Station
September 2001, USAR Update
Replacement Instuctions

Remove	Insert
Section 8, Table of Contents	Section 8, Table of Contents (7/25/01)
Section 8.2	Section 8.2 Rev 5
Section 8.3	Section 8.3 Rev 3
Section 8.4	Section 8.4 Rev 4
Section 8.5	Section 8.5 Rev 4
Section 9, Table of Contents	Section 9, Table of Contents (6/1/01)
Section 9.2	Section 9.2 Rev 9
Section 9.4	Section 9.4 Rev 9
Section 9.5	Section 9.5 Rev 10
Section 9.8	Section 9.8 Rev 8
Section 9.10	Section 9.10 Rev 7
Section 9.11	Section 9.11 Rev 7
Section 9.12	Section 9.12 Rev 5
Section 11, Table of Contents	Section 11, Table of Contents (6/1/01)
Section 11.1	Section 11.1 Rev 7
Section 14, Table of Contents	Section 14, Table of Contents (7/25/01)
Section 14.1	Section 14.1 Rev 4
Section 14.2	Section 14.2 Rev 4
Section 14.3	Section 14.3 Rev 3
Section 14.4	Section 14.4 Rev 4
Section 14.5	Section 14.5 Rev 1
Section 14.6	Section 14.6 Rev 4
Section 14.9	Section 14.9 Rev 3
Section 14.10	Section 14.10 Rev 5
Section 14.11	Section 14.11 Rev 4
Section 14.12	Section 14.12 Rev 5
Section 14.13	Section 14.13 Rev 2
Section 14.14	Section 14.14 Rev 5
Section 14.15	Section 14.15 Rev 7
Section 14.22	Section 14.22 Rev 5
Appendix A	Appendix A Rev 8
Appendix G	Appendix G Rev 1
Appendix M	Appendix M Rev 3
Appendix N	Appendix N Rev 3
Figure 1.2-11	Figure 1.2-11 Rev 13
Figure 1.2-13	Figure 1.2-13 Rev 9

Fort Calhoun Station
September 2001, USAR Update
Replacement Instuctions

Remove	Insert
Figure 3.1-2	Figure 3.1-2 Rev 1
Figure 3.4-1	Figure 3.4-1 Rev 6
Figure 3.4-2	Figure 3.4-2 Rev 6
Figure 3.4-4	Figure 3.4-4 Rev 6
Figure 3.4-5	Figure 3.4-5 Rev 6
Figure 3.4-6	Figure 3.4-6 Rev 6
Figure 3.4-7	Figure 3.4-7 Rev 6
Figure 4.3-9	DELETED
Figure 4.3-10	Figure 4.3-10 Rev 1
Figure 5.9-4	Figure 5.9-4 Rev 12
Figure 5.9-5	Figure 5.9-5 Rev 5
Figure 5.9-6	Figure 5.9-6 Rev 5
Figure 5.9-7	Figure 5.9-7 Rev 5
Figure 5.9-8	Figure 5.9-8 Rev 4
Figure 5.9-9	Figure 5.9-9 Rev 4
Figure 5.9-10	Figure 5.9-10 Rev 5
Figure 5.9-11	Figure 5.9-11 Rev 3
Figure 5.9-12	Figure 5.9-12 Rev 2
Figure 5.9-13 Sheet 1	Figure 5.9-13 Sheet 1, Rev 5
Figure 5.9-13 Sheet 2	Figure 5.9-13 Sheet 2, Rev 5
Figure 5.9-13 Sheet 19	Figure 5.9-13 Sheet 19, Rev 5
Figure 5.9-13 Sheet 23	Figure 5.9-13 Sheet 23, Rev 3
Figure 5.9-13 Sheet 43	Figure 5.9-13 Sheet 43, Rev 3
Figure 5.9-13 Sheet 61	Figure 5.9-13 Sheet 61, Rev 2
Figure 5.9-15	Figure 5.9-15 Rev 16
Figure 6.2-3	Figure 6.2-3 Rev 1
Figure 6.4-1	Figure 6.4-1 Rev 1
Figure 7.2-5	Figure 7.2-5 Rev1
Figure 7.2-9	Figure 7.2-9 Rev 1
Figure 7.3-3	Figure 7.3-3 Rev 1
Figure 7.4-3	Figure 7.4-3 Rev 1
Figure 7.4-4	Figure 7.4-4 Rev 1
Figure 7.5-1	Figure 7.5-1 Rev 1
Figure 8.1-1	Figure 8.1-1 Rev 11
Figure 8.5-2	Figure 8.5 –2 Rev 63
Figure 9.2-1	Figure 9.2-1 Rev 1

Fort Calhoun Station
September 2001, USAR Update
Replacement Instuctions

Remove	Insert
	Figure 14.6-6 through 14.6-11 Rev 0
	Figure 14.13-1 through 14.13-5 Rev 0
Figure 14.15-1 through 14.15-40	DELETED
	Figure 14.15-41 through 14.15-68 Rev 0

Document Number	*Title*	*Revision/Date*
USAR-1	**Introduction and Summary**	
Section 1.1	Introduction	R0 05/23/97
Section 1.2	Summary Plant Description	R8 07/31/01
Section 1.3	Principal Architectural and Engineering Criteria	R2 07/31/01
Section 1.4	Research and Development Requirements	R2 06/01/01
Section 1.5	Items of Special Interest to ACRs	R1 05/23/97
Section 1.6	Significant Design changes since Award of Construction Permit	R2 07/31/01
Section 1.7	Identification of Contractors	R0 05/23/97
Section 1.8	Technical Qualifications of the Applicant	R1 07/27/98
Section 1.9	Design Comparison with Other Plants	R0 05/23/97
Section 1.10	General Conclusions	R0 05/23/97
Section 1.11	Specific References	R1 06/01/01
Section 1.12	General References	R0 05/23/97
USAR-2	**Site and Environs**	
Section 2.1	Introduction	R0 05/23/97
Section 2.2	General Description of Site and Environs	R0 05/23/97
Section 2.3	Topography	R0 05/23/97
Section 2.4	Seismology	R0 05/23/97
Section 2.5	Meteorology	R4 05/17/01

Section 2.6	Geology	R0 05/23/97
Section 2.7	Hydrology	R0 05/23/97
Section 2.8	Demography	R2 06/27/00
Section 2.9	Land Use	R0 05/23/97
Section 2.10	Environmental Radiation Monitoring	R3 05/10/00
Section 2.11	Section 2 References	R1 02/02/98
USAR-3	**Reactor**	
Section 3.1	Summary of Description	R3 07/31/01
Section 3.2	Design Bases	R3 07/31/01
Section 3.3	Reactor and Core Fuel Cycle Considerations	R1 09/24/98
Section 3.4	Nuclear Design and Evaluation	R12 08/23/01
Section 3.5	Cycle 6 Core Power Uprate to 1500 MW_T	R1 07/31/01
Section 3.6	Thermal and Hydraulic Design Evaluation for Limiting Transients and Setpoint Analysis for Cycle 17	R4 06/01/01
Section 3.7	Mechanical Design and Evaluation	R6 06/01/01
Section 3.8	Fuel Performance During Anticipated Transients after Long Burnup	R0 05/23/97
Section 3.9	Specific References	R4 06/01/01
Section 3.10	General References	R0 05/23/97
USAR-4	**Reactor Coolant System**	
Section 4.1	Introduction	R0 05/23/97
Section 4.2	Design Basis	R2 09/24/98

Section 4.3	Component and System Design and Operation	R10 06/01/01
Section 4.4	System Design Evaluation	R1 05/23/97
Section 4.5	Tests and Inspections	R1 06/01/01
Section 4.6	Specific References	R2 06/01/01
Section 4.7	General References	R1 09/24/98
USAR-5	**Structures**	
Section 5.1	Containment Structure	R0 05/23/97
Section 5.2	Materials of Construction	R1 10/14/98
Section 5.3	Construction Procedures	R0 05/23/97
Section 5.4	Containment Loadings	R2 10/10/00
Section 5.5	Containment Design Criteria	R0 05/23/97
Section 5.6	Containment Design Procedures	R0 05/23/97
Section 5.7	Piling	R0 05/23/97
Section 5.8	Missile Protection and Pipe Whipping Restraints	R0 05/23/97
Section 5.9	Containment Penetrations	R6 03/07/00
Section 5.10	Containment Tests and Inspection	R0 05/23/97
Section 5.11	Structures other than Containment	R2 12/16/98
Section 5.12	Specific References	R2 03/07/00
Section 5.13	General References	R1 05/23/97
USAR-6	**Engineered Safeguards**	
Section 6.1	General	R2 06/01/01
Section 6.2	Safety Injection System	R10 12/01/99

Section 6.3	Containment Spray System	R7 06/01/01
Section 6.4	Containment Air Recirculation, Cooling and Iodine Removal System	R2 06/01/01
Section 6.5	Specific References	R2 01/12/99
Section 6.6	General References	R1 02/02/98
USAR-7	**Instrumentation and Control**	
Section 7.1	Introduction	R1 05/23/97
Section 7.2	Reactor Protective System	R3 07/31/01
Section 7.3	Engineered Safeguards Controls and Instrumentation	R3 09/12/00
Section 7.4	Regulating Systems	R1 07/31/01
Section 7.5	Instrumentation Systems	R9 07/31/01
Section 7.6	Operating Control Stations	R7 08/23/01
Section 7.7	General References	R2 09/24/98
USAR-8	**Electrical Systems**	
Section 8.1	Introduction	R1 08/23/01
Section 8.2	Network Interconnections	R6 08/23/01
Section 8.3	Station Distribution	R4 08/23/01
Section 8.4	Emergency Power Sources	R4 09/12/00
Section 8.5	Initial Cable Installation Design Criteria	R4 07/31/01
Section 8.6	General References	R0 05/23/97
USAR-9	**Auxiliary Systems**	

Section 9.1	General	R2 12/22/98
Section 9.2	Chemical and Volume Control System	R9 12/28/00
Section 9.3	Shutdown Cooling System	R2 05/14/99
Section 9.4	Auxiliary Feedwater System	R9 12/28/00
Section 9.5	Refueling System	R10 06/01/01
Section 9.6	Spent Fuel Pool Cooling System	R3 11/12/99
Section 9.7	Component Cooling Water System	R6 11/12/99
Section 9.8	Raw Water System	R8 08/23/01
Section 9.9	Turbine Plant Cooling Water System	R0 05/23/97
Section 9.10	Heating, Ventilating and Air Conditioning Systems	R7 06/01/01
Section 9.11	Fire Protection System	R7 09/12/00
Section 9.12	Compressed Air System	R5 06/27/00
Section 9.13	Sampling Systems	R1 02/02/98
USAR-10	**Steam and Power Conversion Systems**	
Section 10.1	Design Bases	R0 05/23/97
Section 10.2	System Design and Operation	R1 03/28/00
Section 10.3	System Evaluation	R0 05/23/97
Section 10.4	Tests and Inspections	R0 05/23/97
Section 10.5	General References	R0 05/23/97
USAR-11	**Radioactive Waste and Radiation Protection and Monitoring**	
Section 11.1	Radioactive Waste Disposal System	R7 06/01/01

Section 11.2	Radiation Protection and Monitoring	R1 05/23/97
Section 11.3	Radiological Effluent Requirements	R1 11/12/97
Section 11.4	Specific References	R1 05/23/97
Section 11.5	General References	R0 05/23/97
USAR-12	**Conduct of Operations**	
Section 12.1	Organization and Responsibility	R3 11/03/99
Section 12.2	Training	R1 02/02/98
Section 12.3	Written Procedures	R0 05/23/97
Section 12.4	Records	R0 05/23/97
Section 12.5	Review and Audit of Operations	R2 07/27/98
USAR-13	**Initial Tests and Operation**	
Section 13.1	Tests prior to Reactor Fueling	R0 05/23/97
Section 13.2	Reactor Fueling and Physics Tests	R0 05/23/97
Section 13.3	Post-Criticality and Power Escalation	R0 05/23/97
Section 13.4	Operations Restrictions	R0 05/23/97
Section 13.5	References	R0 05/23/97
USAR-14	**Safety Analysis**	
Section 14.1	General	R4 06/01/01
Section 14.2	Control Element Assembly Withdrawal Incident	R4 06/01/01
Section 14.3	Boron Dilution Incident	R3 06/01/01
Section 14.4	Control Element Assembly Drop Incident	R4 06/01/01

Section 14.5	Malpositioning of the Non-Trippable CEAs	R1 06/01/01
Section 14.6	Loss of Coolant Flow Incident	R4 06/01/01
Section 14.7	Idle Loop Startup Incident	R0 05/23/97
Section 14.8	Turbine-Generator Overspeed Incident	R0 05/23/97
Section 14.9	Loss of Load	R3 07/31/01
Section 14.10	Malfunctions of the Feedwater System	R5 07/31/01
Section 14.11	Excess Load Increase	R4 06/01/01
Section 14.12	Main Steam Line Break Accident	R5 06/01/01
Section 14.13	Control Element Assembly Ejection Accident	R2 06/01/01
Section 14.14	Status of Steam Generator Tube Rupture Incident	R5 07/31/01
Section 14.15	Loss of Coolant Accident	R7 06/01/01
Section 14.16	Containment Pressure Analysis	R2 02/02/98
Section 14.17	Generation of Hydrogen in Containment	R0 05/23/97
Section 14.18	Fuel Handling Accident (in Spent Fuel Pool and Containment)	R1 05/23/97
Section 14.19	Gas Decay Tank Rupture	R0 05/23/97
Section 14.20	Waste Liquid Incident	R0 05/23/97
Section 14.21	Maximum Hypothetical Accident	R0 05/23/97
Section 14.22	Reactor Coolant System Depressurization Incident	R5 07/31/01
Section 14.23	Control Room Habitability During Toxic Chemical Release Accidents	R4 05/26/99
Section 14.24	Heavy Load Incident	R5 10/22/99

Appendix A	Quality Assurance Program	R8 04/26/01
Appendix B	Site Environmental Studies	R0 05/23/97
Appendix C	Foundation Studies	R0 05/23/97
Appendix D	Pile Load Test Programs	R0 05/23/97
Appendix E	Appendix E	R0 05/23/97
Appendix F	Classification of Structures and Equipment and Seismic Criteria	R3 06/03/99
Appendix G	Response to 70 Criteria	R1 07/31/01
Appendix H	Training Summaries of Fort Calhoun Station Unit No. 1 Related Personnel	R0 05/23/97
Appendix I	Startup Manual	R0 05/23/97
Appendix J	Supplementary Meteorological Correlations	R0 05/23/97
Appendix K	Revised Analysis of Fort Calhoun Emergency Core Cooling System Performance	R0 05/23/97
Appendix L	Predicted Creep Collapse Behavior for Fort Calhoun Fuel Cladding	R0 05/23/97
Appendix M	Postulated High Energy Line Rupture Outside the Containment	R3 09/12/00
Appendix N	Reclassification of Systems	R3 12/28/00

Figure	Title	Revision
1.2-1	Site Plan (36030)	4
1.2-2	Restricted Area (36031)	4
1.2-3	Primary Plant South Elevation (11405-A-10(12167))	6
1.2-4	Primary Plant West Elevation (11405-A-11(12168))	8
1.2-5	Primary Plant North Elevation (11405-A-12(12169))	7
1.2-6	Primary Plant Roof Plan and Upper Levels (11405-A-9(12166))	17
1.2-7	Turbine and Office Building Cross Sections (11405-A-264(12195))	4
1.2-8	Intake Structure Floor Plans (11405-A-278(12206))	1
1.2-9	Intake Structure Building Sections and Panel Details (11405-A-281(12209))	9
1.2-10	Intake Structure and Tunnels Foundation Plan and Details (11405-S-311(16531))	2
1.2-11	Intake Structure and Tunnels Plan at Elevation 985'-0 and 993'0" and Details (11405-S-312(16532))	13
1.2-12	Intake Structure and Tunnels Plan at Elevation 1007'-6" and Details (11405-S-313(16533))	18
1.2-13	Intake Structure and Tunnels Sections and Details (11405-S-315(16535))	9
1.2-14	Intake Structure and Tunnels Trash Rack and Stop Log Details (11405-S-318(16538))	7
1.2-15	Intake Structure and Tunnels Miscellaneous Details I (11405-S-319(16539))	5
1.2-16	Intake Structure and Tunnels Miscellaneous Details II (11405-S-320(16540))	3

Figure	Title	Revision
1.4-1	Containment Fan - Motors Accident Environment Performance Test Cycle (36034)	0
1.5-1	Perforation Temperature vs Internal Gas Pressure for Zircaloy - 4 Fuel Tubing Tested in Helium Atmosphere (36039)	0
1.5-2	Perforation Temperature vs Heating Rate for Zircaloy -4 Fuel Tubing Tested at 2500 psi in Helium (36040)	0
1.5-3	Circumferential Strain vs Heating Rate for Zircaloy -4 Tubing Tested in Helium (36041)	0
2.2-1	Arial Photograph of Site and Environs (36042)	0
2.2-2	Geographical Features Within 60 Miles of Site (36043)	0
2.2-3	Geographical Features Within 30 Miles of Site (36044)	0
2.2-4	Metes and Bounds (36045)	1
2.3-1	Site Topography (36046)	2
2.7-1	Upper Missouri River Basin (36047)	0
2.7-2	Stage Duration Curve (36048)	0
2.7-3	Flow Duration Curve (36049)	0
2.7-4	Missouri River Rating Curve (36050)	0
2.8-1	Actual Population Distribution as of 1980, 0-5 Miles (36051)	0
2.8-2	Actual Population Distribution as of 1980, 5-50 Miles (36052)	0
2.8-3	Projected Population Distribution as of 2010, -0-5 Miles (36053)	0
2.8-4	Projected Population Distribution as of 2010, 5-50 Miles (36054)	0
2.10-1	Environmental Radiological Sampling Points, On Site (36055)	2

