Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247

May 12, 2000
LIC-00-0050

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

Reference: Docket No. 50-285

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

As required by 10 CFR 50.59(b)(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 revisions to the Fort Calhoun Station Updated Safety Analysis Report (USAR) other than those resulting from 10 CFR 50.59 (i.e., revisions resulting from 10 CFR 50.54, 10 CFR 50.90 and administrative changes). In addition, commitment revisions requiring Commission notification are provided in Attachment B. The attached information accurately represents changes made since the previous submittal necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirements. This includes identification of changes made under the provisions of 10 CFR 50.59 but not previously submitted to the Commission. The attached information covers the period of April 1, 1999 through December 31, 1999.

In this submittal, Omaha Public Power District is reissuing the sections of the USAR that were revised. 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), enclosed is one original set and 10 copies of the affected sections of the USAR. The original set is designated as Copy Number 1 and the 10 copies as Copy Numbers 2 through 11. The Senior Resident Inspector's copy (Number 14) has been updated through the plant's distribution process.

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Employment with Equal Opportunity

Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247

Please contact me if you have any questions.

Sincerely,

W. G. Gates
Vice President

Attachments
Enclosure

WGG/dls

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

Employment with Equal Opportunity

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

AFFIDAVIT

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

W. G. Gates

W. G. Gates
Vice President

STATE OF NEBRASKA)
) ss
COUNTY OF DOUGLAS)

Subscribed and sworn to before me, a Notary Public in and for the State of Nebraska on this 12th day of May 2000.

Kristie E. Thompson
Notary Public

GENERAL NOTARY-State of Nebraska
KRISTIE E. THOMPSON
My Comm. Exp. May 1, 2001

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-038	**Description:** This safety evaluation was written to revise two condensate system piping and instrumentation drawings (P&IDs) to close out the configuration change made under EAR 97-276. This minor configuration change added quick connections to the flanged connections downstream of condenser drain valves FW-676 and FW-678 to aid in draining the condenser hotwell for maintenance. The added connections will allow an easier and safer (industrially) means of attaching drain hoses to the valves to facilitate maintenance activities.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? Failure of the main condenser or the drain valves is not a postulated accident or single failure evaluated in any USAR Chapter 14 Safety Analysis. The addition of the connections to the drain valves for maintenance has no effect on the valves' (or the condenser's) pressure/leakage integrity.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The non-safety-related main condenser (with the drain valves passively closed) is the desired heat sink during an event where decay heat removal is needed. This change has no effect on the capacity or the integrity of the condenser during that event.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? Failure of the main condenser is no more likely with the drain connections added to the condenser drain valves.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? No Technical Specification safety limits or setpoints involve the condenser.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-060 USAR 99-050	**Description:** The last paragraph of USAR Section 5.9.6.1 is being deleted. USAR Section 5.9 describes requirements for containment penetrations, and USAR 5.9.6.1 specifically addresses pipe penetration and isolation valve testing. The containment penetration testing of penetrations M-93 (FW-161) and M-96 (FW-162) was determined to be exempt from Appendix J requirements. Therefore, USAR 5.9.6.1 describes testing that is not required or performed.	Section 5.9
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The proposed "activity" is deletion of outdated information from the USAR. This activity is administrative in nature and has no impact on the probability of occurrence of an accident previously evaluated in the USAR.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The proposed "activity" is deletion of outdated information from the USAR. This activity is administrative in nature and has no impact on the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? The proposed "activity" is deletion of outdated information from the USAR. This activity is administrative in nature and does not create the possibility of an accident of a different type than any previously evaluated in the USAR.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? The proposed "activity" is deletion of outdated information from the USAR. This activity is administrative in nature and has no impact on the margin of safety as defined in the basis for any Technical Specification.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-047 DCR 10981 DCR 10982	**Description:** Guidance was incorporated in the EOPs/AOPs to re-establish auxiliary building ventilation with a minimum of one supply and two exhaust fans following a design basis accident (DBA). This guidance includes direction for local manual operation of the associated dampers if instrument air is not available, jumper of a fan start interlock if instrument air is not available and guidance to realign the electrical distribution system to power the fans to obtain the minimum required fan alignment.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? Restoration of auxiliary building ventilation post DBA was evaluated. The change ensures that safe shutdown equipment will perform as analyzed. The control room habitability analysis assumes the auxiliary building ventilation is the release point. Therefore, there is no change to consequences of an accident previously evaluated in the USAR.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? Restoration of auxiliary building ventilation post DBA was evaluated. The change ensures that safe shutdown equipment will perform as analyzed. Diesel Generator loading and bus loading are maintained within analyzed limits. The probability of equipment malfunction is not increased as a result of this change. The consequences of equipment malfunction remain as analyzed.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? Restoration of auxiliary building ventilation post DBA was evaluated. The change ensures proper operation of safe shutdown equipment in response to an analyzed accident. No new accident initiating mechanisms are introduced as a result of the change. No new accidents are created. No new equipment malfunctions are created.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-047 DCR 10981 DCR 10982	**Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?** The margin of safety remains as analyzed. The change ensures that auxiliary building ventilation is restored and safety injection/containment spray pump motor performance is as analyzed.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-09 USAR 99-021	**Description:** USAR Section 8.5.1 e - The description of the cable spreading room is being changed to describe the presence of systems and equipment in the room that are required for the operation of the cable spreading room and control room complex. These are in addition to the equipment required for cable routing (conduits, junction boxes and cable trays). USAR Section 8.5.4 d - This description of the junction boxes in the cable spreading room was eliminated. USAR Section 8.5.1 e - The cable spreading room, in addition to the USAR content description (contains only electrical equipment necessary to route electric cables/wires), also contains fire extinguishers, fire detectors and panels, fire suppression (Halon), for both room 70 and control room walk-in cabinet, copper instrument air line - abandoned with the valve in room 69 shut, lighting, emergency lighting & Gai-tronics, control room sanitary drain pipe, control room air conditioner room floor drains, drain line which includes valve VD-307 (Mechanical Equip Room 72 Floor Drain Isolation Valve), ventilation dampers, 800 mHz radio system junction box JB-622A, electrical power receptacles, condenser vacuum gauge lines, abandoned wires and cables, control room delta-pressure sensor in the south-west corner ceiling, and lighting panel LP-7. The change ensures that equipment in the room is described to define the different functions in the room. USAR Section 8.5.4 d - The description of the junction boxes in the cable spreading room was eliminated because it is not applicable to the subject of USAR Section 8.5.4 Cables. The necessary description is contained in USAR Section 8.5.1 e. This section (8.5.4 d) as it now is written conflicts with Section 8.5.1 e in that this section allows only two junction boxes while Section 8.5.1 e has no restriction on the number of junction boxes.	Section 8.5

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-09	**Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR?** Failure of the equipment (not presently described in the USAR and located in the cable spreading room) will not increase the probability of an analyzed accident. The components under consideration are not accident initiators. No adverse interactions with equipment (not described) that could cause an analyzed accident have been identified. No adverse interactions (from the equipment not presently discussed in the USAR and located in the cable spreading room) have been identified that could effect the fuel, the reactor coolant system or the containment boundary. The equipment does not have a direct effect on the fuel, the reactor coolant system or the containment boundary. The consequences of the accidents previously evaluated are unchanged by this equipment.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? No new safety system interfaces were identified. No interactions with critical quality equipment (CQE) equipment from the equipment (not presently discussed in the USAR and located in the cable spreading room) that could cause equipment malfunction have been identified. The probability of equipment malfunction is not changed from that assumed in the USAR. Failure of the equipment (not presently discussed in the USAR and located in the cable spreading room) does not affect any safety related equipment. No adverse interactions between the equipment and safe shutdown components have been identified. The consequences of equipment malfunction are as analyzed in the USAR. The equipment failure modes assumed in the USAR remain as analyzed.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-09	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** No adverse direct or indirect effects from the equipment (not presently discussed in the USAR and located in the cable spreading room) have been identified. The equipment not described in the USAR does not change the configuration of any systems. The radiological barriers and mitigation systems configuration is as analyzed with no new accidents created. No new failure modes or the possibility of a different type of accident have been created. No new equipment failure modes have been introduced. Since no new interactions are present, the possibility of a different type of equipment malfunction was not created. Equipment failure modes and malfunctions remain as analyzed.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? No adverse direct or indirect effects from the equipment (not presently discussed in the USAR and located in the cable spreading room) have been identified that could effect safety related equipment. The equipment is expected to function as designed. Since no new interactions are present, and safety related equipment is expected to function as designed, the margin of safety is not changed. The equipment is expected to function as designed.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-043 DCN 10856	**Description:** Core Operating Limit Report (COLR), Section 8.0, "Peaking Factor Limits", Table 2 "Maximum Full Power F_RT and $F_{XY}T$ Limits." F_RT Limit was changed to 1.650, and $F_{XY}T$ was revised to 1.751. Core Operating Limit Report Figure "F_RT, $F_{XY}T$, AND CORE POWER LIMITATIONS," was revised to reflect the following values: Power / Current ($F_RT/F_{XY}T$) / Revised ($F_RT/F_{XY}T$) 100% / 1.768 / 1.876 / 1.65 / 1.751 90% / 1.768 / NA / 1.65 / NA 78% / 1.80 / NA / 1.68 / NA 65% / 1.929 / 1.93 / 1.80 / 1.80 55% / NA / 2.06 / NA / 1.92 A 10 CFR Part 21 error notice from ABB/CE on the Thermal Hydraulics (T/H) methodology required a re-evaluation of the Cycle 18 T/H and Setpoints analysis, (Reference condition report 199801739) which was performed in EA-FC-98-057. From this analysis, a determination was made that the Cycle specific maximum F_RT and $F_{XY}T$ values were required to be limited to values less than or equal to 1.650 and 1.751 respectively to ensure that the trip limits currently installed in the RPS remain conservative until the end of Cycle 18. Changes were required to be made to the Better Axial Shape Selection System (BASSS) input cards in addition to the COLR table and figure to ensure that the current trip limits remain conservative. Engineering Analysis EA-FC-98-057 utilized NRC approved methodologies and was independently reviewed by thermal hydraulics and analog setpoints technical experts at ABB/CE. **Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR?** The COLR provides cycle-specific limits on operation based upon results from the Cycle 18 reload analyses and setpoints analysis in accordance with NRC approved reload methodology. Thus, the revision to the Cycle 18 COLR does not involve changing any previously evaluated accidents in the USAR. Therefore, no increase in occurrence probability is possible and no increase in consequences is possible.	None

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-043 DCN 10856	**Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR?** The revisions to the Cycle 18 COLR do not involve altering equipment that is important to safety previously evaluated in the USAR. Therefore, no increase in malfunction or probability of occurence is possible. NRC approved methodologies are utilized when preparing revisions to the setpoints, which includes BASSS. These changes will ensure that the currently installed setpoints remain conservative for the Cycle 18 operation to the end of cycle.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? The revision to the Cycle 18 COLR does not involve an accident condition not previously evaluated in the USAR. Therefore, there is no possibility of a non-evaluated accident or the creation of a different type of accident. In addition, no possibility of an equipment malfunction exists.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? In this revision of the Cycle 18 COLR, the margin of safety is maintained and guaranteed by the use of NRC approved reload and setpoints methodologies, which were utilized in Engineering Analysis EA-FC-98-057 and Cycle 18 reload analyses.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-36 DCR 10614	**Description:** Plant operating instruction OI-VA-2 was changed. 1. An "Other" reference section is being added to reference condition report 199801916 and LER 285-98014. 2. The index is being updated to reflect a new Table 2. 3. Changed Attachment 1 Note to describe VA-35A and B as supply fans. 4. Corrected Attachment 1A equipment description. The change to OI-VA-2 provides direction for actions on loss of heating, ventilation and air conditioning (HVAC) to ensure the operability of the electrical system, and requires a reactor shutdown if ambient temperatures are expected to exceed established limits.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The change to OI-VA-2 provides direction for actions on loss of HVAC. The uncontrolled (i.e., non-radiological controlled area [RCA]) portion of the auxiliary building HVAC system has no effect on the probability of an analyzed accident. The auxiliary steam to the diesel generator (DG) rooms must be isolated to prevent an auxiliary steam pipe break from having adverse equipment effects. The effect of a high energy line break (HELB) on the switchgear rooms through ventilation fan VA-41 and the associated dampers flow path is mitigated by the fire dampers and trap door as described in the USAR Appendix M. The consequences of the HELB remain as analyzed. Tornado venting remains as analyzed; the doors blocked open are assumed to fail open.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-36 DCR 10614	**Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR?** The OI-VA-2 change provides direction for actions on loss of HVAC to ensure that a malfunction will be within analysis assumptions or require that the reactor be placed in a safe shutdown condition. The auxiliary steam to the DG rooms is isolated to eliminate the possibly of an unanalyzed line break. The HELB in Room 81 effect on the switchgear rooms through the original HVAC and associated ductwork and dampers is mitigated by the fire dampers and trap door as described in the USAR Appendix M. This ensures that the probability of equipment malfunction remains unchanged. Tornado venting remains as analyzed, the doors being blocked open are assumed to fail open. The change ensures that any malfunction will be within analysis assumptions or require that the reactor be placed in a safe shutdown condition. This ensures that the probability or the consequences of equipment malfunction remains unchanged.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? The change to OI-VA-2 provides direction for actions on loss of HVAC. The change ensures the electrical system will not malfunction to create the possibility of a different accident or require that the reactor be placed in a safe shutdown condition. This ensures that the existing accident analysis bounds expected events/DBA. The HELB in Room 81 effect on the switchgear rooms through the original HVAC and associated ductwork and dampers is mitigated by the fire dampers and trap door as described in the USAR Appendix M. Isolation of the auxiliary steam line to the DG rooms prevents any adverse interactions by preventing an auxiliary steam line break. No new accidents have been created. Tornado venting remains as analyzed; the doors being blocked open are assumed to fail open. This change ensures that any malfunction will be within analysis assumptions or require that the reactor be placed in a safe shutdown condition. The possibility of a malfunction of a different type is not created.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? The change to OI-VA-2 provides direction for actions on loss of HVAC to maintain the analyzed margin of safety. If limits are expected to be exceeded a reactor shutdown is required. The change ensures that the margin of safety with respect to the electrical distribution system remains unchanged.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-072 DCR 11539	**Description:** The leak rate across PCV-6681A and PC-6681B (control room normal fresh air makeup isolation dampers) was measured using tracer gas. Each damper's leak rate was measured individually with the ventilation system in the filtered makeup mode. Since the filtered mode closes both dampers, the damper not being measured was opened by means of a temporary electrical jumper. The instrument air supply valve to the damper being measured was closed. With the system in filtered makeup mode, tracer gas was injected into the control room filtered air intake duct at a controlled rate. Tracer gas concentration was measured in the control room and at various duct points, and the concentration values used to calculate the fresh air makeup rate and the total outside air intake rate were also measured. During this measurement, the turbine building and auxiliary building uncontrolled area ventilation systems was operating in modes which pressurize these areas adjacent to the control room envelope. The tracer gas used was sulfur hexafluoride (SF_6). The damper leak rate measurement quantified PCV-6681A/B leakage while in the filtered makeup mode. Closing the instrument air (IA) supply valve to the damper being measured provides added assurance that the damper will remain closed since the redundant damper is open. The rest of the procedure quantified the amount of outside air drawn into the control room envelope while in the filtered makeup mode. The amount of "unfiltered" air being drawn into the envelope was determined by subtracting the measured makeup air flow rate (from the in-service filter train) from the total outside air intake rate as determined by measuring gas concentration inside the control room. The turbine building and auxiliary building uncontrolled area ventilation operating modes during the test created conditions conducive to air infiltration into the control room envelope from these areas, resulting in a bounding/conservative measurement of outside air intake rate.	None