Figure	Title	Revision
2.10-2	Environmental Radiological Sampling Points, 0-5 Miles (36056)	4
2.10-3	Environmental Radiological Sampling Points, 5-20 Miles (36057)	2
3.1-1	Reactor Arrangement (36414)	0
3.1-2	Reactor Core Cross Section (36415)	1
3.4-1	Assy Avg Burnup Distribution, Beginning of Cycle (36416)	6
3.4-2	Assy Avg Burnup Distribution, End of Cycle (17,000 MWD/MTU) (36417)	6
3.4-3	Power Dependent, Insertion Limit (36418)	0
3.4-4	Position of Fuel Assemblies and CEA Groups (36419)	6
3.4-5	Core Radial Power Distribution, Beginning of Cycle ARO (36420)	6
3.4-6	Core Radial Power Distribution at 7000 MWD/MTU Burnup ARO (36421)	6
3.4-7	Core Radial Power Distribution, EOC (17,000 MWD/MTU) ARO (36422)	6
3.4-9	Comparison of Calculated and Measured Plutonium-to-Uranium Mass Ratio in the Asymptotic Neutron Spectrum for Yankee (36424)	0
3.4-10	Plutonium Isotopic Composition vs Fuel Depletion in the Asymptotic Spectrum for Yankee (36425)	0
3.4-11	Reactivity Difference Between Fundamental and Excited States of a Bare Cylindrical Reactor (36426)	0
3.4-12	Thermal Neutron Flux at the Center of the Core vs Time (36427)	0
3.4-13	Damping Coeff vs Reactivity Difference Between Fundamental and Excited State (36428)	0
3.4-14	End of Life Oscillations with Doppler Feedback, Full Power (2 Hour	0

Figure	Title	Revision
	Time Steps) (36429)	
3.4-15	Split Detector Response to Axial Power Profiles in the Core (36430)	0
3.6-3	Two Phase Flow Regime Transitions Stable Flow Limits (36433)	0
3.6-4	Linear Heat Rate SAFDL Protection by APD Trip (36431)	0
3.7-1	Upper Guide Structure Assembly (36441)	0
3.7-2	Control Element Drive Mechanism (36442)	0
3.7-7	Westinghouse Fuel Rod, R Fuel (36447)	2
3.8-1	Fuel Rod Circumferential Strain vs Temperature (36449)	1
3.8-2	Design Curve for Cyclic Strain Usage of Zircaloy-4 at 700 Degrees F (36450)	0
4.3-1	Reactor Vessel (36451)	1
4.3-2	Reactor Vessel: Summary of Areas Having the Highest Cumulative Fatigue Usage Factors (36452)	0
4.3-3	Steam Generator (36453)	3
4.3-4	Reactor Coolant Pump (36454)	0
4.3-5	Enlarged View Pump Seal Area (36455)	0
4.3-6A	GE Reactor Coolant Motor Assembly (791e611 (6858))	11
4.3-6B	ABB Reactor Coolant Pump Motor Assembly Cross-Section Dwg (HTAM125294 (45078))	0
4.3-7	Reactor Coolant Pump Performance (36457)	0
4.3-8	Pressurizer (36458)	0
4.3-10	Pressurizer Water Level Program (36460)	1

Figure	Title	Revision
4.3-11	Pressurizer Level Control Program (36461)	1
4.3-12	Fracture Toughness of Various Materials (36462)	1
4.3-13	Hoop Stress Distribution in Flywheel at 120% Rated Speed (36463)	1
4.3-14	KI vs Crack Length for 120% Rated Speed - GE Motor (36464)	1
4.5-1	Location of Surveillance Capsule Assemblies (36465)	0
4.5-2	Typical Surveillance Capsule Assembly (36466)	0
4.5-3	Typical Charpy Impact Compartment Assembly (36467)	0
4.5-4	Typical Tensile-Monitor Compartment Assembly(36468)	0
4.5-6	Longitudinal Orientation of Omaha Precritical Reactor System Test Program Instrumentation (36470)	1
4.5-7	Circumferential Orientation of Omaha Precritical Instrumentation (36471)	0
5.1-1	Containment Structure (36473)	0
5.1-2	Containment Structure Arrangement (36474)	0
5.4-1	Post-Accident Transient Temperatures and Pressures (Winter) (36475)	0
5.4-2	Containment Structure Wind Loading Diagrams (36476)	0
5.6-1	Ring Girder Finite Element Grid Thermal Gradient and Pressure (36477)	0
5.6-2	Ring Girder Principal Stresses, Thermal Gradient (36478)	0
5.6-3	Ring Girder Principal Stresses, Pressure (36479)	0
5.6-4	Ring Girder Finite Element Grid, Prestress (36480)	0

Figure	Title	Revision
5.6-5	Ring Girder Principal Stresses, Initial Prestress (36481)	0
5.6-6	Grid-Tangential Section of Anchorage Region (36482)	0
5.6-7	Principal Stresses, Tangential Section of Anchorage Region (36483)	0
5.6-8	Principal Stresses, Tangential Section of Anchorage Region (36484)	0
5.6-9	Principal Stresses, Tangential Section of Anchorage Region (36485)	0
5.6-10	Grid-Radial Section of Anchorage Region (36486)	0
5.6-11	Principal Stresses, Radial Section of Anchorage Region (36487)	0
5.6-12	Anchorage Zone Reinforcement (36488)	0
5.6-13	Grid for Large Openings Analysis (36489)	0
5.6-14	Containment Structure Prestressing System, Sheet 2 of 2 (16391)	4
5.6-15	Arrangement of Reinforcing Steel at Airlock and Equipment Hatch (36490)	0
5.6-16	Liner Analysis Model (36491)	0
5.6-17	Finite Element Grid-Liner Analysis (36492)	0
5.6-18	Finite Element Grid-Liner Analysis (36493)	0
5.6-19	Finite Element Grid-Liner Analysis (36494)	0
5.6-20	Finite Element Grid-Liner Analysis (36495)	0
5.6-21	Concrete Stress - Strain Curve (36496)	0
5.6-22	Modified Mohr Rupture Diagram (36497)	0

Figure	Title	Revision
5.6-23	Force-Displacement Diagram (36498)	0
5.6-24	Load Plots-Dead Load (36499)	0
5.6-25	Load Plots-Prestress (36500)	0
5.6-26	Load Plots-Internal Pressure (36501)	0
5.6-27	Load Plots-Thermal Gradient (36502)	0
5.6-28	Load Plots-Design Earthquake (36503)	0
5.6-29	Load Plots-Wind (36504)	0
5.6-30	Stresses, Concrete and Reinforcement, Sheet 1 of 14 (36505)	0
5.6-31	Stresses, Concrete and Reinforcement, Sheet 2 of 14 (36506)	0
5.6-32	Stresses, Concrete and Reinforcement, Sheet 3 of 14 (36507)	0
5.6-33	Stresses, Concrete and Reinforcement, Sheet 4 of 14 (36508)	0
5.6-34	Stresses, Concrete and Reinforcement, Sheet 5 of 14 (36509)	0
5.6-35	Stresses, Concrete and Reinforcement, Sheet 6 of 14 (36510)	0
5.6-36	Stresses, Concrete and Reinforcement, Sheet 7 of 14 (36511)	0
5.6-37	Stresses, Concrete and Reinforcement, Sheet 8 of 14 (36512)	0
5.6-38	Stresses, Concrete and Reinforcement, Sheet 9 of 14 (36513)	0
5.6-39	Stresses, Concrete and Reinforcement, Sheet 10 of 14 (36514)	0
5.6-40	Stresses, Concrete and Reinforcement, Sheet 11 of 14 (36515)	0
5.6-41	Stresses, Concrete and Reinforcement, Sheet 12 of 14 (36516)	0
5.6-42	Stresses, Concrete and Reinforcement, Sheet 13 of 14 (36517)	0

Figure	Title	Revision
5.6-43	Stresses, Concrete and Reinforcement, Sheet 14 of 14 (36518)	0
5.7-1	Piling Plan Containment and Auxiliary Building Sections and Details (11405-S-1(16380))	7
5.7-2	Piling Plan Intake Structure (11405-S-275(16507))	6
5.9-1	Containment Pipe Penetrations, Sheet 1 (11405-M-70(16236))	6
5.9-2	Containment Pipe Penetrations, Type II, Sheet 2 (11405-M-70(16237))	6
5.9-3	Containment Pipe Penetrations, Sheet 3 (11405-M-70(16239))	7
5.9-4	Containment Pipe Penetrations, Sheet 4 (11405-M-70(16246))	12
5.9-5	Containment Pipe Penetrations, Sheet 5 (11405-M-70(16242))	5
5.9-6	Penetration Cooling Fins and Stiffeners, Sheet 6 (11405-M-70(16240))	5
5.9-7	Containment Pipe Penetrations, Sheet 7 (11405-M-70(16243))	5
5.9-8	Penetration Cooling Fins and Stiffeners, Sheet 8 (11405-M-70(16244))	4
5.9-9	Fuel Transfer Penetration, Detail M-100, Sheet 9 (11405-M-70(16241))	4
5.9-10	Containment Pipe Penetrations, Sheet 10 (11405-M-70(16245))	5
5.9-11	Personnel Air Lock (36529)	3
5.9-12	Equipment Access Hatch (36531)	2
5.9-13	Penetration M-HCV-383-3 Containment Sump Recirculation (Sheet 1) (40122)	5
5.9-13	Penetration M-HCV-383-4 Containment Sump Recirculation (Sheet 2) (40123)	5

Figure	Title	Revision
5.9-13	Penetration M-2 Letdown Heat Exchanger (Sheet 3) (40124)	5
5.9-13	Penetration M-3 CVCS Charging (Sheet 4) (40125)	2
5.9-13	Penetration M-5 Safety Injection (Sheet 5) (40126)	6
5.9-13	Penetration M-6 Safety Injection (Sheet 6) (40127)	3
5.9-13	Penetration M-7 RC Pump Bleedoff (Sheet 7) (40128)	3
5.9-13	Penetration M-8 Containment Sump To WDS (Sheet 8) (40129)	3
5.9-13	Penetration M-9 Steam Generator RC-2B Blowdown (Sheet 9) (40130)	2
5.9-13	Penetration M-10 Steam Generator RC-2B Blowdown (Sheet 10) (40131)	2
5.9-13	Penetrations M-11 & M-15 CCW To Nuclear Detector Well Coolers (Sheet 11) (40132)	3
5.9-13	Penetration M-12 Steam Generator RC-2A Blowdown (Sheet 12) (40133)	2
5.9-13	Penetration M-13 Steam Generator RC-2A Blowdown (Sheet 13) (40134)	2
5.9-13	Penetration M-14 RC Drain Tank Vent To WDS (Sheet 14) (40135)	2
5.9-13	Penetration M-16 SI & Shutdown Cooling From RCS (Sheet 15) (40136)	4
5.9-13	Penetration M-17 SI & Shutdown Cooling To RCS (Sheet 16) (40137)	4
5.9-13	Penetration M-18 CCW From RC Pump Lube Oil Coolers & CEDM Seal Coolers (Sheet 17) (40138)	3
5.9-13	Penetration M-19 CCW From RC Pump Lube Oil Coolers & CEDM Seal Coolers (Sheet 18) (40285)	3

Figure	Title	Revision
5.9-13	Penetration M-20 RC Drain Tank Pump Discharge To WDS (Sheet 19) (40139)	5
5.9-13	Penetration M-22 SI To/From Leakage Cooler (Sheet 20) (40140)	3
5.9-13	Penetration M-24 Pressurizer Quench Tank Sampling (Sheet 21) (40141)	1
5.9-13	Penetration M-25 RC Drain Tank Sampling (Sheet 22) (40142)	1
5.9-13	Penetration M-30 Containment Hydrogen Purge (Sheet 23) (40143)	3
5.9-13	Penetration M-31 Hydrogen Sampling (Sheet 24) (40144)	1
5.9-13	Penetration M-38 Containment Pressure (Sheet 25) (40145)	3
5.9-13	Penetration M-39 CCW to Safety Injection Cooler (Sheet 26) (40146)	3
5.9-13	Penetration M-40 Hydrogen Sampling (Sheet 27) (40147)	1
5.9-13	Penetration M-42 N2 Supply (Sheet 28) (40148)	2
5.9-13	Penetration M-43 N2 Supply (Sheet 29) (40149)	2
5.9-13	Penetration M-44 Pressurizer Dead Weight Tester (Sheet 30) (40150)	2
5.9-13	Penetration M-45 Reactor Coolant System Sampling (Sheet 31) (40151)	2
5.9-13	Penetration M-46 Containment Atmosphere Sampling (Sheet 32) (40152)	3
5.9-13	Penetration M-47 Containment Atmosphere Sampling (Sheet 33) (40153)	3
5.9-13	Penetration M-48 Containment Relief (Sheet 34) (40154)	3
5.9-13	Penetration M-49 SG Blowdown RC-2B Sampling (Sheet 35)	2

Figure	Title	Revision
	(40155)	
5.9-13	Penetration M-50 Containment Pressure Signal (Sheet 36) (40156)	3
5.9-13	Penetration M-51 Containment Pressure Signal (Sheet 37) (40157)	3
5.9-13	Penetration M-52 Containment Pressure Signal (Sheet 38) (40158)	3
5.9-13	Penetration M-53 CCW from Safety Injection Cooler (Sheet 39) (40164)	2
5.9-13	Penetration M-57 Hydrogen Sampling (Sheet 40) (40160)	2
5.9-13	Penetration M-58 Hydrogen Sampling (Sheet 41) (40161)	3
5.9-13	Penetration M-63 SG Blowdown (RC-2A) Sampling (Sheet 42) (40162)	3
5.9-13	Penetration M-69 Hydrogen Purge (Sheet 43) (40163)	3
5.9-13	Penetration M-73 Instrument Air Supply (Sheet 44) (40165)	4
5.9-13	Penetration M-74 Service Air Supply (Sheet 45) (40166)	5
5.9-13	Penetration M-75 Containment Cooling and Filtering Unit Return (Sheet 46) (40167)	6
5.9-13	Penetration M-76 Containment Cooling and Filtering Unit Return (Sheet 47) (40168)	8
5.9-13	Penetration M-77 Containment Cooling Unit Supply (Sheet 48) (40169)	6
5.9-13	Penetration M-78 Containment Cooling and Filtering Unit Supply (Sheet 49) (40170)	6
5.9-13	Penetration M-79 Fill & Makeup To Pressurizer Quench Tank (Sheet 50) (40171)	2
5.9-13	Penetration M-80 Demineralized Water Supply (Sheet 51) (40172)	3

Figure	Title	Revision
5.9-13	Penetration M-82 Containment Cooling and Filtering Unit Return (Sheet 52) (40173)	7
5.9-13	Penetration M-83 Containment Cooling Unit Return (Sheet 53) (40174)	7
5.9-13	Penetration M-84 Containment Cooling Unit Supply (Sheet 54) (40175)	6
5.9-13	Penetration M-85 Containment Cooling and Filtering Unit Supply (Sheet 55) (40176)	6
5.9-13	Penetration M-86 SI Containment Spray (Sheet 56) (40177)	5
5.9-13	Penetration M-87 Containment Purge Exhaust (Sheet 57) (40178)	3
5.9-13	Penetration M-88 Containment Purge Inlet (Sheet 58) (40179)	3
5.9-13	Penetration M-89 SI Containment Spray (Sheet 59) (40180)	5
5.9-13	Penetration M-91 Auxiliary Feedwater to Steam Generator RC-2B (Sheet 60) (40181)	6
5.9-13	Penetration M-93 Feedwater Supply (Sheet 61) (40182)	2
5.9-13	Penetration M-94 Main Steam From RC-2B (Sheet 62) (40183)	8
5.9-13	Penetration M-95 Main Steam From RC-2A (Sheet 63) (40184)	8
5.9-13	Penetration M-96 SG Feedwater To RC-2A, Supply (Sheet 64) (40185)	1
5.9-13	Penetration M-97 Auxiliary Feedwater to Steam Generator RC-2A (Sheet 65) (40186)	6
5.9-13	Penetrations: IA-3092, IA-3093 and IA-3094 Personnel Air Lock (Sheet 66) (42080)	2
5.9-14	Containment Structure Steel Liner, Sheet 1 (11405-S-2(16381))	14
5.9-15	Containment Structure Steel Liner, Sheet 2 (11405-S-2(16382))	16