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-072 DCR 11539	**Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR?** The tracer gas measurement methods and techniques do not involve the initiating mechanisms for previously analyzed accidents, therefore these measurements on the control room ventilation system did not make a previously analyzed accident more likely. The tracer gas measurements were performed with the control room ventilation system in the filtered makeup mode, which is the desired mode of operation after a radiological accident. The system remained capable of performing its function of maintaining control room habitability should an accident occur while tracer gas measurements were in progress. The normal fresh air intake duct was isolated by a closed damper should a toxic gas accident have occurred during damper leak rate measurement.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The tracer gas measurement methods and techniques did not affect existing equipment failure mechanisms, so these measurements did not make a previously analyzed equipment failure more likely. Redundant filter trains remained available during the tracer gas measurements, ensuring one operating train in the event that an active failure disabled the other train. During damper leak rate measurement, one of the two redundant normal fresh air intake dampers was open, but the other was in its desired fail-closed position, and was not susceptible to opening due to an active failure. The ventilation system remained capable of performing its function in the event of an equipment failure.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-072 DCR 11539	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** The sulfur hexafluoride tracer gas did not represent a hazard to control room personnel or equipment. During the damper leak rate measurement, a gas cylinder was brought into Room 81, and the cylinder was secured to prevent falling over. Per the guidance in procedure PED-GEI-13, a system is not a credible missile source unless it operates at reactor coolant system pressure (2100 psi) or higher. This is not the case with the SF_6 gas cylinder. The nature of the measurements is such that they did not introduce a new accident-initiating mechanism, so this activity did not create the possibility of a different type of accident. The tracer gas measurements did not introduce a new equipment failure mechanism. The tracer gas was inert and did not adversely affect the control room charcoal filters or other equipment.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? These measurements did not adversely affect the ability of the control room ventilation system to perform its intended function in maintaining a habitable control room environment. Operability requirements of Tech Spec 2.12 was observed. The margin of safety as defined in the Basis for Tech Spec 2.12 was not reduced by this activity. The system could still shut down in the event of a toxic gas accident, so the basis of Tech Spec 2.22 was also unaffected by this activity.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-046 DCN 10116	**Description:** A compressed gas source (N_2 cylinders) was added to act as a backup for instrument air to LCV-1190 and LC-1190A/B. The document updates related to the design include a revision of instrument air P&ID 11405-M-264 sheet 3. The purpose of the change is to enable LCV-1190 to remain closed for a sufficient time to allow an operator to locally isolate the line using the manual valve(s) on either side of LCV-1190 in the event of a loss of instrument air. LCV-1190 fails open. This will prevent draining of the condensate storage tank to the condenser hotwell and ensure a suction source exists for engine-driven auxiliary feedwater pump FW-54 for post-trip heat removal. The instrument air P&ID was revised to reflect changes made to Riser "N".	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? Revising P&ID 11405-M-264 sheet 3 to reflect changes made to Instrument Air Riser "N" does not make an initiating mechanism more likely for a USAR-analyzed accident. In addition, this change does not impair the ability of safety-related equipment to respond to a USAR-analyzed accident, so accident consequences were not increased. Affected portions of the instrument air system are not safety-related.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? Revising P&ID 11405-M-264 sheet 3 to reflect changes made to Instrument Air Riser "N" does not make a safety-related equipment failure mechanism more likely to occur. In addition, this change does not impair the ability of any safety-related equipment to respond to a USAR-analyzed failure of safety-related equipment, so consequences are not increased. The scope of the P&ID change does not affect safety-related components.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-046 DCN 10116	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** Revising P&ID 11405-M-264 sheet 3 to reflect changes made to Instrument Air Riser "N" does not introduce an initiating mechanism for a different type of accident or a different type of safety-related equipment failure. The scope of this change does not affect safety-related equipment.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? Revising P&ID 11405-M-264 sheet 3 to reflect changes made to Instrument Air Riser "N" does not affect the margin of safety as defined in the basis for any Technical Specification because the scope of the P&ID change does not affect equipment covered by Tech Specs.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-05, MR-FC-97-018 USAR 99-08	**Description:** The stem travel of FCV-326 was reduced by installing an extension onto the existing diaphragm stop in the upper actuator housing. Position indication limit switch mounting was modified to accommodate the new stroke length. This was done to decrease the full-open C_V of FCV-326 and add hydraulic resistance to the low pressure safety injection (LPSI) system. This will reduce the flow rate of a single operating LPSI pump during a large break loss of coolant accident (LOCA) so that positive net positive suction head (NPSH) margin exists for that pump for the entire injection phase. Calculations with the existing configuration indicate a potential marginal NPSH deficiency for pump SI-1B at the end of the injection phase if it is operating alone with all four LPSI loop injection valves open. This modification will allow deletion of a step in various EOPs/AOPs which directs operators to throttle LPSI flow (if necessary) if only one LPSI pump is operating. The throttling step has been identified as an Operator Work-Around (OWA).	Section 6.2
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The stem travel length or full-open C_V of FCV-326 does not affect the likelihood of USAR-evaluated accidents. The hydraulic performance of the LPSI system will still meet the LPSI delivery curves used in the LOCA analyses, therefore the consequences of a LOCA are still bounded by existing analyses.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The change made to the FCV-326 actuator will not adversely affect the ability of FCV-326 to remain open during the injection phase of a LOCA. The external pressure boundary of FCV-326 is not affected by this modification. The modified travel stop is passive in design, internal to the FCV-326 actuator, and will not make a failure of FCV-326 or any other component more likely. The hydraulic performance of the LPSI system will still meet the LPSI delivery curves used in the LOCA analysis even if one LPSI pump is operating as a result of a single failure. Consequences of an equipment failure are therefore not increased by this modification.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-05, MR-FC-97-018	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** The modified travel stop in FCV-326 does not introduce an initiator for a different type of accident. The stop functions in a passive manner to limit the stem stroke length of FCV-326. The modified travel stop in FCV-326 does not introduce a new failure mechanism for FCV-326 or any other component in the LPSI system. The stop functions in a passive manner, internal to the actuator, to limit the stem stroke length of FCV-326. The design of the travel stop extension is such that a structural failure is not considered credible. The load on the actuator diaphragm case bolts will be higher while FCV-326 is open against the stop, but this load is less than the load on the bolts if the valve is hand-jacked closed using the handwheel. Since the valve is designed to be hand-jacked closed using the handwheel, the modified full-open travel will not create a new failure mechanism or increase the likelihood of failure of the diaphragm case bolts. FCV-326 is normally in its fail-open position during an accident, and this modification does not introduce a new failure mechanism which would adversely affect the ability of the valve to passively remain in the open position during a DBA. The full-open travel limit does not intrude on the throttle range of FCV-326 during normal shutdown cooling.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? No. Technical Specification 2.3(1)i requires FCV-326 to be locked open when the reactor is critical. The basis section for Tech Spec 2.3 states, "Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked." The Tech Spec is intended to ensure that FCV-326 is not closed, because in the closed position it would block the flow path from the LPSI pumps to the RCS. To ensure that this does not occur, the Tech Spec contains an enhanced administrative control to lock the valve open. This control need not change as a result of this modification. FCV-326 will continue to be locked open as before, using the same method (keylock switch).	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-068 CR 199901621 USAR 99-045	**Description:** The following changes are being made to USAR Table 6.3-2 for the containment spray system: (1) Piping design temperature was changed from 300°F to 350°F (2) Suction piping design pressure was changed from 60 psig to 66 psig (3) Add footnote clarifying that 500 psig design pressure applies between pump discharge and HCV-344/345. (4) Valve design pressures were deleted from table (5) Piping wall thicknesses were deleted from table (6) Valve construction items were deleted from table (7) "Piping Construction" changed to "Piping and Valve Construction" (8) Under "Piping and Valve Construction" include exception for flanged equipment for 2 inches and smaller (9) Delete spray nozzle flow and pressure drop as separate items; replace with a single nozzle flow characteristic (12.4 gpm @ 42 psid). (1) Correct piping design temperature is 350°F per work in progress (WIP) number 32314. (2) Correct suction piping design pressure is 66 psig per WIP 32314 and EA-FC-90-006. (3) 500 psig design pressure does not apply to open-ended spray header piping downstream of air operated valves (AOVs). (4)(6)(7) Valve design pressures were deleted because valve design pressure is the same as for the piping system in which it is installed, so there is no need to list valve design pressures separately. The types of welds on pipes & valves are also similar, so there is no need to list "construction" separately for piping and valves. (5) Pipe wall thicknesses were deleted because this level of detail is not necessary in the USAR. Pipe schedules are driven by the applicable pipe classes designated on the system P&ID, which is already referenced in USAR 6.3.2. (8) Some 2-inch and smaller items are flanged, hence, the need for the exception covering flanged equipment in this size. (9) The flow and pressure drop for a spray nozzle is actually a flow characteristic, so this was changed to avoid misinterpretation as an actual expected or required operating point of the nozzle (Ref. WIP 60962).	Table 6.3-2
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? This USAR change does not make an initiating mechanism more likely for a previously analyzed accident. The performance of the containment spray system is not adversely affected by this change. The system will continue to perform its credited safety function after an accident and therefore, accident consequences are not increased.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-068 CR 199901621	**Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR?** No physical or operational change was made to the containment spray system, so this change did not make an equipment failure mechanism more likely. The ability of the system to respond to an equipment malfunction is not affected, therefore the consequences of an equipment failure are not increased as a result of this change.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? No physical or operational changes were made to the containment spray system, so no new accident-initiating mechanism was created and no new equipment failure mechanism was created.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? The degree of redundancy and the hydraulic performance of the containment spray system are not adversely affected by this change, since no physical or operational change was made. The margin of safety as defined in the basis for the relevant Tech Spec (2.4) is therefore not reduced.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-054 DCR 11135	**Description:** Class 4/non-class boundary flags were added at valves AC-387 and DW-207 on P&ID 11405-M-11. Valves AC-387, AC-388, DW-207, AC-433, and AC-434 were changed to show normal closed position. Isometric drawing references near valve AC-214 were corrected. Valve WD-4282 is moved downstream of AC-214, isometric drawing references AC-4288 was added in the location formerly occupied by WD-4282, and isometric drawing references D-4590 was added near valve NG-237. Demineralized water flushing piping to ion exchanger AC-7 resin outlet line was shown as Class 4 (NNS CL-1) but not all of it is associated with radwaste so not all needs to be designated as Class 4 per USAR Appendix N criteria. The new class boundary flags at valve AC-387 and DW-207 make applicable portions of the piping non-class (NNS CL-2) and leave the rest as Class 4 (NNS CL-1). The valve position designation changes make the positions of valves AC-387, AC-388, DW-207, AC-433, and AC-434 consistent with the normal operating alignment for the spent fuel pool (SFP) cooling system per operating instruction valve alignment check list OI-SFP-1-CL-A. The drawing references were put in their proper place so the P&ID is consistent with the isometric drawings.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The components within the scope of the P&ID 11405-M-11 change are not related to an initiating mechanism for a USAR-analyzed accident, so the likelihood of such an accident is not increased. This P&ID change does not adversely affect the ability of safety-related equipment to respond to an accident, therefore accident consequences are not increased.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-054 DCR 11135	**Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR?** No physical change to operational modes of the spent fuel pool cooling system were made. The class boundary changes affect items presently classified as non-CQE. No safety-related items (i.e., class 3 or higher) are affected. This P&ID change did not make an equipment failure mechanism more likely. This P&ID change did not adversely affect the ability to respond to a safety-related equipment malfunction, nor did it affect the current degree of redundancy in safety-related equipment, therefore the consequences of an equipment malfunction were not increased.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? No change to operational modes of the spent fuel pool cooling system was made. The nature of the P&ID change is such that no mechanism is created for a new or different type of accident and no mechanism is created for a new or different type of equipment malfunction.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? The scope of the P&ID change does not include items which are part of the basis for a Technical Specification.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-063 EA-FC-99-018	**Description:** Fuel inspection completed at the end of Cycle 18 operation, using In-Mast sipping and ultrasonic examination, identified 6 failed fuel assemblies containing 13 failed fuel rods; U006 (1 rod), U015 (1 rod), U016 (2 rods), U028 (1 rod), U032 (5 rods), U040 (3 rods). These six fuel assemblies were reconstituted and returned to the core. The failed fuel rods identified were removed and replaced with 13 stainless steel (ss) rods through the process of fuel reconstitution, before being inserted back into the reactor for Cycle 19 operation. This reconstitution process is performed to reduce radioactive material contamination in the reactor coolant system. Engineering analysis EA-FC-99-018 documents the evaluation of, and justification for, Cycle 19 operation with 13 SS rods inserted. Note: The safety evaluation related to fuel reconstitution activities has been addressed in a separate 10 CFR 50.59 safety evaluation completed in support of the Plant Review Committee (PRC) approved procedures.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? Reconstitution of fuel did not increase probability of occurrence of an accident previously evaluated in the USAR since the mechanical integrity of fuel is maintained. Reconstitution of fuel does not have an impact on radiological consequences calculations, since failed rods are being replaced with SS filler rods. Therefore, by removing already failed fuel and replacing them with SS rods, radiological consequences calculations remain bounding, i.e., existing failed fuel would contribute in releases in events such as seized rotor. Therefore, this change will not increase the consequences of an accident.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-063 EA-FC-99-018	**Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR?** Fuel reconstitution itself will not cause a malfunction of a fuel assembly since the fuel is returned to a mechanically sound condition with filler rods. Installation of filler rods in an assembly, as long as it is minimized, does not create hot channels, voiding, or severe shifts in power production. Thus, the potential for an increase in the failure rate of fuel rods from off normal heat transfer conditions was not created. Installation of SS rods therefore, will not increase the probability of occurrence as long as it is minimized and analyzed. Consequences of fuel failures have already been incorporated in USAR Chapter 14 analyses (1% failed fuel). Installation of SS filler rods will not increase the fuel failure rate and thus, not increase the consequences of radiological releases of evaluated fuel failures.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? Operating the reactor core for Cycle 19 with SS rods inserted in fuel assemblies is evaluated and does not change any existing condition as analyzed for USAR Chapter 14 analyses which remain bounding. No new types of accidents can occur which have not been previously evaluated in the USAR because the SS rods perform the same mechanical function in the fuel assembly as the fuel rods. The only significant change is a decrease in the nuclear reactivity of the assembly which is conservative for that assembly (lowers power production). No new modes of fuel failure are created by the presence of SS rods in fuel assemblies. Existing analyses such as the fuel handling accident and seized rotor are bounding with respect to the number of rods assumed failed.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? The presence of SS rods in fuel assemblies during Cycle 19 operation results in a condition that has been evaluated in EA-FC-99-018 for Cycle 19 (and EA-FC-96-049 for Cycle 17), from a neutronics, thermal hydraulics, mechanical design, and radiological consequences perspective. This evaluation concluded that this condition has no impact on any of the Tech Spec basis, or design basis documents, and does not reduce the margin of safety.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-044 Work Request 11593	**Description:** Air flow through the safety injection (SI) pump room exhaust will be measured with one auxiliary building ventilation exhaust fan operating (preferably VA-40B) and no auxiliary building supply fans (VA-35A/B) operating. This measurement was completed with VA-26A/B (SI pump room exhaust charcoal filters) in the filtered mode. The measurement technique was essentially the same as that used in the VA-26A/B filter flow surveillance test. SI pump room ventilating air flow data in this alignment was not available.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? This activity involved operation with one auxiliary building ventilation exhaust fan and no supply fans operating. There are no USAR-analyzed accidents in which the initiator is related to auxiliary building ventilation. The configuration of the auxiliary building ventilation system during this flow measurement will still maintain negative pressure in the auxiliary building and route exhaust flow through the monitored ventilation exhaust stack. The alignment used for this flow measurement is consistent with assumptions used in existing radiological consequences assessments for USAR-analyzed accidents.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? Temperature in the auxiliary building was monitored during the flow measurement to ensure that it did not reach a level which could adversely affect safety-related equipment or instruments. The VA system alignment could be returned to its normal state at any point during the flow measurement if the need arose. This activity did not make a safety-related equipment malfunction more likely. The configuration of the auxiliary building ventilation system during this flow measurement ensured that a radiological release would have followed the monitored release path up the VA stack. Consequences of an equipment malfunction are therefore not increased by this flow measurement.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-044 Work Request 11593	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** The one-exhaust/no-supply VA fan alignment need only be in service long enough to obtain the desired flow data. Because of the brief duration, it is unlikely that auxiliary building temperatures would reach an unacceptable level during the flow measurement. As an added measure, auxiliary building temperatures were monitored while in this alignment so that if an unanticipated temperature condition arose, the flow measurement could be terminated and the VA system returned to its normal state. The flow measurement did not create an initiating mechanism for a different type of accident. Redundant VA fans remained available for restart if the need arose during the flow measurement. This activity did not create a condition which would initiate a different type of safety-related equipment malfunction.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? Auxiliary building controlled area ventilation is not part of the basis of any Technical Specification, therefore Tech Spec margin of safety was not reduced by this activity.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-064 USAR 99-040	**Description:** Text was added to USAR Section 9.10.4.1 to identify that an EOP/AOP Attachment exists to restore ventilation after an accident, with the purpose being to restore forced air flow through the SI pump rooms for equipment cooling in the event of a DBA coincident with a loss of offsite power. USAR Section 9.10.4.1 describes operation of the Auxiliary Building Ventilation System. USAR Section 9.10.4.1 already stated that the SI pump rooms cannot be isolated or the equipment would overheat (this language existed in the FSAR). It has always been OPPD's position that ventilation will be restored to the SI pump rooms. This change is an enhancement to the USAR to identify that operating steps for restoration of ventilation are formally proceduralized, and to more directly state that the ventilation system will be operated after an accident for the purpose of providing forced air flow and cooling to the SI pump rooms. This change will link the EOP/AOP Attachment steps for ventilation restoration directly to the USAR.	Section 9.10
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The additional USAR statements reflect an existing procedure to operate the auxiliary building controlled area ventilation system after a previously analyzed accident has already occurred. Operation of this system is not relevant to the initiating mechanism for a previously evaluated accident, therefore this change does not increase the likelihood of such an accident. Current accident radiological analyses already assume that the auxiliary building ventilation exhaust stack is the release point, so operation of this system after an accident is consistent with existing radiological analyses. This change will therefore not increase accident consequences.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The additional USAR statements reflect an existing procedure for post-accident use. These statements do not make a previously analyzed equipment failure mechanism more likely. In addition, the additional statements do not increase the consequences of a previously analyzed equipment failure. The procedure already contains contingency steps for powering ventilation fans in the event of a DG failure.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-064	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** The additional USAR statements reflect an existing procedure for post-accident use. The procedure for ventilation restoration does not introduce a new accident-causing mechanism nor does it introduce a new equipment failure mechanism.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? The additional USAR statements reflect an existing procedure for post-accident use. Existing Tech Spec operability requirements for equipment are not affected by this change, so this change does not affect any Tech Spec margin of safety.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-012 DCN 4467	**Description:** Circulating water (CW) system pressure switch PC-1902C, drain valve CW-PC-1902C-D, and associated tubing were removed. Related cable 6597 was disconnected and abandoned in place or removed. Root valve CW-144 was capped and closed shut. Operating Instruction OI-CW-2 was revised to reflect the new configuration change. Pressure switch PC-1902C was originally installed to auto start screenwash pump CW-3A or screenwash pump CW-3B on a low screenwash pressure condition. However, the auto start feature has not been utilized since November 7, 1973. The auto start signal was bypassed by utilizing a jumper across the pressure switch output in control panel AI-120. Modification MR-FC-83-101 permanently installed the jumper wire in AI-120. Removal of the pressure switch will require removal of the associated drain valve and capping/closing CW-144. Removal of the drain valve and abandonment of the root valve required that OI-CW-2 procedure be revised by eliminating steps associated with these valves.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? Failure modes are unchanged. There is no change in the CW system performance. The probability of occurrence of an accident is unchanged. The affected system functions/performs as designed. The activity has no interfaces with other systems including environmental radiological release systems.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? No change in CW system operation is expected. There is no increase in the probability of equipment malfunction. The activity has no impact on any safety related equipment. The CW system performance and operation is not changed. There is no change in the consequences of a malfunction.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? The CW system operation and failure modes are unchanged. The modification does not create the possibility of an accident or malfunction of equipment not previously analyzed.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-012 DCN 4467	**Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?** The CW system operation and failure modes are unchanged. The margin of safety is unchanged. The CW systems will operate as designed.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-033 DCR 10168 USAR 99-024	**Description:** Fort Calhoun Station (FCS) is required to have one control site for environmental air monitoring. Currently the Offsite Dose Calculation Manual (ODCM) lists two (FCS is only using the one at Valley Substation). One of these is at North Omaha Station, the other is at the Valley Substation. The Valley site is the newly installed control location and is the official control location. The North Omaha station is no longer required as a control location. Therefore, the ODCM should indicate there is only one control location for this parameter. The receptor site that provided milk samples has discontinued milking operations. At this time there are no milk cows within a 5 miles radius of the plant and, as such, no milk samples are available. A note in the ODCM specified that vegetation samples be obtained if milk samples are not available. Sample station #34 was added due to results of 1998 Land Use survey which indicated that this receptor site was the limiting location for environmental sampling. Other changes include removing some sample sites as active sites and placing them in the table as historical sites. The on site location was added since it is the closest crop location to FCS. This sample site is being used as an information only sample site. Figure 1 was updated to match the changes in Table 4.2. According to NUREG-0472, Revision 3 states that the LLD (lower level of detection) listed is for drinking water samples. However if no drinking water pathway exists, then the LLD of the gamma isotopic analysis may be used." This change to Table 4.3 and 4.4 was made to be consistent with the NUREG. The changes are: Removal of the E. Ellis farm as a controlling location and putting the W. Jones farm in its place along with the requisite direction, distance and controlling age group information. This information was determined during performance of the 1998 Land Use Survey. Change the average annual X/Q. This value is being changed because the 1998 X/Q was > 10% higher that the average annual X/Q. The average annual X/Q is used when projecting gaseous releases and quarterly estimated doses from the station. It is determined by averaging the past 3 years highest sector X/Q for each year. If it varies by more that 10% from the 3 year average the option is available to recalculate the average annual X/Q. The X/Q is calculated using the GASPAR II software program provided by the NRC. The annual average X/Q was recalculated because the 1998 highest X/Q value was ~25% higher than the 3 year average. Change the D/Q value. This value was changed because the controlling location has changed based on the 1998 Land Use Survey. This value is also a three year average. *The D/Q was recalculated because the controlling location has changed based on the 1998 Land Use Survey.*	Section 2.10 Figure 2.10-2

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-033 DCR 10168	**Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR?** The changes to the ODCM are not associated with plant equipment that would increase the probability of an accident. The changes to the ODCM are to areas of the ODCM that are used to monitor FCS radiological impact to the environment during normal operations and following an accident. The changes to the ODCM do not challenge the barriers that guard against uncontrolled release of radioactivity. The changes affect how the environmental monitoring program monitors the environment for radioactivity during normal operations and following an accident.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The changes to the ODCM do not affect equipment or components. The changes involve methodology in the calculation of dose to the public, where samples are collected, and contingencies if samples such as milk are not available.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? The changes to the ODCM do not affect equipment or components. The changes involve methodology in the calculation of dose to the public, where samples are collected, and contingencies if samples such as milk are not available. These actions do not create an accident different than those evaluated in the USAR. In addition, the proposed changes to the ODCM do not affect equipment or components.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? The changes to the ODCM maintain the requirements of Tech Specs 5.16.1 and 5.1.6.2 in that the changes maintain releases to the environment to a minimum and that FCS is able to calculate doses to the public. In addition, it also maintains the requirements of sampling the environment around the plant in order to assess the impact of plant operations on the environment.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-034 DCN 10128 USAR 99-033	**Description:** A threaded pipe plug replaced CCW flow element FE-417 and wire leads were lifted at terminal points 9 and 10 on flow controller FC-417A. This modification made the "CC WATER FROM COIL VA-1B NO FLOW" annunciator (A1/B-1U) unavailable. The flow element could not be repaired within the existing seven day LCO action statement. VA-1B containment cooling coil was returned to service without CCW flow indication (0417 instrument loop). The pipe plug was installed to maintain system pressure boundary and the wire leads were lifted to inhibit the "auto-closure" feature on CCW valves HCV-401A&C. The associated annunciator card was removed to maintain the "black board" concept. This was a temporary configuration change.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The subject configuration change only affects a valve control logic and is not considered to have potential to initiate an event (or accident) previously evaluated in the USAR. As already discussed in this evaluation, the subject configuration change modifies a design feature that is associated with an event that is not discussed in Chapter 14 of the USAR. Accident consequences of USAR-evaluated events are not increased.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The probability of the "auto closure" feature failing to work has not been specifically evaluated or discussed in the USAR. Only the valve's capability to close on a such condition is mentioned. This change does not make any analyzed equipment failure mechanism more likely. Inhibiting the ability of valve HCV-401C to close on a containment isolation actuation signal (CIAS)and CCW low flow will not increase the consequence of an equipment malfunction because the control logic responds to a condition that is not mentioned in the USAR and is outside the system's design basis.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-034 DCN 10128	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** The subject configuration change modifies a logic circuit that responds to a certain plant condition which is outside of the system design basis. It did not have the potential to initiate an event. The subject configuration change did not create the possibility of a malfunction, i.e., no new equipment failure mechanism was introduced. It did however affect the way a component responds to a condition that is outside of the system design basis.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? The associated Technical Specification basis (2.4) did not credit the subject auto-closure logic that was affected by this temporary configuration change.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-058 DCR 11218 USAR 99-037	**Description:** Raw Water (RW) system operations was secured to allow draining of almost the entire system to facilitate removal of valves HCV-2893, HCV-2894 and CW-188. A temporary pipe spool was installed to replace CW-188 and a set of blind flanges was installed to replace both HCV-2893 and HCV-2894. The system was returned to service to support spent fuel pool cooling operations. The system was taken out of service for a second time approximately four days later to remove the temporary pipe spools and blind flanges and to reinstall the original valves. Both total system outages occurred with the plant in Mode 5 (Refueling Shutdown) and Shutdown Condition 2 (reactor defueled, all fuel in spent fuel pool). There was a possibility that the 120°F administrative limit on bulk spent fuel pool temperature may have been exceeded during these two outage evolutions. As such, the consequences of exceeding this limit was evaluated. RW system operation was evaluated as a result of the temporary configuration change on the system. Valves HCV-2893 and HCV-2894 needed to be removed for maintenance. In order to perform required on-line maintenance on either RW supply header, adequate isolation is required for each valve. On-line maintenance cannot be performed on these two valves because of the system configuration and Technical Specification requirements. Valve CW-188 was removed to perform an internal visual inspection per the station preventative maintenance program.	Section 9.6 Section 9.7 Section 9.8