Figure	Title	Revision
5.9-16	Developed Interior of Electrical Penetrations Viewed from Inside the Containment Building (36437)	1
5.10-1	Containment Structure Instrumentation (11405-S-13(16392))	8
5.10-2	Calculated Stresses at Containment Instrumentation (36532)	1
5.10-3	Deviation of Tendon Forces With Respect to Time (36533)	1
5.11-1	Section Through Engineered Safeguards Equipment Room (36534)	0
6.2-1	Elevation Through Safety Injection Suction Line (36535)	0
6.2-2	Safety Injection System Suction Piping Elevation Through Safety Injection Pump Room (36536)	0
6.2-3	Safety Injection Recirculation Inlet Strainer (36537)	1
6.4-1	Containment Air Cooling and Filtering System, Equipment Arrangement (36541)	1
6.4-2	Air Cooling and Filtering Unit (36542)	0
6.4-3	Air Cooling Unit (36543)	0
6.4-4	Pressure Relief Port (36544)	0
7.2-1	Reactor Protective System (36545)	0
7.2-2	Reactor Protective System Functional Diagram (E-23866-411-003(01582))	10
7.2-3	Typical Measurement Channel Functional Diagram (36547)	1
7.2-4	Nuclear Instrumentation System Functional Diagram (36548)	0
7.2-5	Low Flow Protective System Functional Diagram (36549)	1
7.2-6	TM/LP Trip Channel Block Diagram (40112)	3

Figure	Title	Revision
7.2-6a	Subcooled Margin Monitor Block Diagram (40113)	0
7.2-7	Custom Outline NFMS Ft. Calhoun (900388(44908))	0
7.2-8	Neutron Flux Monitoring System Power Range Channels (E-23866-411-103(1603))	4
7.2-9	Axial Power Distribution Trip System Functional Diagram (36552)	1
7.3-1	Functional Block Logic, Engineered Safeguard Signals (36553)	5
7.3-2	Simplified Master Diagram, Functional Circuit Logic, Engineered Safeguard Signals (Sheet 1) (36554)	3
7.3-2	Simplified Master Diagram Functional Circuit Logic Engineered Safeguard Signals (Sheet 2) (49900)	1
7.3-2	Simplified Master Diagram Functional Circuit Logic Engineered Safeguard Signals (Sheet 3) (49901)	0
7.3-3	Typical Matrix Supervision, Channel "A" or "B" (36555)	1
7.4-1	Reactor Regulating System Block Diagram (D-23866-413-001(1375))	8
7.4-2	CEA Position Setpoints (36557)	0
7.4-3	Pressure Control Program (36558)	1
7.4-4	Pressurizer Level Setpoint (36559)	1
7.4-5	Feedwater Control System Block Diagram (36560)	5
7.4-6	Steam Dump and Bypass System Block Diagram (36561)	0
7.5-1	Nuclear Detector Location (36562)	1
7.5-2	Configuration Drawing Highlighting ICCI System Inputs (37559)	1
7.6-1	Control Room Panels (36563)	20

Figure	Title	Revision
8.1-1	Simplified One Line Diagram Plant Electrical System (54187)	11
8.2-1	Electrical Network Interconnections (36564)	5
8.2-2	Transmission Line Routing (36565)	7
8.4-1	Sequence Starting of Engineered Safeguards, Both Diesels Starting (36566)	1
8.4-2	Sequence Starting of Engineered Safeguards, Both Diesels Starting (36567)	1
8.4-3	Auxiliary Building, Battery Rooms Elevation 1011' -0" (36568)	0
8.5-1	Cable and Conduit Schedule Notes (Sheet 1) (36569)	1
8.5-1	Cable and Conduit Schedule Notes (Sheet 2) (45962)	0
8.5-1	Cable and Conduit Schedule Notes (Sheet 3) (45963)	1
8.5-1	Cable and Conduit Schedule Notes (Sheet 4) (45964)	1
8.5-1	Cable and Conduit Schedule Notes (Sheet 5) (45965)	1
8.5-1	Cable and Conduit Schedule Notes (Sheet 6) (45966)	2
8.5-1	Cable and Conduit Schedule Notes (Sheet 7) (45967)	0
8.5-2	Cable Room Tray and Conduit Layout Plan, Elevation 1025' -0" and Sections (11405-E-78(12309))	63
8.5-3	Reactor Auxiliary Building Tray Conduit Layout Plan, Elevation 971'-0" (11405-E-60(12294))	21
8.5-4	Fire Detection System, Ground Floor Plan (D-4094(21430))	6
8.5-5	Fire Detection System, Basement Floor Plan, Elevation 995' -6" (D-4094(21431))	2
8.5-6	Fire Detection System, Auxiliary Building and Containment,	3

Figure	Title	Revision
	Elevation 1025' -0" (D-4094(21432))	
8.5-7	Fire Detection System, Operating Floor Plan, Elevation 1036' -0" (D-4094(21433))	6
8.5-8	Fire Detection System, Turbine Building, Elevation 990' -0" (D-4094(21434))	2
8.5-9	Fire Detection System, Turbine Building, Elevation 1011' -0" (D-4094(21435))	4
8.5-10	Fire Detection System, Turbine Floor, Elevation 1036'-0" (D-4094(21436))	2
8.5-11	Fire Protection System in the Technical Support Center and Intake Structure (D-4094(21437))	3
9.2-1	Chemical and Volume Control System Flow Schematic (36570)	1
9.2-3	Fort Calhoun MinBAST Level vs Stored BAST Concentration (45445)	0
9.2-4	Boric Acid Solubility in Water (45446)	0
9.3-1	Shutdown Cooling - Flow Diagram (36572)	1
9.5-1	Reactor Refueling Arrangement (36573)	0
9.8-1	Raw Water Pump Protection (36574)	0
9.8-2	Detail of Class I Piping at Entrance to Auxiliary Building (36575)	0
9.10-1	Containment Hydrogen Purge System (36576)	2
9.13-1	Equipment Layout, Auxiliary Building Sample Room (36577)	1
10.2-1	Full Power Heat Balance (40159)	0
11.1-1	Calculated Maximum Source Concentration Downstream of Fort Calhoun Station (36579)	0

Figure	Title	Revision
11.2-2	Fuel Assembly - Fission Product Decay Gammas (36585)	0
12.1-1	Station Organization Chart (36586)	13
12.1-2	Nuclear Assessments and Nuclear Operations-Organization Chart (36587)	12
12.1-3	Management Structure (36588)	10
12.1-4	Nuclear Engineering Division and Nuclear Support Services Division - Organization Chart (36589)	9
14.2-1	Typical CEA Withdrawal (Full Power), Core Power vs Time (36590)	4
14.2-2	Typical CEA Withdrawal (Full Power), Core Average Heat Flux vs Time (36591)	4
14.2-3	Typical CEA Withdrawal (Full Power), RCS Temperatures vs Time (36592)	4
14.2-4	Typical CEA Withdrawal (Full Power), RCS Pressure vs Time (36593)	4
14.2-5	Typical CEA Withdrawal (Zero Power), Core Power vs Time (36594)	4
14.2-6	Typical CEA Withdrawal (Zero Power), Core Average Heat Flux vs Time (36595)	4
14.2-7	Typical CEA Withdrawal (Zero Power), RCS Temperatures vs Time (36596)	4
14.2-8	Typical CEA Withdrawal (Zero Power), RCS Pressures vs Time (36597)	4
14.4-1	Full Length CEA Drop, Core Power vs Time (36598)	0
14.4-2	Full Length CEA Drop, Core Average Heat Flux vs Time (36599)	0
14.4-3	Full Length CEA Drop, RCS Temperatures vs Time (36600)	0

Figure	Title	Revision
14.4-4	Full Length CEA Drop, RCS Pressure vs Time (36601)	0
14.6-1	Loss of Coolant Flow, Core Flow Fraction vs Time (36603)	3
14.6-2	Loss of Coolant Flow, Core Power vs Time (36604)	3
14.6-3	Loss of Coolant Flow, Core Average Heat Flux vs Time (36605)	3
14.6-4	Loss of Coolant Flow, RCS Temperatures vs Time (36606)	3
14.6-5	Loss of Coolant Flow, RCS Pressure vs Time (36037)	3
14.6-6	RCP Speeds for Seized Rotor Event (48496)	0
14.6-7	Core Inlet and RCS Cold Leg Flow Rates for Seized Rotor Event (48497)	0
14.6-8	Vessel Outlet and Core Inlet Temperatures for Seized Rotor Event (48498)	0
14.6-9	Total Reactivitiy for Seized Rotor Event (48499)	0
14.6-10	Reactor Power and Average Rod Heat Flux for Seized Rotor Event (48502)	0
14.6-11	Pressurizer and Biased Core Outlet Pressures for Seized Rotor Event (48511)	0
14.8-1	Arrangement of Turbine Spent Fuel Pool and Main Steam Lines (40115)	0
14.8-2	Determination of Minimum Elevation Angle, 0's (40116)	0
14.8-3	Target Location (40117)	0
14.9-1	Loss of Load Incident (Case 1), Core Power vs Time (36607)	0
14.9-2	Loss of Load Incident (Case 1), Tavg vs Time (36608)	0
14.9-3	Loss of Load Incident (Case 1), Pressurizer Pressure vs Time (36609)	0

Figure	Title	Revision
14.9-4	Loss of Load Incident (Case 1), Steam Generator Pressure vs Time (36610)	0
14.9-5	Loss of Load Event, Core Power vs Time (36611)	2
14.9-6	Loss of Load Event, Tavg RCS Pressure vs Time (36612)	2
14.9-7	Loss of Load Event, Temperatures vs Time (36613)	2
14.9-8	Loss of Load Event, Steam Generator vs Time (36614)	2
14.9-9	Loss of Load/1 Steam Generator, Core Power vs Time (36615)	0
14.9-10	Loss of Load/1 Steam Generator, Core Average Heat Flux vs Time (36616)	0
14.9-11	Loss of Load/1 Steam Generator, Steam Generator Pressure vs Time (36617)	0
14.9-12	Loss of Load/1 Steam Generator, RCS Pressure vs Time (36618)	0
14.9-13	Loss of Load/1 Steam Generator, Tc vs Time (36036)	0
14.9-14	Loss of Load/1 Steam Generator, Average RCS Temperatures vs Time (36035)	0
14.10-1	Loss of Feedwater Flow, Core Power vs Time (36619)	2
14.10-2	Loss of Feedwater Flow, RCS Pressure vs Time (36620)	2
14.10-3	Loss of Feedwater Flow, RCS Temperatures vs Time (36621)	2
14.10-4	Loss of Feedwater Flow, Steam Generator Pressure vs Time (36622)	2
14.11-1	Excess Load Incident, Power vs Time (36623)	2
14.11-2	Excess Load Incident, Heat Flux vs Time (36624)	2
14.11-3	Excess Load Incident, Reactivity vs Time (36625)	2

Figure	Title	Revision
14.11-4	Excess Load Incident, Temperatures vs Time (36626)	2
14.11-5	Excess Load Incident, RCS Pressure vs Time (36627)	2
14.11-6	Excess Load Incident, Steam Generator Pressure vs Time (36628)	2
14.11-7	Excess Load Incident, Steam Generator Water Inventory vs Time (36629)	2
14.12-1	Cycle 19 Main Steam Line Break, Moderator Reactivity vs. Temperature (36630)	1
14.12-2	Cycle 19 HFP Steam Line Break, Core Power vs Time (36631)	1
14.12-3	Cycle 19 HFP Steam Line Break, Core Average Heat Flux vs Time (36632)	1
14.12-4	Cycle 19 HFP Steam Line Break, Reactivity Changes vs Time (36633)	1
14.12-5	Cycle 19 HFP Steam Line Break , Reactor Coolant Temperatures vs Time (36634)	1
14.12-6	Cycle 19 HFP Steam Line Break, Reactor Coolant Pressure vs Time (36635)	1
14.12-7	Cycle 19 HFP Steam Line Break, Steam Generator Pressure vs Time (36636)	1
14.12-8	Cycle 19 HZP Steam Line Break, Core Power vs Time (36637)	1
14.12-9	Cycle 19 HZP Steam Line Break, Core Average Heat Flux vs Time (36638)	1
14.12-10	Cycle 19 HZP Steam Line Break, Reactivity Changes vs Time (36639)	1
14.12-11	Cycle 19 HZP Steam Line Break, Reactor Coolant Temperatures vs Time (36640)	1

Figure	Title	Revision
14.12-12	Cycle 19 HZP Steam Line Break, Reactor Coolant Pressure vs Time (36641)	1
14.12-13	Cycle 19 HZP Steam Line Break, Steam Generator Pressure vs Time (36642)	1
14.13-1	Component and Total Reactivity Responses for the HFP BOC CEA Ejection Case (48513)	0
14.13-2	Core Power Response for the HFP BOC CEA Ejection Case	0
14.13-3	Core Average Heat Flux Response for the HFP BOC CEA Ejection Case (48521)	0
14.13-4	System Pressure Responses for the HFP BOC CEA Junction Case (48522)	0
14.13-5	System Fluid Temperature Responses for the HFP BOC CEA Ejection Case (48523)	0
14.15-41	Normalized Power: 1.0 DECLG EOC NOLPSI (48524)	0
14.15-42	SIT Discharge Rates: 1.0 DECLG EOC NPLPSI (48525)	0
14.15-43	HPSI Flow Rates: 1.0 DECLG EOC NOLPSI (48535)	0
14.15-44	LPSI Flow Rates: 1.0 DECLG EOC NOLPSI (48536)	0
14.15-45	Upper Plenum Pressure: 1.0 DECLG EOC NOLPSI (48537)	0
14.15-46	Total Break Flow Rate: 1.0 DECLG EOC NOLPSI (48753)	0
14.15-47	Average Core Inlet Flow Rate: 1.0 DECLG EOC NOLPSI (48754)	0
14.15-48	Hot Assembly Inlet Flow Rate: 1.0 DECLG EOC NOLPSI (48755)	0
14.15-49	PCT - Node Fluid Quality: 1.0 DECLG EOC NOLPSI (48756)	0
14.15-50	PCT - Node Average Fuel, Cladding Surface, and Fluid Temperatures: 1.0 DECLG EOC NOLPSI (48757)	0

Figure	Title	Revision
14.15-51	PCT - Node Heat Transfer Coefficient: 1.0 DECLG EOC NOLPSI (48758)	0
14.15-52	PCT - Node Heat Flux: 1.0 DECLG EOC NOLPSI (48759)	0
14.15-53	Containment Pressure: 1.0 DECLG EOC NOLPSI (48760)	0
14.15-54	Upper Plenum Pressure After Blowdown: 1.0 DECLG EOC NOLPSI (48761)	0
14.15-55	Downcomer Mixture Level: 1.0 DECLG EOC NOLPSI (48763)	0
14.15-56	Core Effective Flooding Rate: 1.0 DECLG EOC NOLPSI (48765)	0
14.15-57	Core Collapsed Liquid Level: 1.0 DECLG EOC NOLPSI (48766)	0
14.15-58	Core Quench Level: 1.0 DECLG EOC NOLPSI (48768)	0
14.15-59	PCT - Node Heat Transfer Coefficient: 1.0 DECLG EOC NOLPSI (48771)	0
14.15-60	Peak Cladding Temperature: 1.0 DECLG EOC NOLPSI (48772)	0
14.15-61	Pressurizer and Steam Generator Pressures for 0.049 Ft^2 Break Case (48773)	0
14.15-62	Break Mass Flow Rate for 0.049 Ft^2 Break Case (48776)	0
14.15-63	RCS Mass Inventory for 0.49 Ft^2 Break Case (48777)	0
14.15-64	T00DEE2 Hot Channel Mixture Level for 0.49 Ft^2 Break Case (48778)	0
14.15-65	Total HPSI Mass Flow Rate for 0.49 Ft^2 Break Case (48779)	0
14.15-66	Total SIT Mass Flow Rate for 0.049 Ft^2 Break Case (48781)	0
14.15-67	AFW Mass Flow Rate for 0.049 Ft^2 Break Case (48782)	0
14.15-68	T00DEE2 PCT Node and Rupture Node Cladding Temperatures for 0.049 Ft^2 Break Case (48783)	0

Figure	Title	Revision
14.16-1	Containment Pressure Response - Loss of Coolant Accident (36746)	2
14.16-2	Long-Term Pressure Response - Loss of Coolant Accident (36747)	2
14.16-3	Containment Pressure Response - Main Steam Line Break (36748)	1
14.16-4	Containment Temperature Response - Main Steam Line Break (36749)	1
14.17-1	Hydrogen Concentration vs Time (36752)	1
14.18-1	Radial Peaking Factor vs Burnup (45465)	0
14.22-1	RCS Depressurization Incident, Core Power vs Time (36753)	0
14.22-2	RCS Depressurization Incident, Core Average Heat Flux vs Time (36754)	0
14.22-3	RCS Depressurization Incident, Coolant Temperature vs Time (36755)	0
14.22-4	RCS Depressurization Incident, Pressurizer Pressure vs Time (36756)	0
14.24-1	Containment Leak Rate vs Number of Damaged Assemblies (44898)	0
A-1	Nuclear Quality Assurance Program Organization for Fort Calhoun Station (45442)	0
F-1	Response Spectra, Design Earthquake (36758)	0
F-2	Response Spectra, Maximum Hypothetical Earthquake (36759)	0
F-3	Foundation Spring Constants - Mass Data (36760)	0
F-4	Containment Structure - Absolute Accelerations (36761)	0
F-5	Mode Shape Five Mass System Reactor and Aux Building (36762)	0

Figure	Title	Revision
F-6	Mode Shape Five Mass System Reactor and Aux Building (36763)	0
F-7	Mode Shape Five Mass System Reactor and Aux Building (36764)	0
F-8	Mode Shape Five Mass System Reactor and Aux Building (36765)	0
F-9	Mode Shape Five Mass System Reactor and Aux Building (36766)	0
F-10	Mode Shape Five Mass System Reactor and Aux Building (36767)	0
F-11	Mode Shape Five Mass System Reactor and Aux Building (36768)	0
F-12	Equipment Responses (36769)	0
F-13	Equipment Responses (36770)	0
F-14	Equipment Responses (36771)	0
F-15	Equipment Responses (36772)	0
F-16	Equipment Responses (36773)	0
F-17	Equipment Responses (36774)	0
F-18	Floor Response Spectrum (36775)	0
F-19	Floor Response Spectrum (36776)	0
F-20	Floor Response Spectrum (36777)	0
F-21	Floor Response Spectrum (36778)	0
F-22	Floor Response Spectrum (36779)	0
F-23	Floor Response Spectrum (36780)	0
F-24	Floor Response Spectrum (36781)	0
F-25	Floor Response Spectrum (36782)	0

Figure	Title	Revision
F-26	Floor Response Spectrum (36783)	0
F-27	Floor Response Spectrum (36784)	0
F-28	Horizontal and Vertical Absolute Acceleration Induced by Design Earthquake (OBE) (XB-545-S-38(20764))	0
F-29	Horizontal and Vertical Absolute Accelerations Induced by Maximum Hypothetical Earthquake (DBE) (XB-545-S-39(20765))	0
I.3-1	Start-Up and Testing Organization (36785)	0
I.3-2	GHDR Start-Up and Testing Organization (36786)	0
I.3-3	CE Start-Up and Testing Site Organization (36787)	0
I.3-4	OPPD Support Staff (36788)	0
M-1	High Energy Lines Operating Floor (36789)	0
M-2	High Energy Lines Ground Floor (36790)	0
M-3	Main Steam Line Piping Isometric (36791)	0
M-4	Feedwater Line Piping Isometric (36792)	0
M-5	Stress-Analysis Main Steam in Auxiliary Building (36793)	0
M-6	Stress-Analysis Main Steam in Turbine Building (36794)	0
M-7	Stress Analysis-Feedwater (36795)	0
M-8	Steam Flow into Room 81 (36796)	1
M-9	Pressure Build-Up in Room 81 (36797)	0
M-10	Water Flow Rate From Feedwater Line Break (36798)	1
M-11	Water Level Time History in Room 81 (36799)	0

Figure	Title	Revision
M-12	Auxiliary Building Feedwater and Steam Pipe Restraints (36800)	0
M-13	Main Steam and Feedwater Protective Enclosures (36801)	0

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SECTION 1

INTRODUCTION AND SUMMARY

Table of Contents

1. INTRODUCTION AND SUMMARY 1
1.1 INTRODUCTION 1
1.2 SUMMARY PLANT DESCRIPTION 1
1.2.1 Plant Site 1
1.2.2 Plant Arrangement 2
1.2.3 Reactor and Reactor Coolant System 3
1.2.4 Containment 4
1.2.5 Engineered Safeguards Systems 5
1.2.6 Instrumentation and Control 8
1.2.6.1 Control 8
1.2.6.2 Instrumentation 9
1.2.6.3 Plant Computer 10
1.2.7 Electrical Systems 11
1.2.8 Auxiliary Systems 12
1.2.8.1 Chemical and Volume Control System 12
1.2.8.2 Shutdown Cooling System 13
1.2.8.3 Component Cooling and Raw Water Systems 14
1.2.8.4 Fuel Handling and Storage 14
1.2.8.5 Sampling Systems 15
1.2.8.6 Circulating and Turbine Plant Cooling Water Systems 15
1.2.8.7 Plant Ventilation Systems 15
1.2.8.8 Plant Fire Protection 16
1.2.8.9 Auxiliary Steam System 16
1.2.8.10 Compressed Air System 16
1.2.9 Steam and Power Conversion Systems 16
1.2.10 Radioactive Waste Disposal System 17
1.2.11 Plant Design and Operating Characteristics 18
1.3 PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA 1
1.3.1 Station Design 1
1.3.2 Reactor 1
1.3.3 Reactor Coolant and Auxiliary Systems 2
1.3.4 Containment 3
1.3.5 Engineered Safeguards 3
1.3.6 Nil Ductility Transition Temperature 4
1.3.7 Instrumentation and Control 4
1.3.8 Electrical Systems 4
1.3.9 Radioactive Wastes and Radiation Protection 4
1.4 RESEARCH AND DEVELOPMENT REQUIREMENTS 1
1.4.1 Introduction 1
1.4.2 Moderator Temperature Coefficient 1
1.4.3 Control Element Assembly (CEA) Testing 2
1.4.3.1 Critical Experiments 2
1.4.3.2 Control Element Assembly Testing 3

1.4.4 Control Element Drive Mechanism (CEDM) Testing 5
1.4.5 Fuel Assembly Design 7
1.4.5.1 Prototype Tests 7
1.4.5.2 Spacer Grid Pressure Drop Tests 7
1.4.5.3 Fuel Assembly Flow Distribution Tests 8
1.4.5.4 DNB Testing in the Fuel Assembly 9
1.4.5.5 Fuel Assembly Testing 11
1.4.6 Reactor Vessel Flow Tests 12
1.4.7 Fuel Rod Cladding 13
1.4.8 Engineered Safeguards Equipment Testing 15
1.5 ITEMS OF SPECIAL INTEREST TO ACRS 1
1.5.1 Introduction 1
1.5.2 Quality Assurance and In-Service Inspectability 1
1.5.3 Instrumentation for the Detection of Failed Fuel 2
1.5.4 Reactor Vessel Thermal Shock 2
1.5.5 Blowdown Forces on Core and Reactor Coolant System Components . . 4
1.5.6 Effect of Fuel Rod Failure on ECCS Performance 5
1.5.7 Effect of Fuel Inlet Blockage 8
1.6 SIGNIFICANT DESIGN CHANGES SINCE AWARD OF CONSTRUCTION PERMIT 1
1.6.1 Introduction 1
1.6.2 Safety Injection Actuation Signal 1
1.6.3 Reactor Vessel Surveillance Program 1
1.6.4 Engineered Safeguards Pump Radiography 1
1.6.5 Secondary CEA Position Indication 2
1.6.6 Letdown Flow Control in Chemical and Volume Control System (CVCS) 2
1.6.7 On-Site Diesel Oil Storage Capacity 3
1.6.8 Number of Raw Water Pumps 3
1.6.9 Charcoal Filter Spray Dousing System 3
1.6.10 Containment Emergency Access Lock 3
1.6.11 Containment Air Cooling System Ductwork 4
1.6.12 Fire Protection Water Supply 4
1.6.13 Reactor Coolant Ion Exchange Equipment 4
1.6.14 Analytical Model of Containment Liner 4
1.6.15 Ventilation Discharge Duct 5
1.6.16 Electrical Equipment Environmental Qualification - Harsh Environment . . 5
1.6.17 Inadequate Core Cooling Instrumentation 6
1.6.18 Diverse Scram System 6
1.6.19 Steam Generator Orifice Plate Removal 6
1.7 IDENTIFICATION OF CONTRACTORS 1
1.8 TECHNICAL QUALIFICATIONS OF THE APPLICANT 1
1.9 DESIGN COMPARISON WITH OTHER PLANTS 1
1.9.1 Significant Differences 1
1.9.2 Plant Design Parameters and Comparison 2

1.10 GENERAL CONCLUSIONS 1
1.11 SPECIFIC REFERENCES 1
1.12 GENERAL REFERENCES 1

List of Tables

Table 1.2-1 - "Principal Plant Design and Operating Characteristics" 20
Table 1.4-1 - "Analysis of Moderator Temperature Coefficients in the Connecticut Yankee Reactor at Start-Of-Life" 2
Table 1.4-3 - "Comparison of Experimental and Analytical Heat Removal Characteristics"17
Table 1.4-4 - "Cooling Coil Heat Balance Comparison Between Heat Gained by Water and Heat Lost by Mixture" 19
Table 1.7-1 - "Contractors for Fort Calhoun-Unit No. 1" 2
Table 1.9-1 - "Comparison of Plant Parameters" 3

List of Figures

The following figures are controlled drawings and can be viewed and printed from the applicable listed aperture card.

Figure No.	Title	Aperture Card
1.2-1	Site Plan	36030
1.2-2	Restricted Area	36031
1.2-3	Primary Plant South Elevation	12167
1.2-4	Primary Plant West Elevation	12168
1.2-5	Primary Plant North Elevation	12169
1.2-6	Primary Plant Roof Plan and Upper Levels	12166
1.2-7	Turbine and Office Building Cross Sections	12195
1.2-8	Intake Structure Floor Plans	12206
1.2-9	Intake Structure Building Sections and Panel Details	12209
1.2-10	Intake Structure and Tunnels Foundation Plan and Details	16531
1.2-11	Intake Structure and Tunnels Plan at Elevation 985'-0 and 993'0" and Details	16532
1.2-12	Intake Structure and Tunnels Plan at Elevation 1007'-6" and Details	16533
1.2-13	Intake Structure and Tunnels Sections and Details	16535
1.2-14	Intake Structure and Tunnels Trash Rack and Stop Log Details	16538
1.2-15	Intake Structure and Tunnels Miscellaneous Details I	16539
1.2-16	Intake Structure and Tunnels Miscellaneous Details II	16540
1.4-1	Containment Fan - Motors Accident Environment Performance Test Cycle	36034
1.5-1	Perforation Temperature vs Internal Gas Pressure for Zircaloy - 4 Fuel Tubing Tested in Helium Atmosphere	36039
1.5-2	Perforation Temperature vs Heating Rate for Zircaloy -4 Fuel Tubing Tested at 2500 psi in Helium	36040
1.5-3	Circumferential Strain vs Heating Rate for Zircaloy -4 Tubing Tested in Helium	36041

1.2 SUMMARY PLANT DESCRIPTION

1.2.1 Plant Site

The site for the Fort Calhoun Station contains 660.46 acres on the west bank of the Missouri River, approximately 19.4 miles north of Omaha, Nebraska. OPPD has a perpetual easement on 582.18 acres of land on the east bank of the river directly opposite the plant buildings. About 85 percent of the site area is on relatively level ground located in the alluvial plain of the river. On the Western part of the site the ground rises sharply about 60 feet to a higher level area which is bounded on the west by U.S. Highway 75, formerly U.S. Highway 73.

The U.S. Corps of Engineers' river improvement program has led to the development of a stable, well defined river bank in the vicinity of the site. A partially filled-in area, about 450 feet wide, runs parallel to the bank.

The plant buildings are supported by a system of pipe piles which were driven to bedrock approximately 60 feet below the surface. Ground water level is close to existing grade and structures below grade are suitably water-proofed.

The area adjoining the site is farmland and sparsely populated. The minimum exclusion distance is 910 meters (2986 feet). The nearest population center area of more than 25,000 is formed by adjacent cities of Omaha, Nebraska and Council Bluffs, Iowa.

The exclusion zone, as defined in Section 100.3(a) of 10 CFR 100 consists of the 660.46 acres owned by Omaha Public Power District and the 582.18 acres directly across the Missouri River. The restricted area as defined in Section 20.3(a)(14) of 10 CFR Part 20 is shown on Figure 1.2-2.

A rail spur from the Chicago and Northwestern Railway was constructed to serve the construction of the Fort Calhoun Station. The rail line serving this spur has since been removed. Road access is provided from U.S. Highway No. 75. The reactor vessel and steam generators were shipped to the site by barge; other items were transported by rail or truck.

1.2.2 Plant Arrangement

The principal plant structures are the containment building, the auxiliary building, the turbine building, the service building, the technical support center, the maintenance shop, the Radioactive Waste Processing Building and the Chemistry and Radiation Protection Locker facility, Office/Cafeteria Addition. A hazardous material storage building is located to the west of the main buildings.

Elevations of these structures are shown in Figures 1.2-3 to 1.2-5 inclusive. The circulating water pump intake structure is a separate building located at the river bank to the east of the office building.

The reactor, steam generators, reactor coolant pumps and pressurizer are located in the containment, together with other nuclear steam supply system (NSSS) components which do not require visual observation or direct attention by the operator during power operation.

An air lock sized for personnel access and passage of small components connects to the auxiliary building. An equipment access hatch is located at the 1013 foot level and connects to a handling area within the auxiliary building. A polar bridge crane is installed above the operating floor to handle the reactor vessel head and reactor internals during refueling and for maintenance operations within the reactor containment.

The reactor auxiliaries including waste treatment facilities, certain engineered safeguards components, the central control room, reactor plant locker rooms and laboratories, emergency diesel generators and fuel handling and storage facilities are located in the auxiliary building. A traveling bridge crane is located over the fuel handling and storage area.

The layout of the containment and auxiliary buildings is shown in P&ID 11405-A-5 thru 11405-A-8, Figure 1.2-6, P&ID 11405-A-13 and 11405-A-14.

The turbine and service buildings house the turbine generator, condenser, condensate and feedwater pumps, feedwater heaters, other turbine heat cycle components, auxiliary boiler and conventional auxiliaries such as turbine lube oil conditioning equipment. A 150 ton traveling bridge crane is installed and serves the entire high bay portion of the building. The layout of the turbine building and the adjacent office building is shown in P&ID 11405-A-257 thru 11405-A-261, 11405-A-263, and Figure 1.2-7.

The condenser cooling water and raw water pumps are located in the intake structure as shown in Figures 1.2-8, 1.2-9 and P&ID 11405-A-279.

Structural features of the intake structure are shown in Figures 1.2-10 through 1.2-16.

The Radioactive Waste Processing Building contains provisions for dry active waste (DAW) processing, liquid radwaste filtration and ion exchange (FIX) equipment and radwaste solidification equipment.

1.2.3 Reactor and Reactor Coolant System

The system consists of a pressurized water reactor and its associated coolant system, arranged as two closed loops each containing two reactor coolant pumps and a steam generator connected in parallel to the reactor. An electrically heated pressurizer is connected to one of the loops. The system is designed to be operated at a core thermal power of 1500 MWt to provide steam at a nominal 815 psia.

The reactor core is fueled with uranium dioxide pellets enclosed in zircaloy/ZIRLO Tubes with welded end plugs. The tubes are fabricated into assemblies in which end fittings limit axial motion and grids limit lateral motion of the tubes. The control element assemblies (CEA's) consist of Ni-Cr-Fe alloy clad boron carbide absorber rods which are guided by tubes located within the fuel assembly. The core consists of 133 fuel assemblies with an active fuel length of 128 in.

The reactor vessel and its closure head are fabricated from low alloy manganese-molybdenum steel internally clad with stainless steel.

The internal structures include the core support barrel, the core support plate, the core shroud, the thermal shield, the upper guide structure assembly, and the incore instrument thimble assembly. The core support barrel is a right circular cylinder supported from a ring flange from a ledge on the reactor vessel. It carries the entire weight of the core. The core plate transmits the weight of the core to the core support barrel by means of vertical columns and a beam structure. The core shroud surrounds the core and minimizes the amount of bypass flow. The upper guide structure provides a flow shroud for the CEA's and prevents upward motion of the fuel assemblies during pressure transients. Lateral motion limiters are provided at the lower end of the core support barrel assembly.

The reactor coolant system is arranged as two closed loops connected in parallel to the reactor vessel. Each loop consists of one 32 in. ID outlet (hot) pipe, one steam generator, two 24 in. ID inlet (cold) pipes and two pumps. An electrically heated pressurizer is connected to one of the outlet pipes and a safety injection line is connected to each of the four inlet legs.

Over-pressure protection is provided by power-operated relief valves and spring-loaded safety valves connected to the pressurizer. The discharge from the pressurizer safety and relief valves is released under water in the pressurizer quench tank, where it is condensed and cooled. In the event the discharged steam exceeds the capacity of the tank, the tank relieves to the containment atmosphere.

The reactor coolant system operates at a nominal pressure of 2100 psia. The coolant enters near the top of the reactor vessel, then flows downward between the reactor vessel shell and the core support barrel into the plenum. It then flows upward through the core, leaves the reactor vessel, and flows through the tube side of the two vertical U-tube steam generators where heat is transferred to the secondary system. Reactor coolant pumps return the reactor coolant to the reactor vessel.

The two steam generators are vertical shell and U-tube units each producing 3.31×10^6 lb/hr of steam at a nominal 815 psia. The steam generated in the shell side of the steam generator flows upward through moisture separators which reduce its moisture content to less than 0.2 weight percent. All major surfaces in contact with the reactor coolant are either stainless steel or Ni-Cr-Fe alloy in order to maintain reactor coolant purity.