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-058 DCR 11218	**Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR?** Only two accidents (loss-of-coolant accident [LOCA] and main steam line break [MSLB]) could be affected by the temporary loss of the RW system. These accidents however are not credible for the plant mode at the time of this activity. (A Fuel Handling Accident (FHA) does not credit any outside cooling source.) The temporary use of a spool piece and blank flanges in the RW piping system did not increase the likelihood of a previously analyzed accident which is credible for the Shutdown Condition 2 mode in which this configuration existed. The loss of RW could increase the consequences of a LOCA or MSLB if the loss were to occur when these accidents are credible. These accidents are not credible, however, since the activity occurred in Mode 5 and Shutdown Condition 2. The temporary spool piece and blind flanges in the RW piping system has no effect on the consequences of a previously analyzed accident which is credible for the Shutdown Condition 2 mode in which this configuration will exist.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The temporary loss of RW (which indirectly results in a loss of SFP cooling) will increase the average SFP temperature. This temperature increase however, did not create a new failure mechanism for equipment degradation and is within the bounds of the equipment design limits. The temporary spool piece and blank flanges in the RW piping system have equivalent pressure retaining capability as the rest of the system. Equipment failure mechanisms were not made more likely by this configuration. There is no previously evaluated equipment failure which is relevant to the activity of taking the RW system out of service to facilitate maintenance activities while in Shutdown Condition 2. The temporary modified RW system configuration left redundant RW pumps available on opposite electrical buses. Sufficient heat removal capacity existed in the modified configuration to support SFP cooling, which is virtually the only heat load while in Shutdown Condition 2. There is no previously evaluated equipment failure which had its consequences increased by this temporary configuration.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-058 DCR 11218	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** This activity did not have the potential to initiate an accident given the plant mode of operation at the time of the activity. Performance of this activity increased bulk SFP temperature and that of the SFP cooling system. Maximum fluid temperature was maintained below the design temperature of any individual component on the SFP cooling system or any interfacing equipment or component. There is also no adverse affects on other equipment that use RW as its ultimate heat sink such as the containment cooling coils, the shutdown cooling heat exchangers and the ESF pumps. Each of these components remained in service during this activity. Increased SFP bulk temperature did not create a new failure mechanism for any of the related equipment. Maximum operating temperature limits have already been considered and evaluated for the spent fuel pool and its associated cooling system.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? There are no Technical Specifications associated with maximum SFP temperature. Section 4.4.2 does state that pool temperature is maintained below 140°F under "other [not normal] conditions of fuel discharge". This statement is a description of one of the design features of the SFP cooling system and should not be confused as a design or analyzed limit. Section 2.4 is also irrelevant with the plant in Mode 5 and Shutdown Condition 2.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-059 DCR 11217 USAR 99-037	**Description:** The CCW system was taken out of service and partially drained to allow the six CCW pump suction and discharge isolation valves to be removed. Five temporary pipe spools and one set of blind flanges were then installed to replace the removed isolation valves. The system was then returned to service (without AC-3C CCW pump) to support spent fuel pool (SFP) cooling. Approximately four days later, the system was again taken out of service to remove the temporary items and to reinstall the original isolation valves. Both total CCW system outages occurred with the plant in Mode 5 (Refueling Shutdown) and Shutdown Condition 2 (Reactor Defueled, All Fuel in Spent Fuel Pool) and are scheduled to be completed in six hours. CCW system operation will also be evaluated as a result of the temporary configuration change (using pipe spools or blind flanges in place of certain isolation valves and having CCW pump AC-3C unavailable for about four days). The CCW pump suction and discharge isolation valves do not adequately isolate to permit maintenance on the associated CCW pumps. The valves must be rebuilt to permit future on-line pump maintenance	Section 9.6 Section 9.7 Section 9.8
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? Only two USAR Chapter 14 accidents (LOCA and MSLB) could be affected by the temporary loss of the CCW system. These accidents, however, are not credible for the plant mode at the time of this activity. (A Fuel Handling Accident (FHA) does not credit CCW.) The temporary use of spool pieces and blank flanges in the CCW piping system did not increase the likelihood of a previously analyzed accident which is credible for the Shutdown Condition 2 mode in which this configuration existed. The temporary loss of the CCW system could only increase the consequences of a LOCA or MSLB if these accidents were credible during this activity (for which they are not). And, the loss of CCW does not increase the consequences of an FHA. The temporary use of spool pieces and blank flanges in the CCW system had no effect on the consequences of a previously analyzed accident which is credible for the Shutdown Condition 2 mode in which this configuration will exist.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-059 DCR 11217	**Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR?** The temporary loss of CCW (which results in a loss of SFP cooling) will increase the average SFP temperature. This temperature increase however did not create a new failure mechanism for equipment degradation and is within the bounds of the equipment design limits. The temporary spool pieces and blank flanges in the CCW system have equivalent pressure retaining capability as the rest of the system. Equipment failure mechanisms were not made more likely by this configuration. There is no previously evaluated equipment failure which is relevant to the activity of taking the CCW system out of service to facilitate maintenance activities while in Shutdown Condition 2. The temporarily modified CCW system configuration left redundant CCW pumps available on opposite electrical buses. One CCW pump is enough to support spent fuel pool cooling, which is virtually the only heat load while in Shutdown Condition 2. There is no previously evaluated equipment failure which had its consequences worsened by this temporary configuration.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? This activity did not have the potential to initiate an accident given the plant mode at the time of the activity. Performance of this activity will only increase bulk SFP temperature and that of the SFP cooling system. Maximum fluid temperature was below the design temperature of any individual component on the SFP cooling system or any interfacing equipment or component. There is also no adverse affects on other equipment normally cooled by CCW. The containment cooling coils, the shutdown cooling heat exchangers and the engineered safeguards features (ESF) pumps were not in service during this activity. Increased SFP bulk temperature did not create a new failure mechanism for any of the related equipment. Maximum operating temperature limits have already been considered and evaluated for the spent fuel pool and its associated cooling system.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-059 DCR 11217	**Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?** There are no Technical Specifications associated with maximum SFP temperature. Section 4.4.2 does state that pool temperature is maintained below 140°F under "other [not normal] conditions of fuel discharge". This statement is a description of one of the design features of the SFP cooling system and should not be confused as a design or analyzed limit. Section 2.4 is also irrelevant considering the plant mode at the time of the activity (Mode 5, Shutdown Condition 2).	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-031 DCR 10596	**Description:** Plant operating instruction OI-SFP-3A was changed. Precaution and Temporary Modification installation and removal verification signoff steps were added. The procedure was revised to incorporate lessons learned from a previous sluice attempt. Equipment descriptions in checklists and throughout the procedure have been revised to exactly match the equipment descriptions on the equipment labels. In attachment 1, the use of regulated nitrogen gas from bottles for resin agitation was replaced with a higher volume nitrogen supply from the 100 psi nitrogen header. To effectively use the 100 psi nitrogen header for ion exchanger AC-7 resin agitation, an N_2/H_20 sparging/backflush rig was added. Prior to or during the sluice, additional options were given for backflushing the resin sluice line and for supplying sluice water to AC-7 at either the upper or lower lateral retention elements.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? This procedure change improved the spent fuel pool cooling system demineralizer resin agitation/sluice process. In no way is any previously evaluated accident in the USAR affected by this activity. The probability of an accident occurrence is not increased. In addition, the consequences of any accident evaluated in the USAR is not increased.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? No equipment important to safety is adversely affected by the proposed activities. The affected equipment provides only a pressure boundary safety function and no proposed activities challenge this safety function. Thus, the probability of a malfunction of equipment important to safety is not increased and the USAR consequences of a malfunction of equipment important to safety are not increased.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-031 DCR 10596	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** The proposed activity only enhances the resin agitation/sluice process. The changes to the procedure have been carefully considered to ensure that the probability of any accidents are not increased. There is no apparent likelihood that an accident of a different type than previously evaluated in the USAR could result from the proposed activity. No equipment important to safety is adversely affected by the proposed activities. The affected equipment provides only a pressure boundary safety function and no proposed activities challenge this safety function. Thus, there is no credible possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? There are no Technical Specification bases involving margins of safety that are affected by this activity.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-020 ECN 98-200	**Description:** Reactor coolant pump RC-3B motor lower bearing oil level transmitter sensing line was relocated from the oil reservoir to bottom of the oil level indicator. Also, this ECN corrected a drawing error on RC-3B motor upper and lower oil level indicator sensing lines as shown on P&ID D-23866-210-111, sheet 2. Troubleshooting has indicated that the sensing line is subject to air entrainment during motor operation. Therefore, the sensing line was relocated to the bottom of the oil level indicator instead of the oil reservoir.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? No new accident initiating components were introduced. The modification was limited to the lube oil collection system and does not interact with other plant systems. The design basis of the Updated Fire Hazards Analysis has not been changed and the probability of occurrence of a design basis accident is not increased. RC-3B motor lube oil collection or drain capability has not changed and they will function as required during normal and accident conditions. This change did not affect barriers or analysis parameters of previously evaluated accidents. Therefore, the consequences of previously evaluated accidents have not increased.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? Safety related components in the reactor coolant pump motor are the CCW piping and the motor flywheel. These motor components were not affected by this change. The lube oil collection system is designed to withstand design basis earthquakes. Minor oil collection pan modifications have no impact on the oil pan seismic capability. There is no interaction between this change and equipment important to safety. Reactor coolant pump motor lube oil collection system design basis and radiological design basis have not been adversely affected.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-020 ECN 98-200	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** No new failure modes were introduced. RC-3B motor operation, setpoints, or alarms are not changed. Minor modifications to the oil collection pans have no effect on oil collection/drain or seismic capability as previously evaluated in the Updated Fire Hazards Analysis. This change did not modify equipment important to safety. No new lube oil hazards are introduced. Post fire safe shutdown capability was not affected.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? This change complies with the reactor coolant pump motor Updated Fire Hazards Analysis and fire safety margin has not been changed. Technical Specifications margin of safety is not affected.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-039 DCR 10900 USAR 99-028	**Description:** Procedure CH-AD-0004 was changed. As a result, USAR Section 9.2 for the chemical and volume control system (CVCS) was changed in various locations to allow the use Hydrogen Peroxide (H2O2) in the chemical addition section of the CVCS system. Added the option to add hydrogen peroxide to the reactor coolant system (RCS) during plant shutdown when temperature in the RCS is less than 180°F and pressurizer liquid temperature is less than 250°F. Added step to calibrate control panel AI-110 H_2 analyzer with a span gas of 100% versus 10% and then revert to the 10% span after plant startup. Added a caution before Step 7.10.4 about not using a lithiated mixed bed until the nickel cleanup is complete. Changed a caution before Step 7.10.5 to change the amount of hydrazine that is initially added to the RCS from 2 liters to 4 liters. Added "approximately" and "or less" to step 7.2.12 concerning the concentration of H2O2. To allow for the addition of H2O2 to the RCS during shutdown for refueling. Long term reduction of dose during outages by reducing metallic cobalt and nickel from the RCS. Oxygenation of the RCS causes a portion of the RCS crud to slough off internal surfaces. This crud burst can result in an excursion in containment airborne readings when the reactor head is removed and further oxygenation takes place. Injection of H2O2 (forced oxygenation) to oxygenate the RCS allows the crud buildup to be removed by the purification system prior to head removal. This should result in a lower containment airborne activity spike than would be expected using natural oxygenation. This will allow control panel AI-110 to monitor the volume control tank (VCT) hydrogen concentration during plant shutdown and then return to normal after plant startup. This caution was added because if a lithiated mixed bed were placed in service before the nickel cleanup was complete the change in the RCS pH would cause the nickel to plate out and thus cause cobalt activity problems in future outages. This change is being done based on past experience. For the 1996 refueling outage there were 9 liters of hydrazine added. For the 1998 refueling outage there were 7 liters of hydrazine added. These additions were made without detrimental impact upon the RCS. Changing from 2 to 4 liters will help expedite the deoxygenation of the RCS during startup. The requirement to maintain the CVCS ion exchangers isolated during the additions is maintained. Added "approximately" and "or less" to enable the use of other concentrations, if required.	Section 9.2

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-039 DCR 10900	**Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR?** The probability of occurrence of an accident previously evaluated in the USAR was not increased. Limitations on RCS chemistry in regard to dissolved oxygen at temperature ≤250°F are not provided in the USAR. An increase in oxygen concentration at temperatures ≤200°F is not considered an initiator of an evaluated accident. The H2O2 addition did not adversely affect the structural integrity of the affected systems. The consequences of an accident previously evaluated in the USAR was not increased. The process of forced oxygenation of the RCS by addition of H2O2 during shutdown cooling (SDC) results in an accelerated release of activated corrosion products which would occur normally when the RCS is vented to atmosphere. Corrosion products will be removed by the CVCS purification system. As such, this activity did not affect the ability of the plant systems to perform their safety functions and therefore did not increase the consequences of an accident.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The probability of an occurrence of a malfunction of equipment important to safety previously evaluated in the USAR was not increased. The addition of H2O2 is constrained by requirements to comply with Tech Specs and USAR requirements, to prevent creation of explosive mixtures, and to provide chemical control of the H2O2. There are no new failure modes or system interactions introduced by the process. The proposed activity did not increase the consequences of a malfunction of equipment important to safety previously evaluated in the USAR. The RCS is normally oxygenated when SDC cooling is initiated and further oxygenated when the Pressurizer manway and reactor vessel head are removed. Forced oxygenation by the addition of H2O2 simply advances in time the natural oxygenation of the system.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-039 DCR 10900	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** The activity of forced oxygenation of the RCS by addition of H2O2 during SDC did not create the possibility of an accident of a different type than previously evaluated in the USAR. The RCS dissolved hydrogen concentration is limited to preclude explosive mixtures. There are no new failure modes or system interactions introduced by this process. The activity did not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the USAR. Gas binding of system pumps is not a concern. The purification ion exchangers will be bypassed during the H2O2 addition to prevent the rapid release of ions already held in partially spent beds. There are no new failure modes or system interactions introduced by this activity.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? The activity did not reduce the margin of safety as defined in the basis for any Technical Specification. As demonstrated in Table 3-4, Items 1(a)(4) and 1(b)(3), the RCS dissolved oxygen concentration limit is not applicable/required when RCS temperature is ≤250°F. The RCS temperature will be <180°F and pressurizer liquid temperature will be <250°F prior to the addition of H2O2 when the Tech Spec limit on oxygen did not apply. The oxygenation of the RCS coolant would occur normally when the RCS is vented to the atmosphere. Tech Spec 2.1.3(1)a. requirement of dose equivalent iodine being ≤ 1.0 μCi/gm bounds the iodine requirement required to H2O2 addition. The RCS chemistry limits were re-established prior to startup, therefore the forced oxygenation activity has no effect on the Tech Specs or its basis.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-026 DCN 2825 USAR 99-020	**Description:** The existing 161kv and 345kv electrical distribution systems mimic buses and associated indicating lights on the north end of control room panel CB-20 were removed. Switchyard status indication is displayed on the control room emergency response facility computer system (ERFCS) terminals. Switchyard control building house service power transformer T1B-3D, currently powered from the 4160 V bus 1A3 was removed. As a result, plant 4160 V circuit breaker 1A3-15 was opened, racked down, and removed from the switchgear room, and the associated control room switch "CS/T1B-3D", indicating lights, mimic and ammeter "A/T1B-3D-1,2,3" were disconnected and removed from CB-20. Fuses supplying control power to the 161kv mimic were removed. These fuses also power the 161kv line fast transfer control circuit. Control power to the fast transfer circuit was re-established and connected directly to instrument power feed from 125 VDC bus # 2, AI-41B, breaker 16 through main control power fuses in CB-20. Deleted ERFCS points C-Y3313-1 & 2, C-Y3314-1 & 2, and C-Y3315-1 & 2 associated with the 345kv circuit breakers 2 and 6. This modification was required for the Switchyard Upgrade Project which is being done by OPPD-Substation Dept. These plant changes were required to bring the plant into conformance with the equipment being installed under the switchyard upgrade project. Due to limited space on CB-20 for the new switchyard mimic breaker configurations, control room ERFCS terminals were utilized to display the 345/161kv breaker's status. District policy is now to provide both normal and emergency substation house service feeds from switchyard derived power sources. To eliminate a set of fuses that are no longer required hence, the reliability of the fast transfer circuit were enhanced. As a result of the new switchyard 345kv configuration these ERFCS points became meaningless. ERFCS indication points for the 345 kv breakers 4 and 5 are still available and were not deleted.	Section 8.2 Figure 8.1-1