The reactor coolant is circulated by four electric motor driven, single-suction, non-reversing centrifugal pumps. The pump shafts are sealed by mechanical seals. The seal performance is monitored by pressure and temperature sensing devices in the seal system.

1.2.4 Containment

The reactor containment building provides a highly reliable, essentially leak-tight barrier against the escape of radioactivity which might be released from the reactor system in the event of an accident. It is a concrete structure in the form of a vertical cylinder with domed roof and a flat base.

The cylinder and dome are of post-tensioned concrete and the base is of reinforced concrete construction. A continuous carbon steel liner is included for leak tightness. Inside the containment structure, the reactor and other NSSS components are shielded with concrete.

Facilities are provided for pressure and leak rate testing of the entire containment system including provisions for leak rate testing of individual piping and electrical penetrations.

Limited access to the containment is permitted during reactor operation.

1.2.5 Engineered Safeguards Systems

Engineered safeguards systems in conjunction with the containment protect the public and plant personnel in the highly unlikely event of an accidental release of radioactive fission products from the reactor system, particularly as the result of loss-of-coolant accidents. The Engineered Safeguards include Engineered Safety Features systems, Essential Auxiliary Support systems and components, and Engineered Safeguards Controls and Instrumentation. These safeguards function to localize, control, mitigate, and terminate such accidents to hold exposure levels below applicable guidelines.

The engineered safeguards systems are:

- The safety injection system (including high pressure and low pressure safety injection pumps and the safety injection tanks)
- The chemical and volume control system
- The containment spray system
- The containment air cooling and filtering system
- The containment hydrogen purge system
- The auxiliary feedwater system
- The containment isolation system

Four safety injection tanks are provided, each connected to one of the four reactor inlet lines. Each tank has a volume of approximately 1300 ft^3 containing borated water at refueling concentration and are pressurized with nitrogen at 240 psig minimum (255 psig nominal). In the event of a loss-of-coolant accident, the borated water is forced into the reactor coolant system by the expansion of the nitrogen. The water in three tanks adequately cools the entire core. In addition, borated water is injected into the same nozzles by two low pressure and three high pressure injection pumps taking suction from the safety injection and refueling water storage (SIRW) tank.

For maximum reliability, the design capacity of the combined operation of one high pressure and one low pressure pump provides adequate injection flow for any loss of coolant accident; in the event of a design basis accident (DBA) and loss of offsite power, at least one high pressure and one low pressure pump receive power from the emergency power sources. Upon depletion of the storage tank supply, the high pressure pump suction automatically transfers to the containment recirculation inlet of elevation 994'-0" and the low pressure pumps are shut down. One high pressure pump has sufficient capacity to cool the core adequately at the start of recirculation. During recirculation, heat in the recirculating water is removed in the shutdown cooling heat exchangers by the operation of the containment spray system. Further, the suction of the high pressure safety injection pumps may be manually aligned to inject subcooled water from the shutdown cooling heat exchangers directly into the reactor coolant system for core cooling.

Both low and high pressure injection pumps are located outside the containment building to permit access for periodic testing during normal operation. The pumps discharge into separate headers which lead to the containment. Test lines are provided to permit running the pumps for test purposes during plant operation. The safety injection system is further described in Section 6.2.

The chemical and volume control system is described in Section 9.2.

The containment spray system supplies borated water to cool and reduce the pressure of the containment atmosphere. The pumps take suction initially from the SIRW tank. Long term cooling is based on suction from the containment recirculation inlet at elevation 994'-0". In the recirculation mode of operation, heat is transferred from the recirculating borated water through the shutdown cooling heat exchangers to the component cooling system and ultimately to the river water via the component cooling water heat exchangers. The containment spray system is further described in Section 6.3.

The containment air cooling and filtering system is designed to provide capability for mixing and cooling the containment atmosphere. The cooling coils and fans provide containment cooling at DBA conditions. The heat is transferred to the component cooling system. Two of the four units are equipped with both particulate filters and impregnated activated charcoal adsorbers. The containment air cooling and filtering removal system is further described in Section 6.4.

The containment hydrogen purge system is described in Section 9.10.2.5.

The auxiliary feedwater system is described in Section 9.4.

The containment isolation system is described in Sections 5.9 and 7.3.2.5.

The essential auxiliary support systems include part or all of the following:

- The normal station electrical power distribution system
- The emergency power distribution system
- The instrument air system
- The auxiliary building HVAC system
- The control room HVAC
- The component cooling water system
- The raw water system

Instrumentation and controls for the engineered safety features and essential auxiliary support systems are also classified as engineered safeguards systems.

1.2.6 Instrumentation and Control

1.2.6.1 Control

The reactor control system provides for startup and shutdown of the reactor and for adjustment of the reactor power in response to turbine load demand. The nuclear steam system is capable of following a ramp change from 15 percent to 100 percent power at a rate of 10 percent per minute and at greater rates over small load increments up to a step change of 10 percent. This control is normally accomplished by manual control of CEA movement.

The reactor is controlled by a combination of the CEAs and dissolved boric acid in the reactor coolant. Boric acid is used for reactivity changes associated with large but gradual changes in water temperature, xenon effects and fuel burnup. Additions of boric acid also provide an increased shutdown margin during the core loading and subsequent refueling.

CEA movement provides changes in reactivity for shutdown or power changes. The CEAs are actuated by control drive mechanisms mounted on the reactor vessel head. The control element drive mechanisms (CEDMs) are designed to permit rapid insertion of the CEAs into the reactor core by gravity. CEA motion can be initiated manually or automatically in instances of reactor trip.

The pressure in the reactor coolant system is controlled by regulating the temperature of the coolant in the pressurizer, where steam and water are held in thermal equilibrium. Steam is formed by the pressurizer heaters or condensed by the pressurizer spray to reduce variations caused by expansion and contraction of the reactor coolant due to reactor system temperature changes.

1.2.6.2 Instrumentation

The nuclear instrumentation includes out-of-core and in-core neutron flux detectors. Ten channels of out-of-core instrumentation monitor the neutron flux and provide reactor protection and control signals during startup and power operation. Four of the channels follow the neutron flux through the source and startup ranges, and six channels follow the neutron flux from within the startup range through the full power range. Of the latter, four are used for reactor protection and two for reactor control.

The in-core monitors consist of self-powered rhodium neutron detectors and thermocouples to provide information on neutron flux distribution and temperatures in the core.

Reactor vessel level (or inventory) and temperature signals are provided by redundant in-vessel sensors.

The reactor parameters are maintained within the acceptable limits by the inherent characteristics of the reactor, by the CEA system, by boron in the moderator and by the operating procedures. Departures from these limits are audibly and visually indicated in the control room. In addition, in order to preclude unsafe conditions for plant equipment or personnel, the reactor protective system initiates reactor shutdown if selected parameters reach their preset limits. Four independent channels monitor each of the selected plant parameters. The reactor protective system logic is designed to initiate a reactor trip whenever the signal of any two of four channels monitoring the same plant parameter reaches the preset limit. The reactor trip signal functions to de-energize the CEDM clutches, releasing the CEA's. Redundancy is provided in all parts of the reactor protective system to assure that no single failure will prevent a reactor trip.

The process instrumentation monitoring system includes those critical channels which are used for protective action. Additional temperature, pressure, flow and liquid level monitoring is provided, as required, to keep the operating personnel informed of plant conditions, and to provide information from which plant processes can be evaluated and/or regulated.

Instrument signals penetrating the containment are electrical. Instrument signal transmission for the remaining plant instruments is either electric or pneumatic depending on the function to be served.

The plant gaseous and liquid effluents are monitored for radioactivity. Activity levels are displayed and off-normal values are annunciated. Area monitoring stations are provided to measure radioactivity at selected locations in the plant.

High pressure within the containment initiates closure of all automatic containment isolation valves and dampers except those valves required for the operation of engineered safeguards equipment or other equipment such as cooling water to the reactor coolant pump seals. High radiation conditions within the containment initiate closure of the ventilation system isolation dampers.

1.2.6.3 Plant Computer

The plant computer is a real time digital processing system, the basic functions of which are the assimilation of plant data and the surveillance of plant parameters. The computer also provides supplementary information to aid in plant operation and provide positive indication of off-normal conditions as well as the processing and display for the Safety Parameter Display System (SPDS).

The computer scans approximately 700 analog inputs of flow, temperature, pressure and other process parameters, and about 500 digital inputs registering valve positions, circuit breaker and switch conditions, and similar items. The interaction with the reactor control system is described in Section 7.

1.2.7 Electrical Systems

The main generator is a conductor-cooled unit rated at 590,000 kVA at 0.85 power factor. The generator is connected via a force-cooled, isolated-phase bus to a single, three-phase, 22-345-kV step-up transformer, connected in turn by a short overhead circuit to OPPD's 345-kV substation Number 3451 which is located on the plant site and is part of the Mid-Continent Area Power Pool (MAPP) network. The generator main leads are provided with a motor-operated disconnect switch to permit separation of the generator from the system and back-feeding of energy from the 345 kV system to the plant auxiliary power system.

Other sources of energy for plant auxiliaries are 161 kV circuits from several OPPD Substations, and two 2500 kW, diesel-generator sets located in the plant.

Normal supply for the plant auxiliary power system is from two unit auxiliary transformers and two house service transformers. Any two of the four transformers can supply maximum plant demand continuously.

A 4.16-kV system is provided and consists of four main buses with a flexible supply-switching scheme. Two buses supply unit loads; the other two buses, each connected to one of the emergency diesel-generator units, supply 4.16-kV and 480-Volt station auxiliaries and engineered safeguards. The 480-Volt loads are supplied through six 4,160-480-Volt transformers connected to three double-ended 480-Volt load centers. Each load center is arranged with three bus sections; the center (mid-tie) bus section can be connected to either adjacent bus section. This arrangement provides a flexible and diverse supply system for low-voltage essential auxiliaries and engineered safeguards.

Two independent, 125-Volt, d-c control energy systems are provided. Each comprises a lead-calcium control battery, battery charger, main bus, and distribution apparatus. A third battery charger is provided, available to either system as a backup unit.

Four independently supplied, 120-Volt a-c, main instrument buses are provided to meet requirements of logic schemes used for the reactor protective system and for other instrumentation and control systems for which reliability is particularly desirable. Each of these four buses is normally supplied by a 125-Volt d-c/120-V a-c inverter.

A 13.8 kV circuit derived from the 161 kV circuit serves as a limited emergency power source of 480 V power to 1B3C. This 480 V power source has the capacity to maintain the plant in a stable condition until other power souces can be reestablished.

1.2.8 Auxiliary Systems

1.2.8.1 Chemical and Volume Control System

The purity level in the reactor coolant system is controlled by continuous purification of a bypass stream of reactor coolant. Water removed from the reactor coolant system is cooled in the regenerative heat exchanger. From there the coolant flows to the letdown heat exchanger and then through a filter where corrosion and fission products are removed. It is then sprayed into the volume control tank, and returned to the reactor coolant system by the charging pumps.

The volume control system automatically controls the rate and amount of coolant returned to the reactor coolant system to maintain the pressurizer level within the prescribed control band thereby compensating for changes in volume due to coolant temperature changes and reactor coolant pump shaft controlled seal leakage. The volume control tank is sized to accommodate coolant inventory changes resulting from load changes from hot standby to full power. Using the volume control tank as a surge tank decreases the quantity of liquid and gaseous wastes which otherwise would be generated.

Make-up water is taken from the demineralized water storage system and from the two concentrated boric acid tanks. The boric acid solution in these tanks is prepared and stored at a temperature sufficiently high to prevent crystallization. The make-up water is pumped through the regenerative heat exchanger into the reactor coolant loop by the charging pumps.

Bleed from the reactor coolant system during a boron concentration reduction is routed to the radioactive waste disposal system for subsequent processing and disposal to the river. Two deborating ion exchangers are provided which may be used late in cycle life, when the boric acid concentration has been reduced to the point at which further reduction by feed and bleed is inefficient.

Chemical injection equipment is provided for the addition of corrosion control chemicals to the reactor coolant.

1.2.8.2 Shutdown Cooling System

The shutdown cooling system is used to reduce the temperature of the reactor coolant at a controlled rate from 300°F to a refueling temperature and to maintain the proper reactor coolant temperature during refueling. While the plant is shutdown, it also provides an emergency backup for the spent fuel pool cooling system in the event of failure of that system.

The shutdown cooling system utilizes the low pressure safety injection pumps to circulate the reactor coolant through the two shutdown heat exchangers, returning it to the reactor coolant system through the low pressure injection header.

Emergency shutdown cooling during a loss of all normal electrical power sources can be accomplished by allowing natural circulation of the reactor coolant to transfer heat from the core to the steam generators. The steam that is generated is released to the atmosphere as required. An auxiliary electric-driven feedwater pump operating from one emergency diesel generator or an auxiliary turbine-driven feedwater pump supply feedwater to the steam generators during this period. The emergency feedwater storage tank supplies these pumps and is sufficient for 8 hours of cooling. A non safety class diesel engine driven auxiliary feedwater pump is also available. This pump takes suction from the condensate storage tank. This is adequate to maintain the core in a safe condition and reduce the reactor coolant temperature and pressure to permit initiation of the shutdown cooling system.

The component cooling system supplies cooling water for the shutdown cooling heat exchangers.

1.2.8.3 Component Cooling and Raw Water Systems

The component cooling system consists of a number of parallel closed loops which remove heat from the various auxiliary systems, some of which contain reactor coolant. The system provides cooling at the shutdown heat exchangers, the spent fuel pool heat exchanger, the letdown heat exchanger, the sampling heat exchanger, ventilation equipment and various pumps. Heat removed by the component cooling water system is transferred to the raw water system by the component cooling heat exchangers. The raw water system is a once through system operating with screened river water. Redundancy has been provided in the component cooling and raw water systems to provide for both normal and emergency operation with pumping and heat exchange equipment out of service. Further, the system arrangement permits the raw water to be circulated through portions of the component cooling system piping to provide direct cooling of vital engineered safeguards components in the unlikely event of all of the component cooling pumps and heat exchangers being unavailable to fulfill their design function.

1.2.8.4 Fuel Handling and Storage

New fuel is stored dry in vertical racks within a storage area in the auxiliary building. Room is provided for storing one-third of a core. The storage area construction and fuel assembly spacing preclude criticality even in the event of accidental flooding with unborated water.

The spent fuel pool is a reinforced concrete structure lined with stainless steel which provides storage capacity for 1083 fuel assemblies. In region 1, spent fuel assemblies are stored in vertical racks that are spaced and contain the neutron poison material Boral ™ to preclude criticality. In region 2, spent fuel assemblies are stored in vertical racks that contain the neutron poison material Boral ™ to preclude criticality. No credit is taken for the boron dissolved in pool water. Cooling and purification equipment is provided for the fuel pool.

The fuel handling systems provide for the safe handling of fuel assemblies and control element assemblies and for the required assembly, disassembly, and storage of the reactor vessel head and internals.

These systems include a refueling machine located inside containment above the refueling pool, the fuel transfer carriage, the tilting machines, the fuel transfer tube, a fuel handling machine in the spent fuel storage area, and various devices used for handling the reactor vessel head and internals.

1.2.8.5 Sampling Systems

Various sampling systems are provided to monitor both chemical and radiological conditions of the various process fluids used in the plant continuously. These sampling systems are:

a. Primary Plant Sampling System
b. Secondary Plant Sampling System
c. Auxiliary Sampling System
d. Water Plant Sampling System
e. Post Accident Sampling System

In addition to the sampling systems listed above, various grab sampling stations are provided throughout the plant for laboratory analyses to monitor the quality of the fluid used in the plant.

1.2.8.6 Circulating and Turbine Plant Cooling Water Systems

The circulating water system incorporates three pumps which circulate screened river water through the turbine condensers and the turbine plant cooling water system heat exchangers. The turbine plant cooling water system provides cooling for bearings, the lubricating oil system and other equipment in the secondary plant.

1.2.8.7 Plant Ventilation Systems

The plant ventilation systems maintain a suitable environment for equipment and personnel. The ventilation equipment incorporates steam coils to provide heating where necessary. Part of the containment air cooling and filtering system (an engineered safeguards system, see Section 1.2.5) is used to cool the containment atmosphere during normal plant operation. The control room and office areas are air-conditioned. Those systems serving areas subject to possible radioactive airborne contamination contain filtration equipment to restrict the release of activity to the environment.

1.2.8.8 Plant Fire Protection

The fire protection system provides water to fire hydrants, spray systems and hose racks in the various areas of the plant. Fire detectors and alarms are installed throughout the plant.

Where possible, noncombustible and fire resistant materials are used throughout the facility, particularly in areas containing critical portions of the plant such as the containment structure, control room, cable room, and rooms containing components of the engineered safeguards systems. Portable fire extinguishers are placed at key locations for use in extinguishing limited fires.

1.2.8.9 Auxiliary Steam System

The auxiliary steam system, which normally operates with turbine extraction steam, serves the plant ventilation heating coils and other equipment requiring a source of low pressure steam. A package boiler is used when the turbine plant is shutdown. The steam is pressure-reduced before use.