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-026 DCN 2825	**Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR?** The plant electrical distribution system performance and its design basis were maintained. The modification is dealing with system components that have no impact on nuclear safety. Status indication of 345kv breakers 4 and 5 and 161kv breakers 110 and 111 are displayed on CB-20. Therefore, failure of the control room ERFCS terminals or loss of mimic bus display on these ERFCS terminals did not hinder Operator's knowledge of the status of switchyard breakers that directly interact with the plant electrical system. The modification did not alter the design of any engineered safety feature or physical barriers guarding against the uncontrolled release of radioactivity. The modification did not affect fuel integrity, pressure boundary, containment integrity or radiological release monitoring. The modification did not increase the consequences of an accident previously evaluated in the USAR.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The modification shifted the switchyard 345/161kv mimic buses indication from CB-20 to the control room ERFCS terminals, and deleted the switchyard house power feeder. The design changes that were implemented under this modification did not increase the challenges to the plant electrical distribution system or any other equipment important to safety. The design did not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR. This modification did not result in increased radiological consequences from a malfunction of equipment important to safety as evaluated in the USAR. The plant electrical distribution system performance and design basis were maintained.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-026 DCN 2825	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** The plant electrical distribution system performance and design basis were maintained. There was no potential for any new accident introduced by this modification. This modification did not introduce any new failure modes to the plant electrical distribution system. Therefore, no possibility of a malfunction of equipment important to safety of a different type then any previously evaluated in the USAR was created.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? Technical Specification 2.7 addresses the availability of the plant electrical system. This modification did not alter the performance, design basis failure modes, reduce the margin of safety or the independence/redundancy of the electrical system as described in Technical Specifications, Section 2.7.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-051 DCN 10159	**Description:** Replaced steam trap carbon steel steam discharge piping with stainless steel pipe. Replaced carbon steel constant vent piping for the low pressure feedwater heaters with stainless steel piping. During past outage inspections and during the past cycle, the subject pipe was found to be thinning due to flow accelerated corrosion (FAC) and in some cases was found to be leaking. The replacement with stainless steel as opposed to the in-kind replacement with carbon steel pipe will prevent the piping from being susceptible to FAC and eliminate it from the FAC project.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The piping in question does not perform a safety function as defined by the USAR and is not an initiator of any evaluated accidents. The piping in question does not perform a safety function in response to accidents evaluated in the USAR.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The piping is not an initiator for any equipment malfunctions evaluated in the USAR. The piping does not perform a safety function in the mitigation of consequences associated with equipment malfunctions evaluated in the USAR.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? The piping in question is not in a position to be, nor is it addressed as, an accident initiator. It has no safety function and does not support any safety related component or system. The piping in question does not directly or indirectly affect any safety related components or systems and there are no safety related components in the area such that the failure of the pipe could cause damage to the component or system.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-051 DCN 10159	**Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?** The material associated with the vent and drain piping in the secondary system has no bearing on the margin of safety defined in the Technical Specifications.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-03 DCN 2830 USAR 99-009	**Description:** Modification MR-FC-98-006 evaluated long-term solutions to address reliability and maintainability problems experienced with HCV-921 and HCV-922, the radiation monitor RM-064 sample isolation valves. The modification moved main steam valves MS-101 and MS-103 so that they provide isolation for HCV-921 and HCV-922. This change facilitated on-line repair of HCV-921 and/or HCV-922 by providing two-valve isolation of these valves from the main steam headers. This modification replaced the following valves: MS-100, -101, -102, -103, -181, and -182. The modification replaced the existing vertical thermal expansion loop with a U-shaped horizontal expansion loop. This modification improved the availability of RM-064, the Post-Accident Steam Line Radiation Monitor, by facilitating on-line repair of HCV-921 and HCV-922.	Figure 5.9-13 Sheet 62 Figure 5.9-13 Sheet 63
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? This modification installed new CQE components in the main steam system. These components meet the design, material and construction standards applicable to the main steam system (i.e., USAS B31.7, ASME Section III, Class 2, ASME Section XI). Since these standards are met, the modification did not increase the probability of occurrence of an accident previously described in the USAR. The worst case accident related to the systems and components affected by this modification would be a catastrophic failure of the one-inch pipe connected to the main steam headers upstream of valves MS-100 and MS-102. This type of failure would initiate an uncontrolled steam leak in Room 81. This event is bounded by the main steam line break accident postulated in Section 14.12 of the USAR, thus this modification did not increase the consequences of an accident previously described in the USAR.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-03 DCN 2830	**Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR?** The systems affected by this modification provide no safety functions (i.e., functions required for safe shutdown) and the modification introduced no new system interactions that could affect equipment required for safe shutdown. This modification did not affect equipment important to safety (i.e., equipment required for safe shutdown), thus it does not increase the consequences of a malfunction of safety-related equipment.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? The modification introduced no new system interfaces and no new hazards. It does not introduce any new potential accident initiating mechanisms, thus it does not create the possibility of an accident different from those previously considered in the USAR. The modification introduces no new system interactions and does not affect equipment important to safety (i.e., equipment required for safe shutdown), thus it does not create the possibility of a malfunction of equipment important to safety different from those previously evaluated in the USAR.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? This modification affects piping related to RM-064. Technical specification (TS) 2.21 requires that RM-064 be maintained in an operable condition. RM-064, and other post-accident monitors, provide useful information to operators during and following an accident. It is desirable that RM-064 be maintained in an operable condition, but RM-064 is not required for safe-shutdown and provides no control or safety functions. Since no safety functions are affected by this modification, no margin of safety is reduced.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-056 DCR 11119 USAR 99-030	**Description:** The reactor core loading is being changed for Cycle 19 operation. A change to the technical data book (TDB) Section VI (COLR) is being made to support the Cycle 19 core design and reload analysis. Pages 1 and 5-18 were updated with the new cycle number. Section 3.0 and Figure 1 were updated for the new thermal margin low pressure (TM/LP) coefficients that are part of the P_{VAR} equation. Section 8.0 was updated for the new peaking factor limits. Section 14.0 was updated to reflect a more negative end-of-cycle moderator temperature coefficient (MTC). Figure 4 was updated to remove the need to operate at All Rods Out (ARO) above 80% power. Figure 5 was updated for the new allowable power levels and axial shape indexes (ASIs) for when Better Axial Shape Selection System (BASS) is inoperable. Figure 7 was updated for the new allowable trade-off curves for peaking factors allowed as a function of power. Figure 9 was updated with the new departure from nucleate boiling ratio (DNBR) monitoring limit curve. The USAR was updated to reflect new analyses performed for the Cycle 19 core design. Section 3 Table of Contents and List of Tables were updated for the new cycle number. The List of Figures was updated to reflect a burnup change in the title of Figure 3.4-6. Section 3.2 was updated to remove duplicate wording. Section 3.4 was updated with the new core loading design and applicable characteristics determined in engineering analyses EA-FC-98-036 (Design Depletions) and EA-FC-98-037 (Physics Analyses Base Cases). Section 3.6 was updated with the thermal hydraulic characteristics used in the Cycle 19 Transient Analyses and developed in EA-FC-98-045. Section 3.7 was updated with the fuel mechanical design characteristics for the new core. Section 3.9 was updated to reflect new references. Section 7.5 was updated to add a new sub-section describing MINI-CECOR and BASSS (A/R 19574). Chapter 14 List of Figures was updated to reflect typical figures in the title of Figures 14.2-1 through 14.2-8. Section 14.1 was updated with general information applicable to all transient analyses. Section 14.2 was updated with results from the reanalyzed CEA Withdrawal Event (EA-FC-98-047). Section 14.4 was updated with results from the reanalyzed CEA Drop Event (EA-FC-98-048). Section 14.6 was updated with results from the reanalyzed RCP Seized Rotor Event (EA-FC-98-053). Section 14.9 was changed to remove an incorrect unnecessary reference from Table 14.9-1. Section 14.11 was updated with results from the reanalyzed Excess Load Event (EA-FC-98-052). Section 14.12 was updated with results from the reanalyzed Main Steam Line Break Event (EA-FC-98-050). Section 14.13 was updated with results from the reanalyzed CEA Ejection Event (EA-FC-98-054). Section 14.15 was revised to (1) correct the SIT injection pressure used in the large break LOCA analysis from 255 psia to 225 psia,	Section 3.2 Section 3.4 Section 3.6 Section 3.7 Section 3.9 Section 7.5 Section 14.1 Section 14.2 Section 14.4 Section 14.6 Section 14.9 Sec. 14.11 Sec. 14.12 Sec. 14.13 Sec. 14.15 Sec. 14.22 Figure 3.4-1 Figure 3.4-2 Figure 3.4-4 Figure 3.4-5 Figure 3.4-6 Figure 3.4-7 Fig. 14.12-1 through 14.12-13

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-056 DCR 11119	**Description:** (continued) (2) revise the small break safety injection tank (SIT) injection pressure as well, (3) add references for (1) and (2) above and the calculation of the SIT pressure total loop uncertainty, (4) Correct table headings for Table 14.15-4, and (5) correct the value for emergency core cooling system (ECCS) delivery delay. Section 14.22 was updated with results from the reanalyzed RCS Depressurization Event (EA-FC-98-051). All of the USAR changes were compiled and examined for consistency in EA-FC-98-056, "Cycle 19 Reload Analysis USAR Update." The new core design will ensure that enough energy is present during Cycle 19 operations. Changes in the core loading pattern produce changes to the core operating limits, which provide sufficient margin to protect the plant against all of the analyzed USAR Chapter 14 transients. The USAR updates for the Cycle 19 reload reflect changes to the physics parameters, thermal hydraulic characteristics, and limiting transients that were required to be reanalyzed. These changes were used to assess the impact of the new core design and subsequent plant responses to unusual conditions. Revised the large break LOCA SIT injection pressure to that used in the analysis and justify the same pressure for the small break LOCA.	
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The probability of occurrence of an accident evaluated in the USAR is not affected by this new core design or the subsequent required changes to the COLR or USAR. Shutdown margins and times required for operator action are maintained at the same values as in previous reload analyses. The accident consequences from the new core design are evaluated in the appropriate engineering analyses. The LOCA evaluation was performed by Westinghouse. All accident consequences are bounded by the appropriate limits and are evaluated in accordance with NRC approved methods.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-056 DCR 11119	**Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR?** The change to a new core design does not create the possibility of a different malfunction than that already analyzed for the USAR Chapter 14 events. No new materials are being introduced to the RCS. Necessary and bounding equipment malfunctions were analyzed as part of the Cycle 18 Reload Analysis. The malfunctions examined include a MSLB, LOCA, CEA Ejection, Seized RCP Rotor, CEA Drop, CEA Withdrawal, Excess Load Increase, and RCS Depressurization. Other events are bounded by previous analyses.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? No new credible failure types are created by the new core design. Applicable analyzed enrichment limits, fuel design limits, and radiological consequences are bounded by previous analyses. Changes to the COLR allowable operating space ensure that the Specified Acceptable Fuel Design Limits (SAFDLs) for plant transients/accidents are retained within the regulatory allowable limits. These changes did not create a different type of malfunction of equipment because system responses are within the range of previously analyzed values. Core enrichment limits are maintained to the analyzed limits. Fuel design margins are protected by plant trip systems and operating limit restrictions.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? No changes were required to the Technical Specifications (TS) as a result of this new core design. Approach to the TS limits was evaluated as part of the Cycle 19 Reload Analysis. Maintenance of the margin of safety is guaranteed by the evaluations performed for this new core design. The fuel design limits are protected by plant trip systems and operating space limitations. Thus, no reduction in the margin of safety is possible.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-053 USAR 99-031	**Description:** USAR Sections 6.2.3.1 and 6.2.5 were revised to clarify the difference between the safety injection and refueling water tank (SIRWT) minimum inventory credited in the LOCA safety analysis and the Technical Specification limiting condition for operation (LCO) requirement. The application of instrument uncertainty was addressed.	Section 6.2
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? No changes were made to the Tech Spec SIRWT minimum inventory requirement. The intent of this USAR change is to provide clarification on the SIRWT volume design basis. The minimum SIRWT water inventory requirement per Tech Spec 2.3.1 was not changed.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? No physical or operational changes are being made to any system or component. Design basis information on SIRWT inventory is being added. The minimum SIRWT inventory required per TS 2.3.1 has not changed.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? No physical or operational changes to plant components or systems were proposed. This change provides design basis information on the minimum SIRWT volume. This change has no impact on the performance or operability of the SIRWT. Therefore, no malfunction of equipment important to safety of a different type than previously evaluated in the USAR is created.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? This change is consistent with the applicable sections of the Technical Specifications.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-025 USAR 99-022	**Description:** Section 14.15.7.1 of the USAR was revised to clarify the operability requirements for the emergency core cooling system (ECCS) during shutdown conditions. USAR Section 14.15.7.1 addresses a LOCA during shutdown and makes the statement, "The containment pressure high signal (CPHS) will still initiate safety injection and the automatic sequences involved". However, the USAR does not define what constitutes "safety injection" during shutdown conditions. This change clarifies the ECCS equipment requirements necessary to mitigate a LOCA while shutdown. These requirements are based on evaluation performed by an independent vendor which establishes the minimum ECCS equipment necessary to meet ECCS acceptance criteria of 10 CFR 50.46 for a LOCA while shutdown.	Section 14.15
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? This USAR change defines the minimum SI equipment requirements for a LOCA during shutdown to ensure compliance with ECCS acceptance criteria of 10 CFR 50.46. Placing this statement in the USAR clarifies the SI equipment requirements and does not increase the probability of occurrence of an accident previously evaluated in the USAR. Analysis by ABB-CE establishes the minimum SI equipment operability requirements to meet ECCS acceptance criteria for a LOCA during shutdown. These ECCS equipment operability requirements while shutdown are being incorporated into the USAR to establish a design basis. These requirements have been proceduralized to ensure compliance. This activity will ensure the availability of equipment necessary to mitigate the consequences of an accident (Shutdown-LOCA) as described in the USAR.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-025	**Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR?** The operability requirements outlined in this USAR change are consistent with the Technical Specifications and the Operating Instructions. This change does not increase the probability of an occurrence of malfunction of safety related equipment since there are currently no requirements for equipment oberability during these lower operating modes. This change is conservative since it requires certain SI equipment be available during these modes of operation. The USAR change will incorporate ECCS equipment operability requirements during shutdown conditions. The intent is to ensure the necessary equipment is available to meet the ECCS acceptance criteria in accordance with 10 CFR 50.46. This change will not increase the consequences of a malfunction of equipment important to safety.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? This activity is intended to clearly define the ECCS equipment operability requirements for a shutdown LOCA. ECCS equipment requirements for a LOCA when the reactor is critical are currently addressed in the Technical Specifications. This activity does not create the possibility of an accident of a different type. It will ensure the minimum required SI is available during the operating modes in question. Equipment and system performance is not changed by this activity.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? The margin of safety in the Technical Specifications is not reduced by this activity. The availability of the LPSI and HPSI systems as required by the T.S. 2.1.1, 2.3, and 2.15 are not be affected. This guidance provides clarification for minimum SI equipment availability during those operating conditions not addressed in the Technical Specifications.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-017	**Description:** Modification MR-FC-98-016 made the following changes. Flexible hose connects the rigid tubing from the hydraulic reservoirs to the valve blocks on each of the four snubbers installed on the four reactor coolant pumps (RCPs) and at the four snubber locations on each steam generator. - All tubing in the system will be stainless steel tubing. - The hydraulic reservoirs for the snubbers on RCPs RC-3B and RC-3D are relocated to a wall inside each pump bay. - Leak inspection of the hydraulic tubing/reservoir system has been performed. - The flexible hoses simplified normal maintenance/surveillance tasks which require removal and re-attachment of the lines to the snubbers. - Relocating the reservoirs improved access for maintaining their oil levels and shortens the tubing runs and makes their configuration conform to PCPs RC-3A and RC-3B. - Stainless steel tube is higher strength and less susceptible to damage during maintenance than the originally installed copper tubing. - Leak inspection ensured the integrity of the fluid system.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The subject hydraulic snubbers are design basis accident mitigating components and are not accident initiators. Because the snubbers will be fully functional within the time restrictions imposed by Standing Order G-44, there is no effect on accident consequences where the snubbers must function to mitigate the effects of design basis events.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The probability of malfunction of the subject snubbers or the RCS pressure boundary is unaffected during construction because the two-hour time limitation for snubber inoperability was maintained using temporary reservoirs as needed on the affected snubbers. The integrity of the RCS piping in meeting dynamic loads during cold shutdown conditions is not compromised during construction.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-017	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** The subject snubbers are fully operable and able to accommodate design basis loads with a temporary reservoir. The function and capacity of the snubbers is unchanged with temporary reservoirs; therefore their reliability is unchanged, making a malfunction of the RCS piping no more likely than with the normal reservoirs in service.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? No Technical Specification margins of safety relate to equipment supports or hydraulic snubbers.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-029	**Description:** Valves CA-274, 275, 276, 386,387,388 are currently small needle or gage valves which were replaced with ball valves under this modification (SRI DCN 10115). Associated fittings were also replaced to accommodate the change. The valves are small instrument (pressure gauge/pressure switch) isolation valves on the plant air compressors CA-1A, B, and C. Installation was performed as part of a maintenance activity. The P&ID drawings 11405-M-263 Sheet Cover and Sheet 1 Compressed Air P&IDs were revised to depict the change in the type of valves. The valves were replaced to facilitate maintenance and calibration activities on pressure switches on the air compressors.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The compressed air system has no part as an accident initiator and therefore this change did not increase the probability or consequences of an accident previously evaluated in the USAR. Compressed air is not needed to provide a power (pressure) source for any accident mitigating equipment as described in USAR Chapter 14.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The compressed air system has no accident mitigating function, and this configuration (and USAR) change has no effect on the fail position of air-operated valves or their ability to be positioned by local accumulators supplied by the instrument air system. This configuration (and USAR) change has no effect on the fail position of air-operated valves or their ability to be positioned by local accumulators supplied by the instrument air system.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-029	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** The sensing lines which contain the subject valves are connected to the air receivers. The air receivers serve both the compressed air and the instrument air subsystems. Failure of a sensing line or a valve could prevent proper compressor start/stop. However, the receivers and sensing lines have no safety function because instrument air has no safety function. A loss of instrument air is already an input for postulated USAR Chapter 14 accidents and abnormal events such as loss of off-site power.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? No Technical Specification setpoint or safety limit is based on any performance parameter of the compressed air system.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-045	**Description:** RC-146 is the root valve for the liquid leg for LT-106, a pressurizer level transmitter which provides a signal for LI-106, pressurizer level indicator located on CB-3. LI-106's main purpose is to provide non-safety-related indication of pressurizer level during cold shutdown drain/fill. RC-146 is located on the underside of the pressurizer and was replaced with a smaller root valve located outside the pressurizer skirt as a substitute replacement item. The P&ID drawing, which is referenced in the description in the USAR, was changed to show the line size in which RC-146 is located.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? Pressurizer level instrument LI-106 (LT-106) is installed mainly for non-safety-related cold pressurizer level indication. This device has no part in initiating any USAR Chapter 14 event. Because the pressure boundary integrity is equivalent to the original, probability of an RCS loss of coolant event is no more likely. The consequences of the failure of the pressure boundary containing the root valve is within the scope of the current LOCA analysis.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The changes to the root valve for the level transmitter LT-106, which does not provide an automatic safety signal, will not affect the instrument uncertainty, performance or response time of the instrument. The RCS pressure boundary integrity is unaffected. The consequence of a malfunction is an incorrect signal created by LT-106 and indicated by LI-106. This failure is the same one that would be assumed for the original design for this non-safety-related loop. Other pressurizer level transmitters use the same type of sensing line and instrument valve configuration as proposed for LT-106 without having design failures.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-045	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** Only a pressure boundary failure or plugging of the sensing line is credible. Leakage of the tubing is accounted for in existing LOCA analyses. Plugging of the line is of little consequence. The connection source for the transmitter LT-106 is unchanged from the original. Only minor changes to the path for the sensing signal were made. Pressure boundary integrity is equivalent to the existing.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? The subject instrumentation did not change in calibration values and no devices in the electrical loop were changed. There was no change to the hydraulic performance of the system. Therefore, the response and performance of the loop was unchanged. No Technical Specification margin or setpoint is affected.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-071	**Description:** P&ID drawings 11405-M-253 sheet 3 and Cover and 11404-M-255 sheet 1 have been revised to depict the configuration change to the isolation of feedwater heater shell and channel drain valves and piping as authorized by procedure PED-GEI-35. A downstream isolation valve was installed on the 2" header which receives the drainage from the (normally closed) shell drain valves on feedwater heaters FW-16A and B. The 1" channel drain on each heater, and the channel thermal relief valve, which currently connect to the 2" header, have been changed to run together in a new 1" line separately to a point in the 2" line near the floor drain. Feedwater heaters FW-15A and B have had an isolation valve placed downstream of the shell drains and the channel drain. The configuration change was required to provide additional isolation to prevent steam cutting of the seats in the leaking isolation valves.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The subject secondary side located drain valves and piping and thermal relief valves play no part in any analyzed accident event as described in Chapter 14 of the USAR. Failure of a normally closed feedwater heater drain valve or the non-availability of a thermal relief valve has no impact on any analyzed accident. The only related event is an anticipated operational occurrence which involves the loss of feedwater heating. A failure of a drain valve, which is no more likely after this change, is within the assumptions for that event.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? No equipment which has a safety function is located in the area, and there are no interactions between safety-related devices and the feedwater heaters.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-071	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** Other than some failure affecting the temperature of the feedwater to the steam generators, which was already discussed above, the sub-system being modified has no interaction with primary system or accident mitigating systems. The feedwater heaters, the thermal reliefs, and the drain valve subsystems are secondary side components. The performance of the extraction steam and feedwater systems will be unaffected because of this change.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? The operation of the drain valves or the thermal reliefs on the feedwater heaters has no effect on safety limits, setpoints or equipment operability.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-070	**Description:** As identified in Quality Assurance (QA) surveillance report E6-98-2, changes to P&ID drawing 11405-M-253 sheet 4 (and cover sheet as applicable) were needed to correct errors. The changes, which are minor, represent the as-built configuration of the system. The changes ensured the P&ID drawing remains accurate.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? These P&ID drawing clarifications and corrections are related to the auxiliary feedwater (AFW) system, which is an accident mitigating system. The AFW system is not evaluated as an accident initiating system in the USAR. The minor drawing changes have no effect on AFW system performance or test acceptance criteria.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The reliability of the AFW system is unaffected by the changes to the P&IDs which are administrative in nature. No physical changes are being made to the system. These drawing changes affect no operational/performance aspects, surveillance testing acceptance criteria, or any corrective maintenance or preventive maintenance tasks.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? These drawing clarifications and corrections are all supported by the existing design basis and have no effect on equipment performance. The AFW system reliability and performance are unaffected by the essentially administrative changes to the AFW P&ID drawings.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-070	**Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?** No Technical Specification margins are impacted by the drawing changes which involve class boundary flag clarifications, quick connects or caps on drain and vent valves, and a clarification of the packing gland leak-off line size on AFW pump FW-10.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-037	**Description:** Instrument air P&ID, 11405-M-264 sheet 3, was revised to reflect engineering assistance request EAR 97-120 which added a root valve in the IA riser upstream of the speed control loop and the minimum flow recirculation valve for FW-10, the steam turbine driven AFW pump. The added isolation valve facilitates maintenance on the pneumatic components of the speed control loop for FW-10.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The auxiliary feedwater system and FW-10 are part of an accident mitigating system for several USAR Chapter 14 events. Failure of this system is not an accident initiator. The auxiliary feedwater system and FW-10 are needed for decay heat removal when the main feedwater system is unavailable. The performance of FW-10 is unaffected by the addition of a passively open root valve in the IA supply to the speed controller and the minimum flow recirculation valve.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? A loss of air to the speed controller and the recirculation valve is assumed by design, however, having the IA supply available is highly desirable. The reliability of the IA supply is unchanged with the added normally open root valve. The presence of the root valve has no effect on the assumed loss of Instrument Air to FW-10's speed controller which causes the pump to run at maximum governed speed, or to FW-10 recirculation line isolation valve FCV-1369 which fails open on loss of IA (but has an accumulator for repositioning if needed).	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? The root valve is a passive component and is positioned closed only for maintenance. Its normal position is open which is controlled by an operating instruction valve lineup. Loss of instrument air is assumed in the existing plant design, however, loss of IA is no more likely with the addition of the root valve.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-037	**Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?** No Technical Specification safety limits or setpoints involve root valves on instrument air supply risers.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-065 USAR 99-042	**Description:** USAR Section 3.4, Technical Data Book (TDB) Index, Figures, Tables, and Worksheets contained in Sections II and V of the TDB, and surveillance test OP-ST-RX-0001 were revised to reflect Cycle 19 reactivity parameters.	Section 3.4
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The data, figures, and tables were produced in engineering analysis EA-FC-99-010 using computer codes and methodology approved by the NRC. Incorporation of this information into the USAR, TDB, and procedure does not affect any plant hardware or process parameters which could increase the probability of an accident previously evaluated in the USAR. The purpose of these changes is to update the data, figures, tables, and worksheets to reflect reactivity parameters which are specific to Cycle 19 operation. Upon review of the accidents and their consequences evaluated in USAR Chapter 14, the proposed changes do not affect any of the assumptions, data, or models which were used to determine the radiological consequences of previously evaluated accidents. The accident radiological consequences documented in the USAR remain bounding. Therefore, the proposed activity does not increase the consequences of any previously evaluated accidents.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The USAR accident analyses were researched to determine if the proposed changes had any direct or indirect impact on equipment important to safety. No impacts were identified. Therefore, the proposed changes to update the data, figures, tables, and worksheets of the USAR, TDB, and procedure do not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-065	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** The updates of the data, figures, tables, and worksheets of the USAR, TDB, and procedure do not create any credible equipment failures that could initiate a sequence of events or system interaction leading to a new accident. The purpose of these changes is to update the data, figures, tables, and worksheets to reflect reactivity parameters which are specific to Cycle 19 operation. Therefore, the proposed changes do not create the possibility of an accident of a different type or the possibility of a malfunction of equipment of a different type than any previously evaluated in the USAR.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? The Technical Specifications in Section 2.10, Reactor Core, and 3.10, Reactor Core Parameters, and their respective bases were reviewed to determine the impact on the operating limits and the margin of safety. No impacts were identified. Therefore, the proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-040 USQD 99-069 USAR 99-051 USAR 99-038	**Description:** **DESIGN:** 1. 161kv Switchyard (sub 1251) was upgraded to a two ring / three bay breaker-and-a-half scheme. This includes the replacement of existing breakers 109, 111 & 112, and the 161kv west bus from a 3000 Amp rating to a 5000 Amp rating to match the overall buswork rating. New 161kv circuit breakers 104, 105 & 106 was installed to create the new 161kv bay #3, and new breaker 108 complete Bay 2 between existing breakers 107 and 109. 2. A new 161kv transmission line (ckt. 1570) was installed from Blair sub 1226 to FCS sub 1251 and will connect to 161kv Bay 2 between breakers 108 & 109. Existing line (ckt. 1588) from FCS sub 1251 to Cargill sub 1298 was moved to connect in new 161kv bay 3 between new breakers 104 & 105. 3. 345kv Switchyard (sub 3451) was upgraded to a two ring / three bay breaker-and-a-half scheme. This will replace existing breaker 3451-6 and install new breakers 3451-1, -3, -10, -11 & -12. 4. Circuit 3431 to the Raun substation in Sioux City was disconnected from the 345kv east bus and reconnected in new bay 4 between new breakers 3451-10 & 3451-11. 5. Two (2) new 345kv/161kv Autotransformers (500/560MVA) were installed in the FCS switchyard and connected between the 345kv switchyard and the 161kv switchyard. Autotransformer T-3 was connected on the high side in 345kv bay 1, between breakers 3451-1 & 3451-2 and on the low side in 161kv bay 2 between breakers 107 & 108. Autotransformer T-4 was connected on the high side in 345kv bay 4, between breakers 3451-11 & 3451-12 and on the low side in 161kv bay 3, between breakers 105 & 106. **INSTALLATION:** The installation of all new and replaced equipment for this upgrade was completed in numerous phases under numerous switchyard configurations. The installation of this upgrade was done in accordance with OPPD-Substation design documents and specifications. The installation was done under the direction of OPPD-Substation Dept. personnel and the contracted A&E firm. **TEST:** All switchyard equipment installed by this project was tested under the direction of OPPD-Substation Dept. personnel and the contracted A&E firm to verify that the design and installation requirements have been done in accordance with design documents and specifications.	Section 1.2 Section 8.2 Figure 8.2-1 Figure 8.2-2