1.2.8.10 Compressed Air System

The compressed air system provides a supply of dry, oil-free air for instruments, controls and other equipment requiring service air.

1.2.9 Steam and Power Conversion Systems

The steam and power conversion systems consist of the main steam system, the turbine-generator, the condensate-feedwater system and the steam dump and bypass system.

The turbine-generator and heat cycle systems are designed for throttle flow of 6.61×10^6 lb/hr and a generator output of 506,569 KW, corresponding to reactor operation at 1500 MWt. The steam and power conversion systems were designed to accommodate the steam flow corresponding to a reactor core thermal output of 1500 MWt.

The turbine-generator is an 1800 rpm, tandem compound, non-reheat unit with one high pressure and two double-flow low pressure cylinders with 38-inch last stage buckets with external moisture separators between the high and low pressure cylinders. The generator is rated at 590,800 kVA at 0.85 power factor.

The condensate feedwater system is of the closed type with condenser deaeration. Each of the two parallel feedwater heating strings consist of five low pressure heaters and one high pressure heater. The feedwater pumps are electric motor driven. The auxiliary feedwater system provides the feedwater flow required during shutdown and emergency situations even with the loss of off-site power. One auxiliary feedwater pump is steam turbine driven, one is electric motor driven, and the non-safety class pump is diesel engine driven.

The steam dump and turbine bypass system permits the orderly disposal of the heat existent or subsequently produced in the reactor coolant system to the main condenser following shutdown.

A small amount of steam can be dumped directly to atmosphere by means of remote manual control valves.

1.2.10 Radioactive Waste Disposal System

The radioactive waste disposal system is used to collect, store, prepare for disposal, and dispose of radioactive wastes, including liquids, solids, and gases.

The system performs the following functions:

a. Collects all wastes from the containment and auxiliary buildings that are subject to radioactive contamination;

b. Collects the above wastes in three separate groups: liquid, solid, and gaseous;

c. Provides holding facilities to permit laboratory analysis of individual waste batches;

d. Provides treatment facilities for those waste batches that must be further prepared for disposal;

e. Provides for the temporary storage, monitoring, and controlled release of liquid and gaseous wastes;

f. Provides for the concentration, temporary storage, and packaging of contaminated wastes in suitable form for off-site disposal by waste handling firms approved by the Nuclear Regulatory Commission.

1.2.11 Plant Design and Operating Characteristics

The general mechanical contract for Fort Calhoun was let soon after the February 1968 Draft Edition of the Code for Nuclear Piping B31.7, was published; this draft was included in the contract requirements for nuclear piping. This draft differed from the adopted code only in details.

The Draft B31.7 1968 code provides for third party inspection by an authorized ASME organization where required by legal enforcement authorities. Since the local jurisdiction had no requirements for third party inspection, it was determined to proceed with OPPD QA performing the function of the examiner. All the required documentary evidence and traceability are provided. ASME form NP-1 "Partial Data Report" was not used because of the requirement of a signature by the ASME inspector, and the requirement that this form be utilized by an ASME stamp holder.

One deviation from B31.7 1968 was permitted in the marking of spool pieces used in piping fabrications. The code requires that each spool piece be marked with its heat number and that this mark be retained on the spool piece until it is erected, no matter how many pieces the original length of pipe is cut into. It has been found more practical to place an identification mark on each piping fabrication which includes a fabricator drawing number. The fabricator drawing identifies each spool piece and fitting in a given fabrication and states the heat number. This practice minimizes chances of making errors in transferring heat numbers and provides a more permanent record. The same type of procedure is specified in Section III of the ASME Code. The fabricator drawings were retained by Omaha Public Power District.

Instruments were not covered by Draft B31.7 1968. Connections between the body of instruments and connecting tubing were threaded because the instrument manufacturers do not provide for any other type of connections. Such connections were not used elsewhere in nuclear piping systems.

A few piping fittings, e.g., orifice flanges, were let out for contract prior to publication of the Draft B31.7 1968. Such fittings were designed, fabricated, and inspected in accordance with Draft B31.7 1968 as much as practical. In some cases, this required a post delivery inspection. Traceability of such fittings back to heat number is not always available.

The reactor coolant loop piping was also contracted for and designed prior to issue of Draft B31.7 1968. It is in accordance with USAS B31.1 1955, plus supplemental provisions, as described in Section 4.

The following table lists the principal plant design and operating characteristics. Data relate to 100 percent rated power (1500 MWt) unless specifically stated otherwise.

Table 1.2-1 - "Principal Plant Design and Operating Characteristics"

Plant	
Net Electrical Power Output, MWe	486.8 Gross
Electrical Power Output, MWe	501.1 Maximum Gross
Electrical Power Output, MWe at 1500 MWt	509.8
Reactor	
Core Thermal Output, MWt	1500
Maximum Flow Rate, lb/hr	87.2×10^6
Nominal Core Inlet Temperature, °F	545
Nominal Core Outlet Temperature, °F	594
Nominal Operating Pressure, psia	2100
Design Pressure, psia	2500
Design Temperature, °F	650
Number of Fuel Assemblies	133
Number of Control Element Assemblies	49
Number of Loops	2
Number of Pumps	4
Steam Generators	
Number of Units	2
Total Steam Flow, lb/hr	6.61×10^6
Feedwater Temperature, Design °F	439.7
Steam Temperature, Design °F	513.8
Steam Quality, expressed as moisture content, max.	0.2
Shell Side Design Pressure, psia	1000
Shell Side Design Temperature, °F	550
Turbine-Generator	
Turbine Type	Tandem-Compound, 1 HP. 2 LP, 1800 rpm, Non-reheat
First Stage Inlet Pressure, psia	735
Number of Feedwater Heating Stages	6
Generator Design Rating, kVA	590,800
Power Factor	0.85
Short-Circuit Ratio	0.58

1.3 PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA

The principal architectural and engineering criteria for the design of the Fort Calhoun facility are identified in the following Sections, 1.3.1 through 1.3.9.

1.3.1 Station Design

Principal structures and equipment which are necessary to the safe operation of the plant, or to prevent accidents or to mitigate their consequences, were designed, fabricated and erected in accordance with the then applicable codes as a minimum and to withstand the effects of the most severe earthquakes, flooding conditions, windstorms (including tornadoes), ice conditions, temperature and other deleterious natural phenomena reasonably to be anticipated at the site during the lifetime of this unit. The containment and the engineered safeguards system were sized for the maximum expected nuclear steam supply system output.

Redundancy is provided in reactor and safety systems so that no single failure of any active components of the system can prevent action necessary to avoid an unsafe condition. The plant is so designed as to facilitate inspection and testing of systems and components whose reliability is important to the protection of the public and plant personnel.

1.3.2 Reactor

a. The reactor is of the pressurized water type, designed to produce steam to drive a turbine generator. The reactor was initially operated at 1420 thermal megawatts and is currently licensed to operate at 1500 thermal megawatts.

b. The reactor is fueled with slightly enriched uranium dioxide contained in Zircaloy/ZIRLO tubes.

c. The minimum departure from nucleate boiling ratio and maximum fuel center line temperature evaluated at the design overpower condition are below values which could lead to fuel rod failures. The melting point of the UO_2 is not reached during normal operation including expected transients.

d. Fuel rod clad thickness was designed to maintain cladding integrity throughout the anticipated fuel life. Fission gas release within the rods and other factors affecting design life are considered for the maximum exposures.

e. The reactor and control systems were designed so that any xenon transients will be adequately damped.

f. The reactor was designed to accommodate safely, and without fuel damage, tripping of the turbine-generator, loss of power to the reactor coolant pumps, and other expected power transients.

g. Power excursions which could result from any credible reactivity addition accident will not cause damage, either by motion or rupture, to the pressure vessel or impair operation of the facility's engineered safeguards.

h. Neutron absorption for reactivity control is provided by control rods and by dissolved boric acid in the coolant. The boron chemical shim system is independent of the control rod system. Additional reactivity control at BOC conditions is provided by absorber material integrated within the fresh fuel.

i. For all operating conditions, the control rods are capable of providing an adequate shutdown margin at hot, zero power conditions following a trip, even with the most reactive rod stuck in the fully withdrawn position.

j. The boric chemical shim system is capable of adding boric acid to the reactor coolant at a rate sufficient to maintain an adequate shutdown margin during reactor coolant system cooldown at the maximum design rate following a reactor trip.

k. The combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient and the moderator pressure coefficient to an increase in reactor thermal power is a decrease in reactivity. In addition, the reactor power transient remains bounded and damped in response to any expected changes in any operating variable.

l. Automatic and redundant reactor trips are provided to prevent anticipated plant transients from producing fuel or clad damage.

1.3.3 Reactor Coolant and Auxiliary Systems

Heat removal systems are provided which can safely accommodate core heat output under all credible circumstances. Each of these heat removal systems has redundancy to provide reliable operation under all credible circumstances.

1.3.4 Containment

The containment structure, including the associated access openings and penetrations, was designed to contain the pressures and temperatures resulting from a maximum hypothetical accident in which it is assumed that:

a. The reactor coolant system water is released into the containment through a double-ended break of the largest reactor coolant pipe;

b. There is a simultaneous loss of outside electric power;

c. Heat is transferred from the reactor to the containment by water supplied from the emergency core cooling system;

d. The containment air recirculation and cooling system or the containment spray system functions;

e. The containment engineered safeguards do not operate until 60 seconds following the accident.

Means are provided for pressure and leak rate testing of the entire containment system including provisions for leak rate testing of individual piping and electrical penetrations that rely on gasketed seals, sealing compounds, or expansion bellows.

Integrity of the containment is protected against missiles from high-pressure equipment in the reactor coolant system and against credible missiles from external sources.

1.3.5 Engineered Safeguards

Engineered safeguards systems with redundant features to protect integrity of the containment are incorporated in the plant design to provide assurance that the release of fission products to the environment following any credible loss-of-coolant accident will not exceed the guidelines set forth in 10 CFR 100. Protection is afforded against missiles from high-pressure equipment in the reactor coolant system and against credible missiles from external sources.

The engineered safeguards system includes:

a. A containment spray system with redundant features, to remove heat from the containment in order to maintain containment integrity;

b. An emergency core cooling system to prevent fuel and cladding damage which would interfere with adequate emergency core cooling for all break sizes in the reactor coolant system piping up to the double-ended rupture of the largest reactor coolant pipe.

1.3.6 Nil Ductility Transition Temperature

Components of the reactor cooling and containment systems were designed so that, if operated in accordance with procedures, no deleterious pressure or thermal stress is imposed on the structural materials with due regard to the ductile characteristics of the materials at low temperatures.

1.3.7 Instrumentation and Control

Interlocks and automatic protective systems are provided along with administrative controls to ensure safe operation of the plant.

A reactor protective system is provided which initiates reactor trip if the selected plant operating parameters reach predetermined limits.

Redundant instrumentation is installed to allow periodic testing of the reactor protective system and so that failure or removal from service of any one protective system component would not preclude reactor trip or other safety action when required.

1.3.8 Electrical Systems

Normal and emergency sources of electrical power are provided to ensure safe and orderly shutdown of the plant and to maintain a safe shutdown condition under all credible circumstances.

1.3.9 Radioactive Wastes and Radiation Protection

The radioactive waste treatment system is designed so that discharge of radioactivity to the environment is in accordance with the requirements of 10 CFR Part 20.

The radiation shielding in the plant, in combination with plant radiation control procedures, ensures that operating personnel do not receive radiation exposures in excess of the applicable limits of 10 CFR Part 20 during normal operation and maintenance.

The plant is provided with a centralized control room and a technical support center having adequate shielding to permit occupancy during all credible accident situations.

1.4 RESEARCH AND DEVELOPMENT REQUIREMENTS

1.4.1 Introduction

The design of the Fort Calhoun nuclear plant was based upon concepts which have been successfully applied in the designs of pressurized water reactor power plants. However, certain programs of theoretical analysis or experimentation (constituting "research and development" as defined in the Atomic Energy Act, as amended, and in AEC regulations) were undertaken to aid in plant design and to verify the performance characteristics of plant components and systems. This section describes the results of these analytical and test programs, including experimental production and testing of models, devices, equipment and materials.

In carrying out these programs, information which is derived from research and development activities of the Atomic Energy Commission and other organizations in the nuclear industry were taken into account.

1.4.2 Moderator Temperature Coefficient

Analytical studies were completed as part of the detailed plant design to define the least negative moderator temperature coefficient for the Fort Calhoun reactor. The factors which affect the moderator temperature coefficient are discussed in Section 3.2 of this report.

The Fort Calhoun reactor was designed to have a hot, full power, beginning-of-life moderator temperature coefficient of reactivity which was zero or negative; however, a value of $+0.5 \times 10^{-4} \Delta\rho/°F$ was considered in the accident analyses of Section 14, where the consequences of an assumed accident would be adversely affected by a positive moderator coefficient.

Two analytical activities related to the moderator temperature coefficient have been completed:

a. As discussed in Section 3, a design analysis of the core to provide the configuration and concentration of poison shims which was used to reduce the beginning-of-life dissolved boron concentration.

b. An evaluation of the uncertainty band connected with the moderator coefficients.

Analyses of moderator temperature coefficients for the Connecticut Yankee reactor compared with measurements made during the course of the start up experiments are shown in Table 1.4-1. It can be observed from these data that the measured coefficient is at most $+0.16 \times 10^{-4} \Delta\rho/°F$ more positive than the calculated value. This good agreement lends confidence in the ability of the methods used to predict moderator coefficients to within the currently quoted uncertainty.

Table 1.4-1 - "Analysis of Moderator Temperature Coefficients in the Connecticut Yankee Reactor at Start-Of-Life"

Reactor Temperature	Dissolved Boron Concentration, ppm	Rod Worth Inserted, % $\Delta\rho$	Moderator Temperature Coefficient $10^{-4}\ \Delta\rho/°F$	
			Calculated	Measured
260	2040	-	0.46	0.57
560	2305	-	0.84	1.00
551	2045	1.8	0.37	0.47
561	1730	4.5	-0.23	-0.25
551	1610	5.6	-0.30	-0.30

1.4.3 Control Element Assembly (CEA) Testing

1.4.3.1 Critical Experiments

An experimental work program was completed to confirm techniques for calculating CEA worth and local nuclear peaking associated with the fuel assembly design. The work was performed in the CRX facility of the Westinghouse Reactor Evaluation Center at Waltz Mill, Pennsylvania, between June and August, 1967. The basic core configuration was a 30x30 square array of Zr-4 clad UO_2 fuel rods with an enrichment of about 2.7 w/o U^{235}; fuel rods were removed to create internal water holes or channels to accommodate absorber elements.

The experiments demonstrated that the Combustion Engineering Nuclear design package accurately predicts the CEA worth and local peaking in the small critical assemblies. This lent support to the use of these methods in the design of the Fort Calhoun core.

The significant conclusions which were drawn on the basis of the test program include the following:

a. The standard Combustion Engineering design methods were capable of calculating clean, room temperature, critical lattices (lattices which contain no CEA's, water slots, or other heterogeneties) to an accuracy of within 0.03 percent reactivity on the average.

b. The worths of various arrays of cylindrical absorbers containing boron carbide were predicted within 2 percent of the worth on the average, with errors for individual cases ranging from +6 percent to -2.2 percent of the worth. The arrays had worths ranging from 6 to 8 percent reactivity.

c. In assemblies containing water holes, the calculated and measured power peaking agree within 2 percent. Occasional differences between calculated and measured power of up to 4 percent were seen, but only in fuel rods of low power, usually near the reflector of these highly buckled small cores.

1.4.3.2 Control Element Assembly Testing

A series of tests have been completed on single CEA's and dual CEA's to satisfy the following objectives:

a. To determine the mechanical and functional feasibility of the CEA concept.

b. To experimentally determine the relationship between CEA drop time and CEA drop weight, annular clearance between CEA fingers and guide tubes, and coolant flow rate within the guide tube.

c. To experimentally determine the relationship between flow rate and pressure drop within the guide tube as a function of CEA axial position and of finger-to-guide-tube clearance.

d. To determine the effects on drop time of adding a flow restriction or of plugging the lower end of a guide tube (as might occur under accident conditions).

e. To determine the effects of misalignment within the CEA guide tube system on drop time.

Tests have demonstrated that the five finger CEA concept is mechanically and functionally feasible and that the CEA meets criteria established for drop time under the most adverse conditions. The testing also verified that the analytical model used for predicting the drop times gives uniformly conservative results.

The effects on drop time of frictional restraining forces, CEA guide tube plugging, and of guide-tube-to-finger axial clearance were investigated and defined in a cold, unpressurized test facility utilizing a CEDM simulator. Of particular interest, the tests demonstrated that misalignment and distortion in excess of those expected from tolerance buildup would still result in acceptable drop times.

A series of dual CEA tests were conducted in a cold water, unpressurized facility to determine by experimental means:

a. The performance characteristics and suitability of the gripper mechanism.

b. Effect of internals misalignments on dual CEA drop times.

c. Effect of variations in CEA system weight, guide tube ΔP, and CEDM axial restraining force on drop times.