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-040 USQD 99-069	**Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR?** No USAR evaluated accident scenarios are affected by a change in the number of available off-site power sources into or within the FCS switchyard. Therefore, there can be no increases in the probability of such accidents. No postulated accident scenarios are affected by this switchyard upgrade project, therefore, there can be no increase in the consequences of any evaluated accidents.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? This switchyard upgrade project does not affect any equipment that is important to safety. Switchyard equipment is not safety related and this project does not adversely impact any operating parameters for plant equipment. The minimum available and separate off-site power supplies to the station are not being changed or reduced. Since this switchyard upgrade project does not affect any safety equipment, there can be no increases in the consequences of any equipment malfunctions.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? By changing the number of power sources within and into the FCS switchyard, this switchyard upgrade project does not create any new accident scenarios, nor does it affect any previously postulated accidents. This project in no way reduces the required number of off-site power sources or their availability. This project does not affect any plant safety equipment, therefore it cannot create any additional equipment malfunctions that are important to safety. Since the off-site power requirements are not being reduced or changed, the USAR intent is not affected.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-040 USQD 99-069	**Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?** Technical Specifications do not cover switchyard equipment, therefore no plant safety margins are impacted by this switchyard upgrade project. The small increase in FCS safety bus voltage is well within the required voltage limits for proper equipment operation inside the FCS.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-041	**Description:** A blow-down valve and drain piping was added to the auxiliary steam strainers. The PM program requires steam strainers and traps be blown down three times during the heating season. Most of these strainers and traps do not have blow-down valves. Blowing these traps and strainers requires removing a plug or opening the integral blow-down valves which blows steam and dirt on the operators or maintenance people. Installing blow-down valves and drain piping will make the configuration much safer.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? This USAR referenced drawing change is supported by the existing design basis and has no impact on equipment performance. Configuration changes to the auxiliary steam system are not safety related. There is no change in the operation of the plant heating system or the main steam system. The plant heating system is not safety related. The configuration change and drawing revisions involve enhancements to the heating system and main steam system which have no part in accident initiation.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? There is no effect on main steam system performance in providing a decay heat removal path if needed. The auxiliary steam system has no safety function.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? This USAR referenced drawing change is supported by the existing design basis and has no impact on equipment performance. The operation of the plant heating system is not changed. The normal operation of the main steam system is unchanged also. The blow-down valves and piping being installed will enhance personnel safety during the performance of preventative maintenance.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-041	**Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?** There is no change in the margin of safety or any setpoints defined in any Technical Specification as a result of these equipment and drawing changes.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-048	**Description:** P&ID drawing 11405-M-257 and related isometric drawings were revised to show actual field conditions. The 1-inch valves and pressure gauge with root valve were not shown on plant drawings. The valves are a vent valve and a drain valve on the screen wash backwash header. The pressure gauge is on the same header.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The USAR referenced drawing change is supported by the existing design basis and has no impact on equipment performance. There is no change in the operation of the plant.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? There is no effect on the circulating water system performance; this is a drawing change only.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? This USAR referenced drawing change is supported by the existing design basis and has no impact on equipment performance. The operation of the circulating water system is not changed. No equipment important to safety is involved. The drawing revisions involve passive components on a non-safety related system.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? There is no change in the margin of safety or any setpoints defined in any Technical Specification as a result of these drawing changes.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-016	**Description:** Drawings, including P&ID drawings, were revised to show the field configuration for the safety injection and containment spray pump vent valves and piping. The drawings did not agree with field conditions.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? This USAR referenced drawing change is supported by the existing design basis and has no impact on equipment performance. There is no change in the piping configuration or operation of the safety injection system.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? This USAR referenced drawing change is supported by the existing design basis and has no impact on equipment performance. The operation of the safety injection system is not changed.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? This USAR referenced drawing change is supported by the existing design basis and has no impact on equipment performance. The configuration and operation of the safety injection system is not changed.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? There is no change in the margin of safety or any design values defined in any Technical Specification as a result to this drawing change.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-073	**Description:** A pressure reducing valve was added to two of the secondary sampling points. Existing pressure relief valves were bench tested and set to 100 psi. The addition of the pressure reducing valves will assure that the downstream components, having a low design pressure, will not be damaged by inadvertent isolation.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The affected secondary sampling system components are not considered an event initiator or contributor in any accident evaluated in the USAR.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The affected sampling components do not affect the operability or availability of any equipment which is considered safety related or important to nuclear safety.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? The failure of the affected components could not result in a nuclear safety significant event. The affected equipment does not impact any radioactive material handling system nor does it affect any system handling potentially radioactive material.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? Secondary monitoring capability, as required by Technical Specification 5.13, is maintained and the margin of safety is not affected.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-061	**Description:** This field design change request (FDCR) authorized the reconfiguration of the raw water header flow instrument loop 2890. The reconfiguration required the following: • Replacement of magnetic flow element and its flow transmitter with annubar and annubar's transmitter. • Restoration of root valves/isolation valves, power supply, tubing. • Removal of square root extractor and the conduit between the magnetic flow element and its flow transmitter. • Retention of the flow meter. This FDCR also authorized the reconfiguration of the raw water header flow instrument loop 2891. The reconfiguration required the following: • Retention of annubar, annubar's flow transmitter, root valves/isolation valves, power supply, tubing. • Replacement of the flow meter. • Removal of the square root extractor.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The flow element used for FE-2890 will not initiate or increase the probability of an accident previously evaluated accident. Instrumentation separation requirements were adhered to by this modification. The probability of an occurrence of an accident is not made more likely by this activity. This activity will not affect the ability of the raw water system to provide cooling to the CCW system as described in the USAR.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-061	**Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR?** This activity will maintain the pressure rating of the raw water piping. No other mechanical equipment in the raw water system is involved. Instrumentation affected by this modification is not credited in the USAR as equipment important to safety. An annubar will not adversely affect the ability of the raw water system to provide adequate heat removal capacity in the event of an equipment failure. Instrumentation affected by this modification is not credited in the USAR as equipment important to safety.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? The annubar functions as a passive part of the raw water system pressure boundary and does not introduce a new accident mechanism. The annubar does not interact with other systems and does not create the potential for an unanalyzed accident. Panel electrical equipment separation will be maintained and does not create the possibility of an accident of a different type. The new meters are safety related and seismically mounted.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? Installation of an annubar for loop 2890 and the retention of an annubar for loop 2891 does not affect credited equipment operation, it maintains the pressure rating of the raw water system, and it does not affect the raw water system post-accident heat removal capability. The affected electrical equipment is not addressed in Technical Specifications. The margin of safety as defined in Technical Specifications is not reduced.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-050 USAR 98-038	**Description:** Containment isolation actuation system (CIAS) override capability is being provided for containment isolation valves TCV-202 and HCV-204 which are designed to go to the closed position upon receipt of an engineered safeguards signal (ESF). TCV-202 and HCV-204 are associated with the letdown portion of the chemical and volume control system (CVCS) and isolate letdown to limit the radiological consequences of a design basis accident. Override of the CIAS closure signal will be accomplished by means of two new switches, one for each of the two isolation valves, and located on control room panel CB-1,2,3. Timing relays were installed to reinstate CIAS after a fixed amount of time after override is initiated. General purpose relays were installed to configure the letdown valves circuit to provide operations with the flexibility to reinstate CIAS immediately if desired after CIAS override is initiated. Because of the crowded conditions on CB-1,2,3, two existing switches, HC-204-1 and HC-204-2 and their associated indicating lights were combined into a single switch. A spare annunciator above CB-1,2,3 was configured to provide annunciation when either of the two new CIAS override switches is in the override position. This modification was approved by the NRC in Amendment 191.	Section 5.9 Section 5.12 Section 9.2
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The proposed modification does not alter the normal plant configuration in any way that could affect the probability of an accident evaluated in the USAR. When the override switches are in a normal position, all systems will operate as they did prior to the modification. No new modes of normal operation are being introduced which could be considered precursors to an evaluated accident. Operator error could result in release of fission products from the containment to the auxiliary building which were not evaluated in the USAR. This modification has minimized this possibility through design and procedural controls. This modification incorporated the NRC approved USAR evaluation.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-050	**Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR?** Letdown isolation valves are required to be closed in the event of a large break LOCA. This modification creates some non-zero probability that operator error could result in the valves being opened. This non-zero probability is not significant due to all the design and administrative measures taken. However, this was approved by the NRC under Amendment 191. A worst case operator error could result in radiological consequences in the auxiliary building which were not previously analyzed. Specifically, analyses which establish the basis for control room habitability, electrical equipment qualification (EEQ) and auxiliary building access do not contain assumptions which would definitively bound an operator error resulting in the introduction of additional fission products into the auxiliary building. This modification has minimized that possibility by taking appropriate design and administrative measures. However, this was approved by the NRC.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? The proposed modification does not create any failure mode which could impact the operation of the RCS or associated systems in a manner that would create a new or different type of accident. With respect to the letdown isolation function, the plant will operate as it previously has and will respond in exactly the same way, automatically, to an accident signal. No new accidents have been identified. The only new potential for malfunction related to the safety function of the letdown isolation valves is the potential for operator error resulting in the override of CIAS which was previously discussed. No other credible malfunction of equipment affected by this modification would result in a safety function not being met.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-050	**Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?** Technical Specification 2.6 basis section assumes that automatic containment isolation valves are capable of closing in the event of an accident. This modification does not adversely impact that assumption. Administrative and hardware controls will minimize the probability that operator error will result in opening the letdown containment isolation valves when the potential exists for significant fuel damage.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-027	**Description:** Under an emergency temporary modification a cap was installed to isolate the instrument air header pressure sensing line to the pressure switches which control the compressor unloader valves for air compressor CA-1A. The fitting at the header pressure sensing isolation valve CA-274 broke during calibration activities. This cap isolated the break and prevented the loss of air receiver pressure.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? Compressed air (CA) system component failure has no effect on the probability of occurrence or consequences of an accident.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The isolation of the sensing line has the same effect as the tagout of the compressor. It has the same probability as the routine maintenance activities on the CA system.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? CA system component tagout has the same effect as the isolation of the sensing line.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? CA system component tagout has the same effect as the isolation of the sensing line.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-049	**Description:** The refueling boron concentration values, located in COLR Section 11.0, were updated to account for the Cycle 19 loading to guarantee the Technical Specifications refueling boron concentration requirement maintains at least a 5% delta-rho shutdown margin for the Cycle 19 core.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The probability of occurrence of an accident evaluated in the USAR would not be affected with the refueling boron concentration in the SIRWT and SITs. This activity ensures the proper boron concentration will be present for Cycle 19. If the correct boron concentration was not present in the SIRWT and the SITs, the ability to shut down the plant and maintain adequate shutdown margin following an uncontrolled heat extraction or LOCA could be compromised. This activity ensures the properly analyzed boron concentration is present if those accidents mentioned above should occur.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The refueling boron concentration (~1.2 weight %) in the RCS, SIRWT, and SITs does not affect the equipment involved as long as the solubility limit of 23.53 weight % is not exceeded. This value is very small compared to the boron solubility limit. Thus, this activity will not affect the probability of occurrence of a malfunction of equipment. If the correct refueling boron concentration was not present in the SIRWT and the SITs, the ability of the safety injection system to respond to a malfunction of equipment could be compromised. This activity ensures that the proper analyzed boron concentration is present such that the SI system will perform its intended safety function.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-049	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** This activity does not change any equipment or equipment setpoints. Therefore, accidents of a different type cannot occur. This activity cannot create a malfunction of equipment if the solubility of boron in the water is not exceeded. The boron concentration values being changed are consistent with the Cycle 19 core design requirements (per Technical Specifications) and are not near the solubility limit.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? This activity, which changes the COLR refueling boron concentration values, maintains the refueling boron concentration shutdown margin required by Technical Specifications for the Cycle 19 core. Thus, the margin of safety is not reduced as a result of this activity.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-024 USAR 99-032	**Description:** Battery Charger 3 power feed configuration was modified to provide a second power source from bus 1A4 (DG-2) via motor control center MCC-4B1. The modification is utilizing a manual transfer switch to accomplish the change.	Figure 8.1-1
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? Failure modes are unchanged and have been analyzed. The 480 VAC system performance is acceptable. Battery Charger 3 will be a true swing charger for 125 VDC buses 1 and 2 by this modification. The probability of occurrence of an accident is unchanged. The affected 480 VAC MCC functions as originally designed. The circuit protection, provided by this modification, ensures consequences of a 480 VAC system failure remain as previously analyzed. Battery Charger 3 will provide redundancy to either Battery Charger 1 or Battery Charger 2. The 125 VDC system functions as designed.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The probability of a trip device failure is not changed. The trip devices operate in the event of overloads or short circuit. No change in 480 VAC system operation is expected. There is no increase in the probability of equipment malfunction. Battery Charger 3 will provide redundancy to either Battery Charger 1 or Battery Charger 2. The 125 VDC system functions as designed. There is no change in the consequences of any equipment malfunctions.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? The 480 volt system operation and failure modes are unchanged. The modification does not create the possibility of an accident not previously analyzed. The 125 VDC system functions as designed.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-024	**Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?** The 480 volt system operation and failure modes are unchanged. The margin of safety is unchanged. The two independent 480 volt power systems will supply Battery Charger 3 via a transfer switch as designed.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-030 USAR 99-017 DCR 12119	**Description:** The two (2) sustainer sources were removed from the core after Cycle 18 operation. These sustainer sources were original to plant operation and have exceeded the original ten (10) year life. Delayed neutrons from the re-inserted assemblies provide adequate neutrons to the excore detectors. The original engineering assistance request (EAR) and INPO OE9775, dated 3/23/99, both indicate that these sources would create a radiological hazard should they fail and release Antimony to the RCS.	Section 3.7
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? There are no accidents which are initiated by the neutron sources. The mitigation of the Boron Dilution Event is assured by the verification of adequate source strength without the sources in the core. The Boron Dilution Event is mitigated by the operators acknowledging the increasing audible count rate in the control room. Adequate count rate has been verified to exist without the sustainer sources. The Fuel Handling Accident assumes that all of the pins in an assembly rupture at the same time during the accident. The consequences of this accident are due to the release of the fission gasses. Even if the neutron source were to fail, no additional gasses would be released.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? There is no equipment important to safety which is impacted by the removal of the neutron sources.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-030	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** These two sources are stored in the spent fuel pool temporarily and transferred between assemblies during the refueling outage. Permanent storage of the neutron sources in the spent fuel pool does not create the possibility of an accident of a different type. Additionally, the removal of the sources from the core will not initiate any event of any kind associated with the fuel remaining tin the vessel. The sustainer sources do not impact any equipment important to safety when stored in the spent fuel pool. Therefore, permanent storage in the spent fuel pool will not create a malfunction of a different type than previously evaluated in the USAR. Additionally, the removal of the sources from the core will not initiate any malfunction of a different type associated with the fuel or equipment remaining in the vessel.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? There is no reduced margin of safety associated with the storage of the sources in the spent fuel pool. Their removal from the core does not preclude the operators from mitigating the Boron Dilution Event.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-067	**Description:** An isolation valve was installed downstream of valve SI-218. After the valve was installed, an inservice leak check was performed. SI-218 was leaking approximately two drops per minute. This additional insolation valve will eliminate the housekeeping problem associated with contaminated boric acid solution leaking onto the floor inside containment. The leak check was conducted to ensure the leak had been contained.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The material properties of the isolation valve and seismic analysis of the configuration ensure the temporary installation did not increase the probability of an accident previously evaluated in the USAR. There are no USAR analyzed accidents that are directly or indirectly affected by this very minor configuration change. Hence, the consequences of such accidents also remain unaffected.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? This activity does not create or increase the chance of an equipment malfunction. It is only associated with maintaining a secondary pressure boundary which is a passive function. This activity only performs a passive function. If this passive function fails, the leakage past SI-218 resumes. This leakage does not increase the consequences of any equipment malfunction.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-067	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** The only possible failure associated with this activity is the complete failure of the secondary isolation valve which would result in a slight leakage of the safety injection tank (SIT) SI-6A. This event is bounded by current accident analyses. There are no new failure mechanisms being introduced by this activity which could cause an equipment malfunction. Pressure boundary integrity and seismic interactions have already been evaluated.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? The margin of safety is not being reduced by this activity. The design and installation of the component is being done pursuant to system design requirements.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-066 USAR 99-041	**Description:** This modification made three changes to the plant . The first change was to install a signal isolator for the emergency diesel generator DG-2 tachometer circuit Y-6148. The isolator was installed in control room panel AI-133B and will isolate the speed switch from the cable running to the control room. This will protect the speed switch from potential short circuits resulting from a cable spread room/control room fire. The second change added wide range RCS temperature indication for the RCS at alternate shutdown panel AI-185. The modification replaced existing resistance temperature detectors (RTDs) TE-111C/H and TE-121C with dual element RTDs. One element will provide a signal to AI-185 and the other element will be used for the existing Tavg control circuitry. Wide range temperature indication will allow operators to quickly identify a potential rapid cooldown event should this occur due to fire induced spuriously operated equipment. The third change installed the necessary instrumentation to allow for instrument loop B/T-122H to input a signal to loop T-121H. This will allow for bypass of the T-121H thermowell which became obstructed during construction. The changes were made to address concerns identified during the Appendix R self assessment performed in 1997. DG-2 speed switch damage could occur as a result of a short in the cable to the control room indicator. This could prevent the speed switch from operating. The wide range temperature indication will alert operators to a rapid cooldown transient. The possibility of such a transient was identified during a review of spuriously operated equipment resulting from a cable spread room/control room fire. The changes being made to the B/T-122H loop will allow the loop to input into the T-121H loop to bypass the obstructed thermowell.	Section 7.5
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? There are no USAR accidents which occur as the result of a failure in either the T-121H loop and/or the B/T-122H loop. Instrument loop T-121H is not credited for accident mitigation. In addition, USAR Section 14 analysis assumes the functions associated with the reactor regulating system operate in the most adverse way. The function of loop B/T-122H will remain unchanged with respect to the input function to the RPS. CQE isolation will be provided between the two loops.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-066	**Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR?** Modified instrument loop B/T-122H will be isolated from loop T-121H using a CQE isolator. The new modules are CQE and are of the same type as the existing modules used in the loop. Therefore, the probability of malfunction is not increased. The consequences of a malfunction of the B/T-122 loop or the T-121H loop will remain unchanged. A failure of the B/T-122H loop will result in a failure of the T-121H loop. Failure would be alarmed in the control room and the alarm response procedures (ARPs) direct the operator to select the T-111 loop for control should this happen. Also, USAR Section 14 analysis assumes the functions associated with the reactor regulating system operate in the most adverse way. Failure response for B/T-122H is unchanged.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? Failure of either the B/T-122H loop or the T-121H loop will not result in a failure which is more limiting than those previously evaluated in the USAR. A failure of the B/T-122H loop will result in a failure of the T-121H loop. This failure would be alarmed in the control room and ARPs direct the operator to select the T-111 loop for control should this happen. Also, USAR Section 14 analysis assumes the functions associated with the reactor regulating system operate in the most adverse way. Failure response for B/T-122H remains unchanged. Modified instrument loop B/T-122H will be isolated from loop T-121H using a CQE isolator. The new modules are CQE and are of the same type as the existing modules used in the loop. Therefore, the probability of a different type of malfunction is not increased.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? For the B/T-122H instrument loop, the margin of safety is associated with the loop's trip setpoint (i.e. the difference between the trip setpoint and the analytical limit) and the fact that the RPS has four channels associated with that function. This modification did not alter the setpoint or the number of protection channels available and will not impact the margin of safety.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-042	**Description:** The change to alarm response procedure ARP-AI-187/A187 provides direction for actions on loss of HVAC. This revision enhanced steps associated with fan trip or low flow for switchgear ventilation, corrected labels, and provided better guidance.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The uncontrolled (i.e., non-radiologically controlled area) portion of the auxiliary building HVAC system has no effect on the probability of an analyzed accident. The change to ARP-AI-187/A187 provides direction for actions on high temperature in the switchgear room to ensure the operability of the station electrical system. The effect of a high energy line break (HELB) in room 81 due to the use of the original plant HVAC system and associated dampers is mitigated by fire dampers and associated trap door as described in the USAR Appendix M. The consequences of the HELB remain as analyzed.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The change ensures that any malfunction will be within analysis assumptions. The effect of an HELB in room 81 due to the use of the original plant HVAC system and associated dampers is mitigated as described in the USAR Appendix M. This ensures that the probability of equipment malfunction remains unchanged. The change to ARP-AI-187/A187 provides direction for actions on high switchgear room temperature. The change ensures that any malfunction will be within analysis assumptions by providing HVAC to the room. This ensures that the probability of the consequences of equipment malfunction remains unchanged.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-042	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** This change ensures that the electrical system will not malfunction to create the possibility of a different accident by providing an additional option for HVAC to the room. The existing accident analysis bounds the expected events/DBAs. The effect of an HELB in room 81 due to the use of the original plant HVAC system and associated dampers is mitigated as described in the USAR Appendix M. This ensures that no new accidents have been created. The possibility of a malfunction of a different type is not created.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? The effect of an HELB in room 81 due to the use of the original plant HVAC system and associated dampers is mitigated as described in the USAR Appendix M. The changes ensures that the margin of safety with respect to the electrical distribution system remains unchanged.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-032 USAR 99-039	**Description:** The actuator/air intensifier pump for safety injection system valve HCV-2987 was modified to: Eliminate the existing high pressure air intensifier. Replace the existing high pressure actuator with a standard model that operates using normal air pressure. Replace the existing high pressure solenoid with a standard model that operates using normal air pressure. Install a new air accumulator. USAR Figure 5.9-13, sheet 5, and Table 9.12-1 incorrectly listed HCV-2987 as a fail open valve. Since the existing actuator and its replacement do not have a spring the valve will fail as is on loss of air. This discrepancy was documented in condition report 199900527.	Figure 5.9-13 Sheet 5 Section 8.5 Table 9.12-1
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The existing valve actuator and associated components were replaced with components that meet the performance requirements for the valve to function during accident conditions. HCV-2987 is not associated with an initiating mechanism for any USAR related accident. This modification did not change the functional requirement for HCV-2987 to close once for simultaneous hot and cold leg injection. The purpose of this modification was to increase the reliability of HCV-2987 so that it is capable of functioning as designed. This is considered to be an enhancement to nuclear safety.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? This modification simplified the pneumatic arrangement for HCV-2987, thereby increasing the reliability of the valve and decreasing the likelihood that it would fail to close for long term core cooling initiation. This modification did not change the failure modes for HCV-2987 or its ability to function during an accident. Therefore, the consequences of malfunction of equipment important to safety previously evaluated in the USAR is not increased.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-032	**Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR?** No pressure boundary changes were made to HCV-2987 under this modification. The new valve actuator will be capable of meeting the performance requirements necessary for the valve to function under accident conditions. The operation and function of the valve will not be changed. There is no increased possibility of an accident of a different type than any previously evaluated in the USAR. The valve will remain locked open during normal operation and will only be closed to initiate simultaneous hot and cold leg injection. It's failure mode will be "As-is" on loss of air. This is the preferred failure mode since the valve is required to remain closed indefinitely for long term core cooling. A fail-open valve would reopen when all air is bled from the accumulator and terminate hot leg injection. This activity will not increase the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? This modification will not reduce the margin of safety for those components required for emergency core cooling system as defined in the basis section of Technical Specification 2.3.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-035 USAR 99-019	**Description:** USAR Appendix F, Section F.2.2.2 was revised to add an alternative method for seismic qualification of mechanical and electrical equipment, electrical relays, and cable and conduit raceway systems. The method is based on generic implementation procedure (GIP)-3, as supplemented by Seismic Qualification Utility Group (SQUG) safety evaluation report SSER No. 2 and SSER No. 3, which was used to resolve unresolved safety issue (USI) A-46 by showing that equipment essential to a safe shutdown is capable of functioning during and following a Maximum Hypothetical Earthquake. Sections F.2.1.2, F.2.1.3, F.2.2, F.2.5, and F.3 of USAR Appendix F are also revised for consistency. This method was approved by the NRC for FCS in "Safety Evaluation Report Closing Out USI A-46 for Fort Calhoun Station," dated July 30, 1998.	Appendix F