Approximately 50 latchings and unlatchings with the gripper assembly were performed during this program. The force required to insert the grippers into the CEA sockets was approximately 28 pounds for the dual CEA; 14 pounds for each single CEA. Withdrawal forces were 46 and 23 pounds respectively. No malfunction was observed during this program.

CEA drop times for one-quarter, one-half, three-quarter and full height drops showed insignificant changes due to varying conditions of fuel bundle bowing and cocking and other internals misalignments. Within the bounds of the misalignments, bowing and cocking tested, no "worst" geometric misalignment configuration was evident.

A full height drop (122 inches) with a system weight of 415 pounds dry (348 pounds wet), a guide tube $\Delta\rho$ of 10 psi, and an axial restraining force of 85 pounds, was made in approximately 1.6 seconds.

From the drop test results, it can be stated that CEA scram time will be less than 2.5 seconds for the range of expected reactor conditions. Predicted drop times for a full height drop, in cold water, under varying conditions of $\Delta\rho$, weight and axial restraining force were determined from a multiple regression equation which was arrived at through analyses of all the drop time data.

A 1089 hour test run to verify the adequacy of the fuel assembly CEA system was completed. This test was conducted in the hot test facility at plant operating conditions of pressure, temperature, and flow. During this test the CEA accumulated 5000 feet of travel coupled to a prototype CEDM.

At the conclusion of this portion of hot testing the test section was disassembled and the components were inspected. Most indications of wear were found to be immeasurable, none were found deeper than 0.001 inch. Hot loop testing is discussed further in Section 1.4.4.

1.4.4 Control Element Drive Mechanism (CEDM) Testing

A prototype CEDM was subjected to accelerated life testing under simulated reactor conditions. The CEDM underwent more than 100,000 feet of travel and 200 full height trips. Additional tests were conducted on the first production mechanism, including torque evaluation both prior to the following performance tests, and hydrostatic testing. The test program demonstrated that the production mechanism will meet the established specifications and requirements.

Of particular importance, the capability of tripping, operating, coupling and uncoupling was demonstrated on the prototype under conditions of several misalignment (up to 0.25 inches); the ability of the mechanism to function without cooling for up to four hours in the "hold" or "drive" mode was proved.

A long-term program of loop testing of the CEDM-CEA at operating temperature and pressure was made in order to verify the results of those tests which were performed on the prototype. This phase of testing included performance tests and torque and force measurements.

Further discussion of development testing of the rack-and-pinion mechanism (CEDM) can be found in Section 3 of the Fort Calhoun USAR, in the Calvert Cliffs PSAR (AEC Docket No. 50-317, 50-318) and in the Hutchinson Island PSAR (AEC Docket No. 50-335).

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1.4.5 Fuel Assembly Design

1.4.5.1 Prototype Tests

A series of single phase dye injection mixing tests were conducted in 1968, using a portion of a CEA type assembly which was instrumented to permit measurement of the individual lateral flows across the boundaries of the twelve subchannels. The tests were used to determine the effect of subchannel geometry and location on the average level of turbulent mixing. The inverse Peclet Number calculated from the average of 56 individual turbulent mixing flows was 0.0035, which demonstrated the relative insensitivity of this parameter to moderate difference in fuel assembly geometry.

1.4.5.2 Spacer Grid Pressure Drop Tests

The spacer grid loss coefficient was originally a by-product of various component and coolant turbulent mixing tests on prototype fuel assemblies performed in 1966. The value was 0.3 velocity heads, where the velocity is that just upstream of the spacer grid.

In 1970, hot component tests on production CE fuel assemblies yielded a loss coefficient for the CE spacer grid which was weakly inversely dependent on Reynolds' Number. For a Reynolds' Number of $5x10^5$, corresponding to reactor nominal operating conditions, the value is 0.45. Confirmation of this value was achieved during additional hot component tests in 1971.

Tests conducted on a Combustion Engineering Maine Yankee fuel assembly (April 1976) and an Exxon Nuclear Maine Yankee reload assembly (December 1978) established grid spacer loss coefficients for both assemblies under the same conditions. Both ENC and CE Fort Calhoun assemblies are sufficiently similar to the respective Maine Yankee assemblies to justify adoption of the Maine Yankee hydraulic loss coefficients to the Fort Calhoun assemblies (Ref. 1-2).

The results are detailed in Reference 1-3.

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Evaluation of this increase in grid loss coefficient has shown a negligible effect on DNBR, primarily because it represents a small portion of the overall core pressure drop and because it applies to every subchannel of the fuel assembly.

1.4.5.3 Fuel Assembly Flow Distribution Tests

Velocity and static pressure measurements were made in an oversized model of a fuel assembly in order to determine the flow distribution present. The effects of the distributions on thermal behavior and margin were evaluated, where necessary, with the use of CORAL, a CE version of the COBRA subchannel thermal and hydraulic code. Subjects investigated include the following:

a. Assembly inlet flow maldistribution caused by the core support plate and header plate.

   The flow distribution was measured at several axial positions in the model, and the data were used with CORAL to evaluate the effect on thermal margin of the flow maldistribution at the assembly inlet. It was concluded that the effect was small enough so that no special allowance for it need be made in the context of CE's current conservative calculational techniques.

b. Assembly inlet flow maldistribution caused by blockage of a core support plate flow hole.

   Data were taken on flow recovery in the model with one core support plate flow hole completely blocked. CORAL evaluation of these data indicated that even the complete blockage of a core support plate flow hole during 120% of full-power operation would not produce a W-3 DNBR of less than 1.3.

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c. Flow maldistribution within the assembly caused by complete blockage of one to nine subchannels.

The flow distributions were measured at positions upstream and downstream of a blockage of one to nine channels. The influence of the blockage diminished very rapidly in the upstream direction. An analysis was made of the effect on DNBR downstream of a blockage of a single subchannel. The CORAL analysis of the data indicated that complete blockage of a single subchannel during 120% of full power operation would not produce a W-3 DNBR of less than 1.3 downstream of the blockage.

d. Flow distribution below the top header plate, as affected by the header plate and alignment plate flow hole geometry and by the presence of the CEA shroud.

Measurement of the upstream flow pattern, in what would be the active core region, showed no appreciable effect on an artificial exit flow distribution which was even more non-uniform than that characteristic of the geometry of the header plate, alignment plate, and CEA shroud.

1.4.5.4 DNB Testing in the Fuel Assembly

ENC/ANF Fuel Assemblies

The prediction of the onset of departure from nucleate boiling (DNB) in a PWR was performed by Exxon Nuclear Company (ENC) and Advanced Nuclear Fuels Corporation (ANF), and presently Framatome ANP Richland, Inc., using local flow conditions from subchannel analysis and the W-3 correlation plus correction factors for the effects of axial heat flux distribution, unheated boundary and mixing vane and spacers.

The analysis covered a variety of test conditions including variations and spacer design, grid spacer pitch, radial and axial power distributions and a range of hydraulic conditions. The DNB heat flux was predicted using the W-3 DNB correlation with applicable corrections in conjunction with the subchannel computer code XCOBRA-IIIC.

The W-3 predictions were compared to actual ENC DNB test conducted in the high pressure heat transfer facility at the Chemical Engineering Research Laboratories of Columbia University. Test conditions comprised a coolant inlet temperature range of 480°F to 630°F, a mass velocity range of 1.5×10^6 to 3.6×10^6 lbm/hr-ft^2 and a system pressure range of 1500 to 2400 psia.

Based on a comparison of the predicted to measured DNB heat flux, it was determined that a minimum DNB ratio (MDNBR) $\geq$ 1.199 assures with 95% confidence a rod having such a DNBR will have a 95% probability of avoiding DNB (Reference 1.12.1). Therefore, the 1.3 W-3 DNBR provides more than adequate DNB protection.

CE-1 Critical Heat Flux Correlation

The CE-1 critical heat flux correlation predicts DNB. It ensures, with a 95% confidence level, that at a MDNBR value of 1.15, 95% of the rods at this value will not experience DNB.

Based on experiments that were conducted at Columbia University the CE-1 CHF correlation is valid for a coolant inlet temperature range of 382 to 644°F, a pressure range of 1785 to 2415 psia and a mass velocity of 0.87×10^6 to 3.2×10^6 lbm/hr-ft^2 (Reference 1.12.2). This correlation has been used in conjunction with D-TORC and the CETOP thermal-hydraulics computer codes.

HTP DNB Correlation

The Framatome ANP Richland, Inc. HTP DNB correlation predicts DNB in their high thermal performance fuel. The 95/95 safety limit for the HTP correlation is 1.141. The HTP correlation is presented in the Reference 1-4. The range of coolant conditions represented in the HTP correlation data base is shown in Table 1.1 of Reference 1-4. The HTP correlation is applicable in this region. The ange of fuel design parameters repesented in the HTP correaltion data base is shown in Table1.2 of Reference 1-4. The HTP correlation is aplicable to fuels with design characteristics that fall within the data base range.

1.4.5.5 Fuel Assembly Testing

During the initial fuel assembly design, considerable testing was performed by Combustion Engineering (CE), now ABB/CE, to evaluate the effects of assembly and fuel rod vibration on fretting. Dynamic loop testing under simulated reactor operating conditions and mechanically induced autoclave vibration tests have been carried out.

Over 13,000 hours of test time have been accumulated on sub-size and full-size test assemblies in dynamic test loops. Test conditions have duplicated reactor temperature, water chemistry, pressure and flow velocity. Intentional cross flow and forced bundle vibrations have been used to accentuate any vibration between the fuel rods and the spacer grids. In addition, the spacer grid spring tabs have been individually set to simulate relaxed spring conditions.

In addition to the dynamic loop tests, forced vibration tests were performed. Fuel rods supported by spacer grids were vibrated at various frequencies and amplitudes. The tests were conducted in a static autoclave at operating pressure and temperature. Test variables included in addition to the vibration frequency and amplitude, spring preset of the spacer grids and time under test. The spring tabs were varied from design interference fits to gaps.

These tests did not reproduce reactor flow conditions but were designed to develop trends in the degree of fretting as a function of the test variables.

Even under unreasonably severe conditions (i.e., high frequencies, large amplitudes and gaps between the grid spring and fuel tube) no serious fretting has been observed in these tests.

An inspection of a fuel assembly after 1089 hours of exposure to plant operating conditions in the hot test facility revealed no unusual signs of wear. No indications deeper than 0.001 inch were found.

In Cycle 14 Fort Calhoun Station began using fuel fabricated by Westinghouse Electric Corporation, now Westinghouse Electric Company. Fuel assembly testing was conducted at their Columbia, South Carolina Facility in the "Fuel Assembly Compatability Test System" (FACTS) and later at the Science and Technology Center in Pittsburg, PA.

In Cycle 20, Fort Calhoun Station began using fuel fabricated by Framatome ANP Richland, Inc. Fuel assembly pressure drop tests were performed at their Richland, Washington facility in the Portable Hydraulic Test Facility (PHTF). The pressure drop testing characterized the component flow loss coefficients of the lower tie plate, bare rod friction, spacers, and the upper tie plate (Reference 1-5).

1.4.6 Reactor Vessel Flow Tests

Tests were conducted on a 1/4 scale model of the Fort Calhoun reactor to determine the hydraulic performance for normal and part-loop reactor configurations. The model was geometrically similar to the reactor except for the core region, where each fuel bundle was simulated by a cylindrical tube having the appropriate flow resistance. Air was used as the test medium.

The test program was completed and the evaluation of test data was completed. The following have been established:

a. The extrapolation of the pertinent model data to reactor conditions indicates a nozzle-to-nozzle pressure loss for four pump operation of 23.8 psi.

b. Core flow distribution for normal operation is generally symmetrical, with the lowest flows occurring in the peripheral core tubes. The minimum flow ratio, W/Wavg, is 94 percent and the maximum 103 percent. Physics calculations have shown that peripheral assembly power is significantly less than average, even though the hottest CEA may be part of the assembly.

c. Core tube flow ratios observed in the part-loop test were nearly as close to unity as those for the normal full flow simulation, with the exception of the situation involving two pumps operating in one loop. In that configuration, the minimum flow was determined to be 90 percent for those peripheral assemblies below the reverse flow outlet duct.

1.4.7 Fuel Rod Cladding

A substantial amount of information has been generated in the course of Combustion Engineering's continuing test program on Zircaloy-4 cladding.

a. Creep collapse tests on unsupported Zircaloy-4 tube specimens with t/OD ratios of between 0.050 and 0.071 have been performed at 650°F and 750°F. All tests were performed at an external pressure of 2400 psia. Results of tests performed at 750°F show that specimens with a t/OD of 0.58 (the reference design for Fort Calhoun) will collapse between 100 and 1000 hours at this temperature. Tests conducted at 650°F show collapse will occur between 5000 and approximately 30,000 hours.

   Zr-4 specimens supported with plenum springs and mandrels with machined defects to simulate chipping and separated pellets were also being creep collapse tested at 650°F and 750°F. Tests at 650°F after 14,000 hours show no indications of the cladding deforming into the spaces between the springs or mandrels. Specimens tested at 750°F have accumulated in excess of 10,000 hours and the cladding has shown some deformation into the intentional defects. Grooves 3/16 in. wide representing separated pellets caused a maximum of 2.0 mils deformation of the clad into the groove. Grooves 1/16 inch wide show no measurable deformation after 10,000 hours at 750°F.

b. Long term corrosion tests were performed on Zircaloy-4 fuel cladding under simulated reactor coolant conditions. Results of weight gain and hydrogen pickup were evaluated with respect to results obtained using demineralized water. The tests were conducted in coolant which has approximately 1100 ppm boron and less than 10 ppm NH_4OH added for pH control. Results after 2000 hours of test at 650°F show no effect of the coolant additives on corrosion rates or hydrogen pickup over similar tests conducted in demineralized water. Samples tested included as-received Zircaloy-4 tubing, and both air oxidized and 750°F steam autoclaved samples. Similar tests performed using LiOH as a pH control additive ran in excess of 8000 hours. Under the test conditions, the corrosion characteristics exhibited by the material were equivalent to those observed in demineralized water.

Dynamic corrosion tests were also done at 650°F to determine the effect of contamination on non-autoclaved material corrosion rates. Samples intentionally contaminated with dilute acid and machine oils were tested. Samples tested included as-received and 750°F steam autoclaved Zircaloy-4 tubing.

c. In addition to the corrosion studies mentioned above, CE participated in two "round robin" studies being conducted by ASTM.

The first study, "Corrosion Testing of ZR-4 and its Effects on Hydrogen Absorption," was performed by the ASTM-G Committee (Corrosion of Metals), Subcommittee No. VIII (Corrosion of Zirconium in Water Systems).

The second study, "Task Force on Hydride Orientation," was performed by the ASTM-B10 Committee (Reactive and Refractory Metals and Alloys), Subcommittee No. II (Zirconium and Hafnium).

Areas of concern to the committee include:

1. Methods of hydriding;
2. Effect of fuel tubing fabrication on platelet orientation;
3. Means of measuring hydride orientation; and
4. Effect of autoclaving on hydrogen pickup.

The studies included the analysis of the experimental effort and an interpretation of the effects.

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1.4.8 Engineered Safeguards Equipment Testing

1.4.8.1 Containment Air Recirculation, Cooling and Iodine Removal System Equipment

Tests on the fan motor combination, the cooling coils, and the equipment housing pressure relief ports were performed during 1970.

Flexible Connections for the Fans

Connections were designed to withstand a pressure increase from atmospheric to 60 psig in seven seconds.

Mist Eliminators

See Section 6, paragraph 6.4.3.2. These items were designed to survive the accident. Each cell contains a medium of a successful and proven type that has received the approval of the AEC for the given application.

High Efficiency Particulate Air (HEPA) Filters

See Section 6, paragraph 6.4.3.3. The filter media were subject to the quality assurance inspection test of AEC Health and Safety Information Issue No. 212, June 25, 1965, "Minimal Specification for the Fire-Resistant High Efficiency Filter Unit."

An air flow test at normal atmospheric conditions required a pressure drop less than 1.0 inch w.g. at 3000 CFM.

A dust capacity test in conformance with NBS Stanford Dust Loading Test using NBS Cottrell Precipitate with air flowing through the filter at the rated capacity, required the filter hold 4 lbs. of dust before the resistance reaches two inches w.g.

After successfully passing the preceding tests, the filter was subjected to an ultimate strength test with the deposited dust load still on the filter. A sufficient air flow was passed through the filter to maintain a pressure drop of six inches w.g. across the filter for ten minutes.

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At the end of the series, the filter was subjected to a DOP penetration test. The filter was tested according to MIL Standard 282. Penetration did not exceed 0.03% for 0.3 micron diameter homogeneous particles of dioctylphthalate (DOP).

Charcoal Filters

Each filter was subjected to an air flow capacity and resistance test in conformance with MIL-F-50048. Pressure drop was required to be less than 1.1 inch w.g. ± 10% at the rated flow.

Each filter was also subjected to a Freon leakage test in conformance with AEC-DP-1082. Each filter was required to remove at least 99.9% of the Freon.

Cooling Coil Tests

To experimentally determine the containment cooler's performance, a coil of reduced dimensions, but otherwise identical construction, was fabricated.