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-035	**Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR?** The response to this question addresses the impact of the proposed change on a seismic event both as a potential accident initiator and as an occurrence considered in equipment design. The proposed change involves allowance of GIP-3 method as an alternative method for demonstration of seismic adequacy of equipment relative to the current Fort Calhoun Station licensing basis. The GIP-3 method provides an equivalent or superior level of assurance that equipment will perform required safety functions during and following a seismic event. As such, the proposed change will have no impact on a seismic event as a potential initiator of accidents previously analyzed in the USAR. The only accidents in the USAR that could potentially be affected by the use of the GIP-3 method are the Design Earthquake and the Maximum Hypothetical Earthquake. However, the GIP-3 method, being a method for demonstrating seismic adequacy of equipment, will not increase the likelihood of the occurrence of an Design Earthquake or the Maximum Hypothetical Earthquake. Therefore, with respect to the seismic event as an occurrence considered in equipment design, the proposed change will not increase the probability of occurrence of a seismic event because this event is the result of natural phenomena. The only accidents in the USAR that could potentially have radiological release consequences affected by the use of the GIP-3 method are those accidents analyzed in the USAR associated with the Design Earthquake and the Maximum Hypothetical Earthquake. However, the proposed change will have no effect on, and will not change any accident consequences as describe above. Use of a new method for demonstrating equipment seismic adequacy could potentially affect the ability of safety-related equipment or equipment important to safety to perform required safety functions during or after a seismic event, thus affecting radiological release consequences. However, the use of the GIP-3 method will provide equivalent or superior assurance of equipment seismic adequacy to that provided by the current Fort Calhoun Station licensing basis. Thus, use of the GIP-3 method will have no affect on radiological release consequences.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 99-035	**Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR?** The proposed change involves allowance of the GIP-3 method as an alternative method for demonstration of seismic adequacy of equipment. Relative to the current Fort Calhoun Station licensing basis, the GIP-3 method provides an equivalent or superior level of assurance that equipment will perform required safety functions during and following a seismic event. As such, there is no decrease in the seismic adequacy of equipment.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? Since the proposed change does not adversely affect the set of equipment which must meet seismic adequacy requirements or the level of seismic adequacy as defined in the USAR, the proposed change does not create the possibility of an accident of a different type than previously evaluated in the USAR.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? Detailed comparison of the Fort Calhoun Station licensing bases to the GIP-3 method in order to assess the relative margin of safety was provided. The GIP-3 method will not result in a reduction of safety margin. Differences between the GIP-3 method and the Fort Calhoun Station licensing basis were identified and the effect of the differences on the overall cumulative relative safety margin was determined. The results demonstrated that the use of GIP-3 method will not reduce the plant margin of safety.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 98-072	**Description:** This minor configuration change installed a ball valve between CA-123 and PA-1701. The GEI-35 evaluation has shown that the ball-valve is satisfactory for the application. The installation took place during maintenance of an upstream isolation valve, CA-123, which was performed under the scope of a standing order G-87 Action Plan.	None
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? There are no evaluated accidents which would be more probable because of this new valve. The affected equipment is not required to mitigate any accident nor prevent an accident from occurring.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? There is no equipment important to safety which would be more likely to experience a malfunction due to the addition of this new valve. The new valve configuration is physically separated from safety related equipment, and in a non safety related system and cannot increase the probability of occurrence of safety related equipment malfunctions. There is no equipment important to safety which would suffer worse consequences during a malfunction due to the addition of this new valve. The new valve configuration, being located in a non-safety related system, cannot increase the consequences of a malfunction of safety related equipment.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? No new possible accidents will be created due to the addition of this new valve, which is non-CQE, non-safety related, and not associated with equipment important to safety.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
USQD 98-072	**Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?** There are no technical specifications associated with the compressed air system. Consequently, no margins of safety as defined in the basis for any Technical Specifications are reduced as a result of the addition of this new compressed air valve.	

CHANGES, TESTS AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR COMMISSION APPROVAL

Source	Description	USAR Section(s) Revised
DCR 12105 MR-FC-89-017	**Description:** Modification MR-FC-89-017 abandoned parts of the radioactive waste treatment system that were not used and were not planned to be used during the life of the plant. This modification reduced the probability of contamination by decreasing waste inventory. This modification also reduces the amount of time it takes to find sources of airborne contamination leaks.	Section 1.2 Section 7.6 Section 11.1
	Could the proposed activity increase the probability or consequences of an accident previously evaluated in the USAR? The design, material, and construction standards applicable to the systems that were modified are unchanged. Performance for all associated systems was unchanged. There are no changes to the performance of barriers to the release of radioactive materials, fuel integrity, or containment integrity. The system pressure retaining capabilities are retained.	
	Could the proposed activity increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR? The probability of occurrence is decreased by eliminating current system interfaces. Seismic integrity is maintained. The function of the affected systems was unchanged. The consequences of a malfunction are unaffected by removing inactive components from the active ones.	
	Could the proposed activity create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR? The function and operation of the affected systems was unchanged. Unused equipment was removed or abandoned in place.	
	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification? All system design parameters are maintained. No parameter used to establish any margin of safety was affected.	

USAR CHANGES OTHER THAN THOSE RESULTING FROM 10 CFR 50.59 AND QUALITY ASSURANCE PLAN CHANGES IN ACCORDANCE WITH 10 CFR 50.54

Source	Description	USAR Section(s) Revised
USAR 99-046	Deleted a nebulous sentence from USAR Section 4.3. The sentence was related to the function of the reactor coolant pump antireverse rotation device and it provided no meaningful information.	Section 4.3
USAR 99-044	The words "normally de-energized but are" were removed from the description of how the pressurizer backup heaters operate. The old description was misleading about how the station operates with the pressurizer backup heaters on.	Section 4.3
USAR 99-047	Deleted the statement that CEA manipulation can be performed within the reactor vessel if an extension piece is installed on the portable handling tool. Although this operation can be performed, it can not be performed using the fuel handling machine. It would have to be performed using the polar crane, which is not indexed for core locations.	Section 9.5
USAR 99-043	Revised USAR Section 9.5 to: 1) Clarify the new fuel storage rack is positioned by its location on an elevated balcony to preclude the possibility of a flooded condition. 2) Clarify that one cell of the new fuel storage rack is prohibited from use as a storage cell. 3) Eliminate a duplicated reference to engineering analysis EA-FC-94-029 and corrects the format of the remaining reference.	Section 9.5
USAR 99-29	Revised USAR Section 12 to reflect organizational changes.	Section 12.1 Figure 12.1-1 Figure 12.1-2 Figure 12.1-3 Figure 12.1-4
USAR 99-25	Revised USAR Section 5.4.4 to include discussion of why 100 second long temperature excursion above the accident temperature of 305 degrees does not adversely affect the thermal loading inputs of the containment structure analysis.	Section 5.4
USAR 99-48	Revised USAR Section 9.8 to remove misleading information concerning the actual operating configuration of the raw water system.	Section 9.8
DCR 11956	Revised USAR Sections 10.2 and 11.1 to clarify that there are two sources of water for screen wash pumps and to clarify that the 305,000 gallons per minute average annual flow in the discharge tunnel is assumed for the purpose of calculating effluent concentrations and is not intended to be an absolute flow value.	Section 10.2 Section 11.1
	Figures 5.9-14 and 5.9-15 were re-issued.	Figure 5.9-14 Figure 5.9-15
	Minor revisions were made to Figures 4.3-9, 7.4-1 and 8.5-9.	Figure 4.3-9 Figure 7.4-1 Figure 8.5-9

USAR CHANGES OTHER THAN THOSE RESULTING FROM 10 CFR 50.59 AND QUALITY ASSURANCE PLAN CHANGES IN ACCORDANCE WITH 10 CFR 50.54

Source	Description	USAR Section(s) Revised
DCR 11953	Revised the USAR due to the use of the term "flux tilt" which is not consistent with what is being described in present terminology, as demonstrated by the listed equation. A typographical error was corrected in Table 1.4-1.	Section 1.4 Section 3.4 Section 3.7 Section 9.5
USAR 99-023	Changed the QA program to reflect the most recent organization of OPPD's Production Operations Division.	Appendix A, QA Program

REGULATORY COMMITMENTS REVISED REQUIRING NRC NOTIFICATION IN ACCORDANCE WITH NEI 99-04

SOURCE	DESCRIPTION
AR-1032 / LIC-79-0121, Response to Information Bulletin 79-19	The response to NRC Information Bulletin IB 79-19 stated that OPPD would perform an audit of transport packaging and transport activities on an annual basis. This commitment is being revised to conform with the normal Quality Assurance Program (QAP) cycle of performing audits on these types of activities. At the time of this commitment no other audits or surveillances were in place. OPPD now uses a four tier, three year audit program as described in the QAP.
AR-4358 / LIC-79-0128, Description of FCS Alternate Shutdown Capability	In the description of the FCS Alternate Shutdown Capability, a commitment was made, or could be implied, that a surveillance test would be performed on a "refueling outage" (RFO) interval for the DG-2 "183/MES" switch. This interval is being revised to every third RFO. The 183/MES switch has performed perfectly since installation, there is no indication of degradation, and there is no reason to believe that it will degrade during the extended testing interval. If any sign of degradation is indicated, the interval will be decreased.

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Fort Calhoun Station
May 2000, USAR Update
Replacement Instructions

Remove	Insert
USAR Revision Status	USAR Revision Status (05/10/00)
Section 1, Table of Contents	Section 1, Table of Contents (4/26/00)
Section 1.2	Section 1.2 Rev 5
Section 1.4	Section 1.4 Rev 1
Section 2, Table of Contents	Section 2, Table of Contents (5/8/00)
Section 2.10	Section 2.10 Rev 3
Section 3, Table of Contents	Section 3, Table of Contents (4/26/00)
Section 3.2	Section 3.2 Rev 1
Section 3.4	Section 3.4 Rev 9
Section 3.6	Section 3.6 Rev 3
Section 3.7	Section 3.7 Rev 5
Section 3.9	Section 3.9 Rev 3
Section 4, Table of Contents	Section 4, Table of Contents (1/4/00)
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Section 5, Table of Contents	Section 5, Table of Contents (3/2/00)
Section 5.4	Section 5.4 Rev 1
Section 5.9	Section 5.9 Rev 6
Section 5.12	Section 5.12 Rev 2
Section 6, Table of Contents	Section 6, Table of Contents (12/2/99)
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Section 7, Table of Contents	Section 7, Table of Contents (5/8/00)
Section 7.5	Section 7.5 Rev 6
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Section 8, Table of Contents	Section 8, Table of Contents (2/1/00)
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Section 9.2	Section 9.2 Rev 7
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Section 9.8	Section 9.8 Rev 7
Section 9.10	Section 9.10 Rev 6
Section 9.12	Section 9.12 Rev 4

Fort Calhoun Station
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Section 10, Table of Contents	Section 10, Table of Contents (3/15/00)
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Section 11, Table of Contents	Section 11, Table of Contents (4/18/00)
Section 11.1	Section 11.1 Rev 5
Section 12, Table of Contents	Section 12, Table of Contents (10/28/99)
Section 12.1	Section 12.1 Rev 3
Section 14, Table of Contents	Section 14, Table of Contents (5/10/00)
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Appendix A	Appendix A Rev 4
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Figure 2.10-2	Figure 2.10-2 Rev 4
Figure 3.4-1	Figure 3.4-1 Rev 5
Figure 3.4-2	Figure 3.4-2 Rev 5
Figure 3.4-4	Figure 3.4-4 Rev 5
Figure 3.4-5	Figure 3.4-5 Rev 5
Figure 3.4-6	Figure 3.4-6 Rev 5
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Figure 5.9-13 Sheet 5	Figure 5.9-13 Sheet 5, Rev 6
Figure 5.9-13 Sheet 62	Figure 5.9-13 Sheet 62, Rev 8
Figure 5.9-13 Sheet 63	Figure 5.9-13 Sheet 63, Rev 8
Figure 5.9-14	Figure 5.9-14 Rev 14
Figure 5.9-15	Figure 5.9-15 Rev 16
Figure 5.9-20	DELETED
Figure 5.9-21	DELETED

Remove	**Insert**
Figure 7.4-1	Figure 7.4-1 Rev 8
Figure 8.1-1	Figure 8.1-1 Rev 10
Figure 8.2-1	Figure 8.2-1 Rev 5
Figure 8.2-2	Figure 8.2-2 Rev 7
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Figure 12.1-1	Figure 12.1-1 Rev 13
Figure 12.1-2	Figure 12.1-2 Rev 12
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Figure 12.1-5	DELETED
Figure 14.12-1	Figure 14.12-1 Rev 1
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Figure 14.12-10	Figure 14.12-10 Rev 1
Figure 14.12-11	Figure 14.12-11 Rev 1
Figure 14.12-12	Figure 14.12-12 Rev 1
Figure 14.12-13	Figure 14.12-13 Rev 1
Figure 14.12-14 through 14.12-32	DELETED
Figure M-8	Figure M-8 Rev 1
Figure M-10	Figure M-10 Rev 1

Document Number	Title	Revision/Date
USAR-1	Introduction and Summary	
	❖ **Section 1.1** - Introduction	R0 05/23/97
	❖ **Section 1.2** - Summary Plant Description	R5 04/26/00
	❖ **Section 1.3** - Principal Architectural and Engineering Criteria	R1 09/24/98
	❖ **Section 1.4** - Research and Development Requirements	R1 03/28/00
	❖ **Section 1.5** - Items of Special Interest to ACRs	R1 05/23/97
	❖ **Section 1.6** - Significant Design changes since Award of Construction Permit	R1 10/14/98
	❖ **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
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	❖ **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	R2 02/02/98
	❖ **Section 2.6** - Geology	R0 05/23/97
	❖ **Section 2.7** - Hydrology	R0 05/23/97
	❖ **Section 2.8** - Demography	R1 07/11/97
	❖ **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	R1 09/24/98
	❖ **Section 3.2** - Design Bases	R1 11/24/99
	❖ **Section 3.3** - Reactor and Core Fuel Cycle Considerations	R1 09/24/98
	❖ **Section 3.4** - Nuclear Design and Evaluation	R9 03/28/00
	❖ **Section 3.5** - Cycle 6 Core Power Uprate to 1500 MW_T	R0 05/23/97
	❖ **Section 3.6** - Thermal and Hydraulic Design Evaluation for Limiting Transients and Setpoint Analysis for Cycle 17	R3 11/24/99
	❖ **Section 3.7** - Mechanical Design and Evaluation	R5 04/28/00
	❖ **Section 3.8** - Fuel Performance During Anticipated Transients after Long Burnup	R0 05/23/97
	❖ **Section 3.9** - Specific References	R3 11/24/99
	❖ **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 R8 01/05/00
- **Section 4.4** - System Design Evaluation R1 05/23/97
- **Section 4.5** - Tests and Inspections R0 05/23/97
- **Section 4.6** - Specific References R1 05/23/97
- **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 R1 08/12/99
- **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 R1 02/02/98
- **Section 6.2** - Safety Injection System R10 12/01/99
- **Section 6.3** - Containment Spray System R6 12/20/99
- **Section 6.4** - Containment Air Recirculation, Cooling and Iodine Removal System R1 09/24/97
- **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 R1 05/23/97
- **Section 7.3** - Engineered Safeguards Controls and Instrumentation R2 09/24/98
- **Section 7.4** - Regulating Systems R0 05/23/97
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- **Section 7.7** - General References R2 09/24/98

USAR-8 Electrical Systems

- **Section 8.1** - Introduction R0 05/23/97
- **Section 8.2** - Network Interconnections R4 02/01/00
- **Section 8.3** - Station Distribution R2 07/27/98
- **Section 8.4** - Emergency Power Sources R3 09/15/98
- **Section 8.5** - Initial Cable Installation Design Criteria R1 06/30/99
- **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 R7 03/07/00
- **Section 9.3** - Shutdown Cooling System R2 05/14/99
- **Section 9.4** - Auxiliary Feedwater System R7 11/18/98
- **Section 9.5** - Refueling System R8 03/28/00
- **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 R7 03/28/00
- **Section 9.9** - Turbine Plant Cooling Water System R0 05/23/97
- **Section 9.10** - Heating, Ventilating and Air Conditioning Systems R6 11/18/99
- **Section 9.11** - Fire Protection System R6 05/14/99
- **Section 9.12** - Compressed Air System R4 11/18/99
- **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 R5 04/26/00
- **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 R3 11/24/99
- **Section 14.2** - Control Element Assembly Withdrawal Incident R3 11/24/99
- **Section 14.3** - Boron Dilution Incident R2 09/24/98
- **Section 14.4** - Control Element Assembly Drop Incident R3 11/24/99
- **Section 14.5** - Malpositioning of the Non-Trippable CEAs R0 05/23/97
- **Section 14.6** - Loss of Coolant Flow Incident R3 11/24/99
- **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 R2 11/24/99
- **Section 14.10** - Malfunctions of the Feedwater System R2 10/06/98
- **Section 14.11** - Excess Load Increase R3 11/24/99
- **Section 14.12** - Main Steam Line Break Accident R4 11/24/99
- **Section 14.13** - Control Element Assembly Ejection Accident R1 11/24/99
- **Section 14.14** - Status of Steam Generator Tube Rupture Incident R3 10/22/98
- **Section 14.15** - Loss of Coolant Accident R6 11/24/99
- **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 R3 11/24/99
- **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

USAR

Appendix A - Quality Assurance Program R4 08/12/99
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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 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 power 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 recirculation, cooling and iodine removal 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 recirculation, cooling and iodine removal 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 particulate filters and with impregnated activated charcoal adsorbers for iodine removal. The containment air recirculation, cooling and iodine 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 normally monitor each of the selected plant parameters. The reactor protective system logic is designed to initiate protective action whenever the signal of any two of four channels 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 protective action when it is required.

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. The boron concentration in the reactor coolant water is also monitored and concentration information is presented in the control room.

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 supplied by a 125-Volt d-c/120-V a-c inverter.

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. Spent fuel assemblies are stored in vertical racks spaced to preclude criticality with no credit taken for the boron dissolved in the pool water. Cooling and purification equipment is provided for the fuel pool water.

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 recirculation cooling and iodine removal 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

1.4 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, % Δρ	Moderator Temperature Coefficient 10^{-4} Δρ/°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 Siemens Power Corporation, 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[4]. 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.[5] This correlation has been used in conjunction with D-TORC and the CETOP thermal-hydraulics computer codes.

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 Sytem" (FACTS) and later at the Science and Technology Center in Pittsburg, PA.

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.

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2.10 ENVIRONMENTAL RADIATION MONITORING

2.10.1 General

The environmental monitoring program is designed to provide data concerning the types and amount of radioactivity present in the environment of the Fort Calhoun Station. The preoperational program was designed to assess environmental conditions before the arrival of fuel. Subsequent analysis during the operational program is being used to demonstrate that plant operations do not have a significant effect on the environment.

2.10.2 Preoperational Survey Program

The purpose of the preoperational survey program was to determine the base level of existing radioactivity to which future analytical results can be compared; the program extended for four consecutive years. The monitoring program was developed in cooperation with the regulatory agencies of Nebraska and Iowa and the Fish and Wildlife Service of the United States Government Department of the Interior.

Specific radionuclide and/or gross radioactivity analyses were performed on the selected samples. Table 2.10-1 summarizes the types of samples and analyses included in the preoperational program.

Table 2.10-1 - "Gross and Specific Radionuclide Analyses"

	Gross α	Gross β-γ	γ-Spec	Sr-90	H-3	K-40	I-131	Cs-137
Surface Water	X	X	X	X	X			
Well Water	X	X	X	X	X			
Mud and Silt	X	X	X					
Aquatic Biota		X	X	X		X		
Milk		X	X	X		X	X	X
Vegetation	X	X	X	X	X	X		
Air Particulate	X	X	X					
Wildlife				X			X	

2.10.3 Preoperation Survey Results

2.10.3.1 Trial Monitoring Period

The first nine months of the program, starting in September 1968, was a trial period designed to verify the availability of adequate sample types and to select and test analytical procedures.

Results obtained during the trial period were preliminary. The trial period results are included in this report because they describe the background conditions and illustrated the preoperational surveillance program. No significant peaks were evident in any of the gamma scans performed on samples.

Water

Surface water samples were collected at six stations: one at the Desoto National Wildlife Refuge Lake area and five from the Missouri River at sampling stations located above and below the plant site, including the municipal water supplies at Omaha, Nebraska, and Council Bluffs, Iowa.

Well waters were sampled at eleven wells within a four-mile radius of the plant. Table 2.10-2 is a summary of the surface and well water data.

Table 2.10-2 - "Average Radioactivity of Well and Surface Waters November 1968 - June 1969"

	Activity Concentration, pc/liter	
	Well Water (11 Samples)	Surface Water (6 Samples)
Alpha	0.0	0.7
Beta-Gamma	10.9	26.2
Strontium 90	0.1	1.3
Tritium	550	1000

Mud and Silt

Mud and silt samples were taken from the Missouri River downstream of the plant. No alpha radiation was detectable; the analysis for beta-gamma gross activity showed 18 picocuries per kilogram for the mud and silt.

Aquatic Biota

The basis for sampling aquatic biota was formulated from specific recommendations of the Nebraska Game, Forestation and Parks Commission. The fish species selected were chosen because their food habits include organisms within many of the lower trophic levels and because they are important from the standpoint of sport and commercial fishing.

The food habits and radioactivity of the fish samples, which were taken from the Missouri River, are shown in Table 2.10-3.

Table 2.10-3 - "Food Habits and Radioactivity of Missouri River Fish October 1968 - June 1969"

Specie	Food Habits	(β-γ)-(K-40) nc/kg	K-40 nc/kg	Sr-90 pc/kg
Flathead Catfish #	Fish	3.2	2.6	0.0
Flathead Catfish *	Insects	7.8	10.6	0.0
Channel Catfish #	Fish	3.2	6.7	100.00
Channel Catfish *	Insects	1.6	6.5	0.0
Carp	Omnivorous	8.5	8.4	24.0
Paddlefish	Plankton	---	---	---
Buffalo	Algae & Insects	4.6	9.5	0.0
Shad	Plankton	---	---	---

Greater than 10 inches long
* Less than 10 inches long

The paddlefish is difficult to collect but was included where possible because it feeds exclusively on plankton; the shad and buffalo with food habits similar to the paddlefish are acceptable substitutes. During its lifetime, the flathead catfish remains within approximately one mile of its origin and is therefore, sampled downstream of the plant site. Catfish and carp are the most abundant of the commercial fish varieties.

The Missouri River has a sand bottom which moves with the water flow; therefore, benthos and other bottom organisms are extremely scarce. Joint efforts with the Nebraska Game Commission to obtain sufficient samples for analysis of periphyton have failed; a cooperative study continues as a separate project.

Milk

Milk from large Grade A milk producers in the local milkshed was sampled in cooperation with the Omaha Douglas County Health Department. The dairy herds of these Grade A milk producers are located downwind of the plant site. Radioactivity levels in the milk samples analyzed are shown in Table 2.10-4.

Table 2.10-4 - "Radioactivity in Milk January - March, 1969"

	Fresh Milk		Preserved Milk		
	I-131 pc/1	Cs-137 pc/1	(β-γ)-(K-40) nc/1	K-40 nc/1	Sr-90 pc/gm Calcium
Farm A	0	0	0.53	0.73	1.0
Farm B	0	0	0.81	0.74	1.0
Farm C	0	0	0.71	0.78	0.9

Vegetation

Foods normally consumed by the general population constitute the vegetation samples. Six stations with a total of ten varieties of food were sampled during the 1968 growing season. The variation in analytical results is shown in Table 2.10-5.

Table 2.10-5 - "Radioactivity in Vegetation October, 1968"

	Maximum nc/kg	Minimum nc/kg
Alpha	0.0	0.0
Beta-Gamma minus K-40	14.0	0.3
K-40	39.2	3.2
Sr-90	0.143	0.000
H-3	6	0

Air Particulate

Airborne particulate matter was collected at the plant site on 0.45 micron pore size filters; the filter was removed from the sampler and counted after the radioactivity had decayed for at least seventy-two hours. The air volume passed through the filter was approximately 1,000 cubic feet. None of the 32 samples analyzed showed any indication of alpha activity; the average beta-gamma concentration was 0.26 pc/m^3 with a maximum of 0.78 pc/m^3 and a minimum of 0.08 pc/m^3.

Background radiation readings measured with a Geiger-Mueller survey meter at sixteen stations around the plant site were all in the 0.00-0.02 mr/hr range. Results of the combination film badge-thermoluminescent dosimeters, at eleven stations, were all less than 30 mrem per quarter.

Wildlife

A wild rabbit sample was included to represent wildlife normally consumed in the area. These rabbits are free to wander, but they normally remain in the immediate vicinity. The radioactive content was 20 picocuries of Strontium-90 per gram of calcium in the femur and no iodine-131 was detectable in the thyroid.

2.10.3.2 Preoperational Monitoring Period

Following the trial period, the formal preoperational surveillance monitoring program was started in July, 1969, and continued for three years. This formal preoperational survey was an intensified continuation of the trial period already discussed. The program included soil samples and vegetation which are stored for possible future analyses.

The preoperational program results were documented for future reference and comparison; they defined the pre-operational background levels. Future background conditions may vary due to influences such as fallout from nuclear testing; however, the continuing environmental survey programs will provide adequate data to document changes in the background conditions.

2.10.3.3 Operational Survey Program

The purpose of the operational survey program is to provide public assurance that the Fort Calhoun contribution to naturally existing radioactivity is negligible. The program verifies the effectiveness of the waste disposal systems and radiological safety procedures incorporated in the plant.

Since plant operations began, samples similar to those taken during the preoperational program have continued to be collected routinely. The samples which would show changes in radioactivity first, primarily water and air, are sampled most frequently. Table 2.10-6 shows the types of samples taken, the frequency and number of these samples and the analysis frequency. Figures 2.10-1 thru 2.10-3 show the sampling locations and Figure 2.10-4 shows the types of sampling to be done at each location.

Deviations from the monitoring program may occur concerning sample location sites. If samples are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of equipment or participants ceasing participation in the program, corrective actions will be taken. Alternate samples and/or alternate sample locations will become part of the program as soon as practical in order to satisfy Offsite Dose Calculation Manual (ODCM) requirements. Figures 2.10-1 thru 2.10-4 and Table 2.10-6 will be updated when required to reflect changes in sample locations.

Table 2.10-6 - "Sample Types and Frequency "

Sample Class	Collection Frequency	Analysis Frequency	No. of Samples
Direct Radiation TLD	Quarterly	Quarterly	11
Emergency TLD	Replaced Annually	Site Area and General Emergencies Only	32
Air Monitoring	Weekly	Weekly (Gross Beta and I-131)Quarterly Composite of weekly filters (Gamma Isotopic)	5
Water	Weekly	Monthly (Gamma Isotopic) Analysis) Quarterly (H-3 Analysis)	3
Milk	Semi-monthly during grazing season. (May to October)	Semi-monthly during grazing season. (May to October)	2
Fish	Once per season (May to October)	Once per season (May to October)	5
Sediment	Semi-annual	Semi-annual	1

A land use survey is required to be conducted once every 24 months in order to identify changes in the use of the unrestricted area. As a result of this census, locations other than those presently sampled may be identified as potential higher exposure pathways and will be added to the environmental program. The sample numbers and location may vary from those presented in Table 2.10-6. In addition, other sample classes such as well water and vegetation may be added to the program. However, Table 2.10-6 represents the minimal operating requirements of the sample program.

Table 2.10-7 - "Radiological Environmental Sampling Locations and Media"

Location Number	Location Description	Distance from FCS Reactor Bldg (miles)	Direction (Degrees from north)	Airborne Particulate	Airborne Iodine	TLD	Surface Water	Fresh Milk	Bottom Sediment	Fish	Vegetation
1	Onsite Station No. 1, 110-meter weather tower	0.5	293°	X	X	X					
2	Onsite Station No. 2, adjacent to old plant access road	0.6	208°	X	X	X					
3	Offsite Station No. 3, intersection of Hwy. 75 and farm access road	0.8	145°	X	X	X					
4	Blair OPPD Office	3.0	303°	X	X	X					
5	EOF Building, North Omaha Power Station	17.5	157°	X	X	X					
6	Fort Calhoun City Hall	4.8	149°			X					
7	Fence around intake gate, Desoto Wildlife Refuge	2.0	101°			X					
8	Entrance to Plant Site from Hwy. 75	0.6	180°			X					
9	NW of Plant	1.0	310°			X					
10	WSW of Plant	0.7	250°			X					
11	SE of Plant	0.9	130°			X					
12	Met. Utilities Dist., Florence Treatment Plant North Omaha, NE	17.0	156°				X				
13	West bank Missouri River, downstream from reactor building	0.5	106°				X		X		
14	125' upstream from intake bldg., west bank of river	0.1	345°				X		X(1)		
15	Smith Farm(1)	1.9	133°				X				
16	OPPD Onsite Well(1)	0.1	154°				X				
17	Headquarters Bldg (1) Desoto Wildlife Refuge	3.1	53°				X				

Table 2.10-7 (Continued)

Location Number	Location Description	Distance from FCS Reactor Bldg (miles)	Direction (Degrees from north)	Airborne Particulate	Airborne Iodine	TLD	Well Water	Fresh Milk	Bottom Sediment	Fish	Vegetation
18	Miller Farm[4]	0.8	206°					X			X
19	Flynn Dairy[4]	3.4	310°					X			
20	Mohr Dairy	9.8	187°					x(2)			X
21	Japp Dairy[2]	6.3	219°					X			
22	Fish Sampling Area - Missouri River	R.M. 645.0	-							X	
23	Fish Sampling Area - [5] Missouri River	R.M. 666.0	-							X	
24	Legenhausen Farm[4]	0.7	207°								X
25	Seltz Farm[2]	2.7	168°					X			
26	John Welchert Farm	2.7	138°								X
27	Jerry Welchert Farm	2.0	296°								X
28	Alvin Pechnik Farm[4]	0.9	164°								X
29	E. Ellis Farm	0.7	180°								X[1]
30	Axtell Acreage	0.7	207°								X
31	Hakanson Farm	1.1	205°								X
32	Valley Substation #902	19.5	219°	X	X	X					
33	Bansen Farm[4]	0.7	207°								X
34	W.Jones Farm[6]	0.89	165°								X[3]
35	Onsite Farm Field[1]	0.74	110°								x

NOTES:
(1) Sampling not required for pathway modeling, collections performed for additional information only.
(2) When a milk sample is not available at a location, a broad leaf vegetation sample will be collected at that location as a substitute.
(3) Vegetable/food products sites chosen based on Land Use Survey and calculated doses.
(4) Location currently discontinued. Documented in table for historical reference only.
(5) Location is 21.0 river miels north of plant. Exact location can not be illustrated on map.
(6) Residence at 1.1 mile/155°, Actual garden on farm property at 0.89 miles/165°

Direct Radiation (TLD)

Quarterly TLDs are analyzed for ambient gamma after each respective replacement. Emergency TLDs are replaced annually or would be collected after a Site Area or General Emergency.