This coil and all required piping and instrumentation were placed inside a high temperature test facility and exposed to three separate conditions of pressure, temperature and humidity postulated to be present following an incident, one being the DBA conditions, the two others being reduced pressures and temperatures to predict performance as the containment atmosphere conditions are gradually returned to normal.

The test facility consisted of a large pressure shell capable of recirculating an air-steam mixture in the horizontal plane as required to duplicate the actual coil installation. Mixture flow through the coil was induced by a large internally mounted blower and the flow was measured by a calibrated orifice arrangement. The temperature, humidity, and pressure conditions were produced by suitable adjustments to various steam and air injection valves. The condensate produced was collected, utilizing a downstream sump arrangement, and the rate of removal was measured using a tapered tube flow meter (see Figure 1.4-1).

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The required temperatures were observed and recorded using a multi-point thermocouple recorder with other pertinent data being manually noted on the recorder chart. The coolant flow was measured several times during each test and was recorded in a similar manner.

The coil's performance was determined using the measured water flow rate and temperature differential data and is listed in Table 1.4-3. Verification of this rating was accomplished by the installation of sufficient instrumentation necessary to accurately compute the heat lost by the steam-air mixture. The percent variation of the two quantities is given in Table 1.4-4.

An analytical method used to predict the performance of cooling coils operating on condensing vapors has been made (Reference 1-1).

This method was used by various AAF personnel to produce a computer program which can predict any cooling coil's performance once the specific coil configuration and operating conditions are defined.

This method and program were utilized to predict the performance of a coil identical in configuration when operating under the same conditions used in testing the experimental coil. The analytical solution was to be considered accurate, and available for use in predicting the performance of a full-sized coil, if the prediction of the performance obtained by this method compared favorably with that determined experimentally. This comparison is shown in Table 1.4-3.

Table 1.4-3 - "Comparison of Experimental and Analytical Heat Removal Characteristics"

	Mixture		Condensate Rate		Heat Removal Rate			
Test No	Flow CFM	Press PSIA	Temp °F	Experimental Lbs/Hr	Analytical Lbs/Hr	Experimental BTU/Hr	Analytical BTU/Hr	% Deviation
1	2272	24.7	185	1278	1318	1.404×10^6	1.393×10^6	.8
2	2272	44.7	244	3510	3565	3.570×10^6	3.554×10^6	.8
3	2272	74.7	288	6180	5800	5.599×10^6	5.530×10^6	1.2

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The heat removal rate was experimentally determined to be 5.599 x 106 Btu/hr at the maximum postulated conditions of 288°F and 74.7 psia. This heat removal rate compares favorably with 5.530 x 106 Btu/hr rate determined analytically with a conservative deviation. The analytical method was therefore acceptable to predict the heat removal performance of similar coils of the same configuration.

Housings

The housings were designed to withstand a pressure differential of 2 psi without permanent deflection and to resist the containment pressure increase from atmospheric to 60 psig in seven seconds and the temperature increases from 120°F to 288°F at the commencement of the design basis accident. This pressure increase was assumed to be linear with time. Design took account of all modes and combinations of unit operation.

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Table 1.4-4 - "Cooling Coil Heat Balance Comparison Between Heat Gained by Water and Heat Lost by Mixture"

Test No.	Test Heat Pressure Deviation PSIA	Test Temp °F	Water Percent Flow Lbs/Min	Diff °F	Water Temp By Water Btu/Hr	Heat Gained Flow CFM	Mixture Outlet Temp °F	Temp °F	Mixture Outlet By Mixture	Condensate Lost
1	24.7	185	1200	19.5	1.404×10^6	2272	166	167	1.392×10^6	0.85
2	44.7	244	1190	30	3.570×10^6	2272	231	228	3.431×10^6	3.9
3	74.7	288	1152	81	5.599×10^6	2272	280	278	5.475×10^6	2.2

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All housing joints were tested on-site for leak tightness. The acceptance criterion shall be no evidence of light transmission through joints. Leak testing shall be carried out before painting touch-up.

Exhaust Plenum

Like the housings, the exhaust plenum is designed to withstand a pressure differential of 2 psi and to resist the containment pressure and temperature increase.

Housing and Plenum Pressure Relief Ports Tests

The ventilation housing ports are illustrated in Figure 6.4-4. These units are required to operate once during the containment pressure increase that is part of a loss-of-coolant accident to protect the housing against high pressure differentials. After the initial pressure increase, the differential pressure across the housing decreases. The valves close and do not operate again.

These valves were tested by their manufacturer, the American Air Filter Company, to establish design characteristics. Testing included runs wherein the pressure pulse expected at Fort Calhoun was essentially duplicated. Additional tests were run to establish orifice coefficient and opening time for the valves.

These valves are kept on their seats by gravity. Opening is done directly by the pressure difference they are to protect against. There are no mechanisms associated with these valves. There are also no close clearances or tight fits. The pivot is a loose fit. The seating surface is foam rubber. In view of these design features, it was considered that room temperature tests would be sufficient to demonstrate valve adequacy. These valves can be lifted manually to demonstrate that no deterioration occurs in service.

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Plenum Hatches with Fusible Links

The plenum hatches can be regarded as devices which go to their fail position during a loss-of-coolant accident. Instead of removing air pressure as is done in an air operated valve, the fusible link fails. Thereupon, springs take over to open the hatch. The fusible links are in accordance with the standard of Underwriter's Laboratories which requires qualification testing of fusible links as well as destructive tests of samples from each day's production. The hatches can be tested periodically by applying heat to the fusible link.

The hatches used to release air from the bottom of the fan discharge plenum consist of five independent sections, each actuated by its own fusible link. Each section consists of an opposed blade damper. The individual blades control flow by rotating, thereby being positioned either across or in line with the flow. There are no studs or similar devices which could impede the operation of these blades.

The gasket material at the edges of the blades is ethylene-propylene terpolymer. This material has been used by the damper manufacturer for the past eight years, and no degradation of the material has been observed. The manufacturer of the material also states that it will neither soften or embrittle in air atmosphere.

Melting of the fusible link releases a weight which actuates the damper blades. There is nothing which could impede the operation of the damper other than failure of the damper itself.

1.4.8.2 Safety Injection Equipment

Safety Injection Valve Operator Tests

This equipment has been qualified for harsh environment conditions as discussed in Section 1.6.16.

Safety Injection Flow Transmitter Tests

This equipment has been qualified for harsh environment conditions as discussed in Section 1.6.16.

1.6 SIGNIFICANT DESIGN CHANGES SINCE AWARD OF CONSTRUCTION PERMIT

1.6.1 Introduction

The principal changes which have evolved in the design of the Fort Calhoun plant since the construction permit was issued are identified in this section.

1.6.2 Safety Injection Actuation Signal

Diversification of the safety injection actuation signal (SIAS) has been provided. SIAS is initialed by either low pressurizer pressure or high containment pressure. (Formerly, SIAS required coincident low pressurizer pressure/low pressurizer level.) This signal is discussed in Section 7.3 of this report.

1.6.3 Reactor Vessel Surveillance Program

The program defined in Section IV of the PSAR has been modified. The program is described in detail in Section 4.5 of this report. There are six irradiation surveillance capsules, each containing Charpy impact and tensile specimens from base metal, weld metal, and heat affected zone test material; each capsule contains Charpy impact specimen correlation monitors from a standard heat of A533-B material provided by the USAEC Heavy Section Steel Technology (HSST) program; each capsule contains flux monitors and temperature monitors.

The six capsules are mounted on the vessel wall. The removal schedule ensures that at least one capsule will be irradiated to the peak lifetime fluence of the reactor vessel.

The irradiation surveillance program has been updated and will be conducted within the guidelines of ASTM E-185-82, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors."

1.6.4 Engineered Safeguards Pump Radiography

In supplement 3 to the PSAR, (page I.I-1, paragraph a) under High Pressure Injection, Low Pressure Injection and Containment Spray Pumps, it was stated that "Castings will be radiographed in accordance with ASTM E-71-64 or ASTM E-186-65T as applicable. Radiographs will be made in accordance with ASTME-94-62." This program has been altered as discussed below.

The casing configuration is not amenable to radiographic inspection, because of the irregularity of the pressure retaining castings.

All surfaces of pressure containing castings have been liquid penetrant inspected in accordance with the techniques specified in Appendix VIII of Section VIII of the ASME Boiler and Pressure Vessel Code. The standard of acceptance was that given by USAS B31.1, Case N-10. All fillet welds were liquid penetrant inspected in accordance with the techniques and acceptance standards specified in Appendix VIII of Section VIII of the ASME Boiler and Pressure Vessel Code.

All butt welds of pressure parts were fully radiographed in accordance with the techniques and acceptance standards specified in paragraph UW-51, Section VIII of the ASME Boiler and Pressure Vessel Code.

Pressure containing parts of the completed pump assembly were hydrostatically tested in accordance with the API Standard 610, paragraph 34, except that 1-1/2 times the design pressure was used instead of the maximum discharge pressure.

The pumps are discussed in detail in Sections 6.2 and 6.3 of this report.

1.6.5 Secondary CEA Position Indication

The CEA position indication systems were discussed in general terms in the PSAR and in Supplement No. 2; however, the type of display for the secondary system was not defined. The final design of these systems is described in detail in Section 7.5.3.3 of this report.

The secondary CEA position indication system provides a simultaneous visual display of individual position for all CEA's on a cathode ray tube position indicator. This system derives information from a series of reed switches actuated by a permanent magnet fixed to the top of each CEA extension.

1.6.6 Letdown Flow Control in Chemical and Volume Control System (CVCS)

The CVCS has been modified. The system now includes three constant 40 gpm, positive displacement charging pumps; there is no longer a variable speed charging pump. The variation in letdown flow rate required to maintain coolant system volume is accomplished by either of two letdown control valves. These regulating valves respond to the pressurizer level controller signal.

The system is described in Section 9.2 of this report. The third constant speed charging pump and the letdown control valves supplant the variable speed pump and the letdown orifice described in the PSAR.

1.6.7 On-Site Diesel Oil Storage Capacity

The diesel oil storage capacity presented on Page VIII-2-6 of the PSAR was incorrect due to a typographical error. The on-site diesel fuel storage capacity is sufficient to supply post accident power requirements for seven days as required by IEEE Criteria No. 308 for Class IE Electrical Systems for Nuclear Power Generating Stations. Additional fuel oil is readily available at short notice. This information is presented in Section 8.4.1 of this report.

1.6.8 Number of Raw Water Pumps

In Section IX-8 of the PSAR it was stated that three raw water pumps are provided. The number of pumps has been increased to four to allow for continued plant operation with one pump down for maintenance. The raw water system provides sufficient cooling capacity for any credible failure mode situation following a loss-of-coolant accident. The pumps are discussed in Section 9.8 of this report.

1.6.9 Charcoal Filter Spray Dousing System

The containment charcoal filters have now been provided with a spray dousing system to maintain the temperature of the charcoal below its ignition temperature. The containment spray water system is the source of the dousing water. The equipment provided is described in Section 6.4.3.5 of this report.

1.6.10 Containment Emergency Access Lock

In the supplement to the Summary of Application it was indicated that an emergency containment air lock is provided. In fact, this referred to an equipment hatch that is not usable as a personnel access lock. In common with other nuclear plants, the Fort Calhoun containment has one personnel access lock for both normal and emergency use. The air locks are discussed in detail in Section 5.9.4 of this report.

1.6.11 Containment Air Cooling System Ductwork

In Supplement 7 to the PSAR it was stated that the containment air recirculation ductwork would be designed so that it could not be collapsed during an accident. The present design takes no credit for the greater part of this ductwork after a major loss-of-coolant accident. The distribution plenum, however, is designed to resist collapse and the discharge from the cooling units is made through hatches at the lower end of this plenum in the event of a loss-of-coolant accident. This equipment is described in Section 6.4.3.8 of this report.

1.6.12 Fire Protection Water Supply

In Section IX-11 of the PSAR it was indicated that the source of fire protection water would be on-site wells. The fire protection system has since been extensively reviewed with the Nuclear Energy Property Insurance Association (NEPIA); with their approval the source has been changed from well water to the Missouri River. Two pumps are provided; one is electric motor driven and the other is diesel driven. This system is described in Section 9.11 of this report.

1.6.13 Reactor Coolant Ion Exchange Equipment

In Section IX-2 of the PSAR it was stated that the chemical and volume control system included two mixed bed purification demineralizers and one deborating demineralizer. The system has since been expanded to include two mixed bed demineralizers, two deborating demineralizers and one cation unit. The chemical and volume control system is discussed in Section 9.2 of this report.

1.6.14 Analytical Model of Containment Liner

The analytical model of the containment liner described in supplement 8 to the PSAR has been modified to recognize the effect of the four adjoining panels. The original model considered a single panel with fixed ends. The analysis has, as before, been carried out by the Franklin Institute Research Laboratory and is discussed in Section 5.6 of this report.

1.6.15 Ventilation Discharge Duct

In Supplement 2 to the PSAR it was stated that, as a Class I structure, the ventilation discharge duct would be capable of withstanding design tornado forces.

However, since no credit is taken for the height of this duct in the evaluation of off-site exposures, both during normal operation and accident situations, it is not now designed to withstand tornadoes. The discharge duct is described in Section 9.10.3.1 of this report.

1.6.16 Electrical Equipment Environmental Qualification - Harsh Environment

In order that the consequences of an accident can be mitigated by the engineered safeguards system, specific safeguards equipment must operate in an environment which is outside the bounds of a normal operating environment. A harsh environment subjects the safeguards equipment to stresses which could, if not accounted for in the equipment's design, lead to equipment failure. To insure the safeguards equipment will perform its intended function in the harsh environment, all electrical equipment required for safe shutdown has undergone a series of qualification tests and/or analyses to meet the requirements of 10 CFR 50.49.

An environmental stress may be induced during or following a Design Basis Accident (DBA) by any one or combination of the following parameters:

Pressure
Temperature
Relative Humidity
Chemical Spray
Radiation
Submergence
Aging

These stress parameters are plant-specific and were determined from analyses described in Section 14 (Safety Analysis) and Appendix M (High Energy Line Break) of the FSAR and from the definition of environmental parameters completed as a result of NUREG's 0578, 0696, and 0737.

Aging or qualified life must be accounted for although it is not a direct result of a DBA/Post DBA stress but may cause a deterioration with time which could result in the failure of the equipment in a DBA/Post DBA situation.

For equipment that demonstrates less than a forty year qualified life (aging qualification), either by test or analysis, the Qualified Life Program has been instituted to refurbish, test or replace the equipment on a periodic basis. This program will be incorporated into the existing preventive maintenance program to ensure the equipment is returned to its initial qualification standards prior to the end of each qualified life period.

1.6.17 Inadequate Core Cooling Instrumentation

In response to NUREG-0737, instrumentation has been installed to detect the approach to, existence of, and recovery from an inadequate core cooling situation. This system includes upgraded core exit thermocouples (CET's), heated junction thermocouple (HJTC) probes for reactor vessel level indication, associated cabling, and the Qualified Safety Parameter Display System (QSPDS) microprocessors and displays. This system provides various information regarding the status of the system. This system is described further in Section 7.5.6.

1.6.18 Diverse Scram System

In response to 10-CFR-50.62, Requirements for Reduction of Risk from Anticipated Transient Without SCRAM System (ATWS Rule), the Diverse Scram System has been designed and installed to reduce the risk of an Anticipated Transient without SCRAM (ATWS) by providing an independent trip function. The Diverse Scram System independently monitors pressurizer pressure and, based on a 2-out-of-4, logic initiates a Reactor trip at 2450 psia. The Reactor Protective System Trip Setpoint of 2350 psia ensures that the Section 14 analysis setpoint of 2400 psia is met. The Diverse Scram System is further described in Section 7.2.11.

1.6.19 Steam Generator Orifice Plate Removal

The Reactor Coolant System has been modified. Orifice plates installed on the hot leg tubesheet in each steam generator were removed. Removal of the orifice plates reduces the hydraulic resistance for reactor coolant through the steam generators. The decreased system resistance is predicted to increase the reactor coolant flow rate by approximately 5% from 208,000 gpm to 218,000 gpm. The resulting increase in reactor coolant flow is predicted to decrease T hot by 2.6°F

1.11 SPECIFIC REFERENCES

1-1 Rivers, R. D., York, J. D., and Pasha, M. Prediction of Heat and Mass Transfer in Staggered-Tube Chilled Water Coils Operating on Steam-Air Mixtures, RR#271, American Air Filter Company, Inc., Louisville, Kentucky, January 27, 1967.

1-2 Fort Calhoun Reload Fuel Design Report: Mechanical, Thermal-Hydraulics and Neutronic Analysis. XN-NF-79-69, September, 1979.

1-3 Single Phase Hydraulic Performance of Exxon Nuclear and Combustion Engineering Maine Yankee Fuel Assemblies, XN-NF-78-21, February, 1979.

1-4 EMF-92-153 (P) (A) and Supplement 1, "HTP: Departure from Nuclente Boiling Correlation for High Thermal Performance Fuel," March 1994.

1-5 EMF-2455 (P), Revision 0, "Fort Calhoun Cycle 20 Thermal Hydraulic Compatability Analysis," Siemens Power Corporation, August 2000.