Air Monitoring

Air particulate samples are analyzed by gross beta count weekly. A gamma spectral analysis is performed on a quarterly composite of the weekly samples. Iodine cartridges at the air particulate stations are analyzed for I-131 on a weekly basis.

Water

A gamma scan is performed on monthly surface water composites. In addition, quarterly composites of surface water samples undergo analysis for tritium.

Environmental sample analyses are performed to provide compliance with 10 CFR 20 and to differentiate plant releases from natural or other sources of environmental radiation. Local public regulatory agencies who have assisted in the development of the environmental surveillance program are informed of survey results.

During plant operation, waste discharges are analyzed prior to release and are continuously monitored during release. The amount of radioactivity released is documented as a standard plant operating procedure. The environmental surveillance program is an independent survey verifying that the operating procedure for waste releases is effective and plant operations do not have a significant effect on the environment. In the unlikely event of an accidental release, samples will be collected and analyzed at all applicable environmental stations. Additional samples may also be obtained to better evaluate the magnitude of such a release.

Milk

During the time the cows are on pasture (May to October) samples of milk are collected semi-monthly within a five mile radius of the plant and analyzed for radioiodine content, calculated as I-131. Analyses are accomplished within eight days of sampling. The milk samples are also analyzed by gamma scan.

Fish

Samples of fish are collected once per season (May to October). *The fish selected for analysis are:

Category	Species	Size	Basic Food Habits
1	Flathead Catfish (juvenile)	<10" length	Carnivorous (insects)
	or Goldeye (adult)	N/A	
	or Buffalo (adult)	N/A	
	or Freshwater Drum (adult)	N/A	
2	Flathead Catfish (adult)	>10" length	Carnivorous (fish)
3	Carp (adult)	N/A	Omnivorous (insects, plants, fish)
4	Gizzard Shad (adult)	N/A	Planktonic/ Carnivorous (drift, single cell organisms, insects)

*Species obtained may vary if desired fish are unavailable or are considered a rare or protected species.
A Gamma scan analysis is performed on fish samples.

Sediment

Gamma scan analysis is performed on these samples semi-annually.

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3.2 DESIGN BASES

3.2.1 Performance Objectives

The initial full-power thermal rating of the core was 1420 MWt, which corresponds to a gross electrical output of 481 MWe. Although the plant was designed for a full-power rating of 1500 MWt, the initial license application and the first five fuel cycles of operation were at this lower power rating of 1420 MWt. On August 15, 1980, Fort Calhoun Station was issued a license amendment (Amendment No. 50) to allow operation at a steady state full rated power level of 1500 MWt, and the safety analysis described in this USAR was performed for a full rated power level of 1500 MWt.

3.2.2 Design Objectives

During normal operating conditions and anticipated transients, the reactor core, together with its control systems and the reactor protective system, is designed to function over its lifetime to prevent fuel damage based upon application of conservative limits for excessive fuel temperature, cladding strain, and cladding stress as specified in Section 3.2.3.

The combined response of all reactivity feedback mechanisms to an increase in reactor thermal power is a net decrease in reactivity. The combined effect of all reactivity coefficients in conjunction with the reactor control system provides stable reactor operation. If power oscillations do occur, their magnitude will be such that the fuel damage limits are not exceeded.

The maximum reactivity worth of the CEA's and the associated reactivity addition rate are limited by core, CEA and control element drive system designs to prevent rapid, large reactivity increases. Such reactivity increases are precluded in order to avoid violation of the fuel damage limits, rupture of the reactor coolant pressure boundary, or disruption of the core or other internals sufficient to impair the effectiveness of emergency cooling.

3.2.3 Design Criteria and Limits

3.2.3.1 Nuclear Criteria and Limits

The design of the core is based upon the following nuclear criteria and limitations:

a. The local fuel pellet burnup limit is determined by material and mechanical design rather than nuclear considerations. The conservatism of the resulting limit is confirmed by actual irradiation of demonstration fuel assemblies to the corresponding limit specified for that particular fuel design.

b. The combined response of all reactivity coefficients to an increase in reactor thermal power yields a net decrease in reactivity.

c. As noted in Section 3.1, CEA's are moved in groups to satisfy the requirements of shutdown, power level changes and operational maneuvering. The control systems are designed to produce power distributions that are within the acceptable limits on the overall nuclear heat flux factor (F^N_q) and departure from nucleate boiling ratio (DNBR) limits. The reactor protective system and the limiting conditions for operation assure that these limits are not exceeded.

d. Axial power distributions are manually controlled by trippable full length CEA's, using information provided by the out-of-core detectors.

e. The melting point of the UO_2 fuel shall not be reached during normal operation and anticipated transients.

3.2.3.2 Reactivity Control Criteria and Limits

The control system and operating procedures provide for adequate control of the core reactivity and power distributions such that the following are met:

a. Sufficient CEA's are withdrawn to provide an adequate shutdown reactivity margin following a reactor trip.

b. The shutdown margin is maintained with the highest worth CEA assumed stuck in its fully withdrawn position.

c. The chemical and volume control system is capable of adding boric acid to the reactor coolant at a rate sufficient to maintain the shutdown margin during a reactor coolant system cooldown at the design rate following a reactor trip.

3.2.3.3 Thermal and Hydraulic Criteria and Limits

The principal criterion for the thermal and hydraulic design is to avoid thermally induced fuel damage during normal operation and anticipated transients. It is recognized that there is a small probability of limited fuel damage in certain postulated events as discussed in Section 14.

The following corollary thermal and hydraulic design bases are established, but violation of either does not necessarily result in fuel damage:

a. There shall be a high confidence level that DNB is avoided during normal operation and anticipated transients.

b. For limiting Safety System Settings (LSSSs), Limiting Conditions of Operation (LCOs) and certain transients (Section 3.6) a design lower limit of 1.18 on minimum DNBR as calculated according to the CE-1 correlation (References 3-53, 3-54 & 3-55) was used.

The reactor protective system and the reactor control system provide for automatic reactor trip or corrective actions before these design limits are violated.

Reactor internal flow passages and fuel coolant channels are designed to prevent hydraulic instabilities. Flow maldistributions are limited by design to be compatible with the specified thermal design criteria.

3.2.3.4 Mechanical Design Criteria and Limits

The reactor internals are designed to safely perform their functions during steady state conditions and normal operating transients. The internals can safely withstand the forces due to deadweight, handling, system pressure, flow-inducted pressure drop, flow impingement, temperature differential, shock, and vibration. The structural components satisfy stress values given in Section III of the ASME Boiler and Pressure Vessel Code.

The following limitations on stresses or deformations are employed to assure the capability exists for a safe and orderly shutdown in the event of earthquake and major loss-of-coolant accident loading conditions. For reactor vessel internal structures, the stress criteria are given in Table 3.2-1. The intent of the limits in this table can be described as follows:

a. Under design loadings plus design earthquake forces, (see Appendix F) the critical reactor vessel internal structures are designed in accordance with the stress criteria established in Section III of the ASME Boiler and Pressure Vessel Code, Article 4.

b. Under normal operating loadings plus maximum hypothetical earthquake forces, the design criteria permit a small amount of local yielding.

c. Under normal operating loadings plus coolant pipe rupture loadings plus maximum hypothetical earthquake forces, permanent deformations are permitted by the design criteria.

In the loading combinations listed in Table 3.2-1, the earthquake forces include both horizontal and vertical seismic excitations acting simultaneously.

To properly perform their functions, the critical reactor internal structures are designed to satisfy the additional deflection limits described below, in addition to the stress limits given in Table 3.2-1.

Under loading combinations (a) and (b) of Table 3.2-1, deflections are limited, so that the CEA's can function and adequate core cooling is maintained. Under loading combination (c) of Table 3.2-1, the deflection design criteria depend on the size of the piping break. If the equivalent diameter of the pipe break is no larger than the largest line connected to the main reactor coolant lines, deflections are limited, so that the core is held in place, the CEA's function normally, and adequate core cooling is maintained. Those deflections which would influence CEA movement are limited to less than two-thirds of the deflection required to prevent CEA function. For pipe breaks larger than the above, the criteria are that the fuel is held in place in a manner permitting core cooling and that adequate coolant flow passages are maintained. Further, although not required for shutdown, all CEA's will be insertable. For the larger break sizes, critical components which meet the stress criteria of Table 3.2-1 are also restrained from buckling by further limiting the stress levels to two-thirds of the stress level calculated to produce buckling.

Table 3.2-1 - "Primary Stress Limits for Critical Reactor Vessel Internal Structures"

a.	Design Loadings Plus Design Earthquake	$P_m \leq S_m$
	Forces	$P_B + P_L \leq 1.5\ S_m$
b.	Normal Operating Loadings Plus Maximum	$P_m \leq S_D$
	Hypothetical Earthquake Forces	$P_B \leq 1.5\ (1 - (P_m)^2) S_D$ S_D
c.	Normal Operating Loadings Plus Maximum	$P_m \leq S_L$
	Hypothetical Earthquake Forces Plus	
	Pipe Rupture Loadings	$P_B \leq 1.5\ (1 - (P_m)^2) S_L$ S_L

where:

P_L, P_m, P_B, S_m, S_y are defined in the ASME Boiler and Pressure Vessel Code, Section III, Article 4

S_u = Minimum tensile strength of material at temperature

$S_L = S_y + (1/3)(S_u - S_y)$

S_D = Design Stresses = $1.2\ S_m$

Fuel Assemblies

The fuel assemblies are designed to maintain their structural integrity under steady state and transient operating conditions, as well as under normal handling, shipping, and refueling loads. The design takes into account differential thermal expansion of fuel rods, thermal bowing of fuel rods and CEA guide tubes, irradiation effects, and wear of all components. Mechanical tolerances and clearances have been established on the basis of the functional requirements of the components. All components including welds are highly resistant to the corrosive action of the reactor environment.
The fuel rod design takes into account external pressure, differential expansion of the fuel and clad, fuel swelling, clad creep, fission and other gas releases, thermal stress, pressure and temperature cycling and flow-induced vibrations. The structural criteria are based on the following:

a. The maximum tensile stress during steady state operation, expected transients, and depressurization is limited to two-thirds of the minimum yield strength of the material at temperature;

b. The net unrecoverable circumferential strain shall not exceed one percent as predicted by computations considering clad creep and fuel-clad interaction effects.

Control Element Assemblies (CEA's)

The CEA's are designed to maintain their structural integrity both under all steady state and transient operating conditions, and under handling, shipping and refueling loads. Thermal distortion, mechanical tolerances, vibration and wear are all taken into account in the design. Clearances and corresponding fuel assembly alignment are established so that the possible stackup of mechanical tolerances and thermal distortion would not result in frictional forces that could prevent reliable operation of the system. The structural criteria are based on limiting the maximum stress intensity to those values specified in Section III of the ASME Boiler and Pressure Vessel Code.

The control element drive mechanisms (CEDM's) are capable of actuating the CEA's under steady state and transient operating conditions and during hypothetical seismic occurrences. For pipe rupture accident loads, the CEDM's are designed to support and maintain the position of the CEA's in the core and to be capable of actuating them when these loads have diminished.

The speed at which the CEA's are inserted or withdrawn from the core is consistent with the reactivity change requirements during reactor operation. For conditions that require a rapid shutdown of the reactor, the CEDM clutches release to allow the CEA's and the connecting CEDM components to drop by gravity into the core. The reactivity is reduced during such a CEA drop at a rate sufficient to prevent violation of fuel damage limits.

The CEDM pressure housings are an extension of the reactor vessel, providing a part of the reactor coolant boundary, and are therefore, designed to meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels. Pressure and thermal transients as well as steady state loadings were considered in the design analysis.

3.4 NUCLEAR DESIGN AND EVALUATION

This section summarizes the nuclear characteristics of the design and discusses the design parameters which are of significance to the performance of the core in normal transient and steady state operational conditions. A discussion of the nuclear design methods employed and comparisons with experiments which support the use of these methods is included.

The numerical values presented are based on the Cycle 19 core. The analysis performed for this fuel cycle shows that all necessary requirements for safe operation have been met. Table 3.4-1 shows a summary of the nuclear design parameters for the fuel cycle. Figures 3.4-1 and 3.4-2 show the assembly average burnup distributions for both the beginning and end of Cycle 19, respectively.

Table 3.4-1 - "Nuclear Design Parameters"

Performance Characteristics

Fuel Management		19-Batch, Mixed Central Zone
Average Cycle Burnup, MWD/MTU		14,850 (Coast Down)
U-235 Enrichment, w/o (initial)		
Type M/	(5 assemblies, 8 Shims)	3.80
Type U1	(4 assemblies, 0 IFBA TAILS)	0.30
Type U2	(8 assemblies, 0 IFBA)	4.25
Type U3	(4 assemblies, 20 IFBA)	4.25
Type U4	(8 assemblies, 40 IFBA)	4.25
Type U5	(8 assemblies, 48 IFBA)	4.25
Type U6	(8 assemblies, 56 IFBA)	4.25
Type U7	(4 assemblies, 64 IFBA)	4.25
Type W1	(12 assemblies, 0 IFBA)	4.22
Type W2	(8 assemblies, 48 IFBA)	3.59
Type W3	(16 assemblies, 56 IFBA)	3.59
Type W4	(4 assemblies, 84 IFBA)	3.59
Type W5	(4 assemblies, 64 IFBA)	4.22
Type X1	(4 assemblies, 84 IFBA)	3.50
Type X2	(8 assemblies, 0 IFBA)	4.18
Type X3	(8 assemblies, 48 IFBA)	4.18
Type X4	(4 assemblies, 56 IFBA)	4.18
Type X5	(4 assemblies, 64 IFBA)	4.18
Type X6	(12 assemblies, 84 IFBA)	4.18
$H_20/U0_2$ Volume Ratio, Unit Cell (Cold)		1.66

Table 3.4-1 (Continued)

Control Characteristics	
k_{eff}, Beginning-of-Cycle, No Control Element Assemblies	
Cold (68°F)	1.17
Hot (532°F), Zero Power	1.14
Hot, Equilibrium Xe, Full Power	1.09
Number of Control Element Assemblies (CEA's)	
Full Length	45
Non-trippable	4
Total CEA Worth, %Δρ	
Beginning-of-Cycle	
Hot Zero Power (532°F)	7.64
End-of-Cycle	
Hot Zero Power (532°F)	8.32
Dissolved Boron	
Dissolved Boron Content for Criticality, ppm, (CEAs withdrawn, BOC)	
Cold (68°F), ppm	1455
Hot (532°F), Zero Power, ppm	1585
Hot (570°F), Equilibrium Xe, Full Power, ppm	1054
Dissolved Boron Content for Refueling, ppm	See COLR (TDB-VI)
Boron Worth, ppm/%Δρ	
Hot (570°F)	-129.2
Cold (68°F)	-93.6

Table 3.4-1 (Continued)

Reactivity Coefficients (CEA's Withdrawn)

Moderator Temperature Coefficient,	
mod, Δρ/°F	
Hot, Full Power (570°F)	
Beginning-of-Cycle	-0.60×10^{-4}
End-of-Cycle	-3.04×10^{-4}
Hot Zero Power (532°F)	
Beginning-of-Cycle	$+0.49 \times 10^{-4}$
Fuel Temperature Coefficient,	
fuel,Δρ/°F	
Hot, Zero Power (532°F, BOC)	-2.10×10^{-5}
Full Power (1065°F, BOC)	-1.61×10^{-5}
Moderator Void Coefficient,	
void, Δρ/% Void	
Hot, Operating (572°F)	
Beginning-of-Cycle	-0.18×10^{-3}
End-of-Cycle	-1.46×10^{-3}
Moderator Pressure Coefficient,	
p, Δρ/psi	
Hot, Operating, (572°F)	
Beginning-of-Cycle	$+0.4 \times 10^{-6}$
End-of-Cycle	$+2.1 \times 10^{-6}$

3.4.1 Reactivity and Control Requirements

Table 3.4-2 lists the effective multiplication factors (k_{eff}) and reactivity (ρ) under various conditions.

Table 3.4-2 - "Effective Multiplication Factors and Reactivity Under Various Conditions" (No Control Element Assemblies or Dissolved Boron, Beginning-of-Cycle Cycle 19 Core)

	k_{eff}	ρ
Cold (68°F), ARO	1.17	0.155
Hot (532°F), Zero Power	1.14	0.131
Hot, Full Power, Equilibrium Xe, ARO	1.09	0.082

The maximum excess reactivity ρ is 15.5 percent for the cold, unborated core at beginning of cycle. The reactivity decrease from zero to full power is due to the change in fuel temperature which causes Doppler broadening of the U-238 resonances and the change in moderator temperature coefficient which becomes more negative (due to a lower reactor coolant system boron concentration).

Reactivity control in the reactor is accomplished by adjusting both the position of the CEAs and the concentration of boric acid dissolved in the reactor coolant system. The CEAs permit rapid changes in reactivity, as required for reactor trip and to compensate for changes in moderator and fuel temperature and void formation associated with changes in power level. There are 45 standard and four full length non-trippable CEAs. The standard CEAs are used for shutdown and for regulation. The CEAs designated as shutdown CEAs are divided into two separately controlled groups; those designated as regulating CEAs are divided into four groups. During power operation, the shutdown groups are fully withdrawn while the position of the regulating groups is adjusted to meet reactivity and power distribution requirements. All CEAs except the full length non-trippable CEAs drop to a fully inserted position upon reactor trip.

Adjustment of the boric acid concentration is used to control the relatively slow reactivity changes associated with plant heatup and cooldown, fuel burnup, and certain xenon variations. Also, additional boric acid is used to provide a large shutdown margin for refueling. The use of boric acid dissolved in the reactor coolant makes it possible to maintain most of the CEAs in a withdrawn position during full power operation, thus minimizing the distortions in power distribution. Table 3.4-3 lists the concentrations of natural boron required to maintain the core critical under various conditions, assuming all control element assemblies are fully withdrawn.

Table 3.4-3 -"Dissolved Boron Requirements For Criticality (Control Element Assemblies Fully Withdrawn, Beginning-Of-Cycle)"

	Natural Boron, ppm
Cold (68°F)	1455
Hot (532°F), Zero Power, Clean	1585
Hot, Full Power, Equilibrium Xe	1054
Refueling	See COLR (TDB-VI)

The refueling boron concentration (without considering the worth of the CEAs) provides a reactivity shutdown of approximately 5 percent for the cold condition. The refueling concentration is specified as an equivalent ppm boric acid (H_3B0_3) in the coolant in the COLR which is approximately 10 percent of the solubility limit at refueling temperatures. After a normal shutdown or reactor trip, boric acid may be injected into the reactor coolant system to compensate for reactivity increases due to normal cooldown and xenon decay. Although the boric acid system reduces reactivity relatively slowly, the rate of reduction is more than sufficient to maintain the shutdown margin against the effects of normal cooldown and xenon decay.

Sufficient worth is available in the regulating CEAs to compensate for the rapid changes in reactivity associated with power level changes. In addition, these CEAs may be used for partial control of xenon transients and minor variations in moderator temperature and boron concentration. Table 3.4-4 summarizes the hot zero and hot full power control rod worths at both beginning and end-of-cycle. The reactivity variations are described in Sections 3.4.1.1 through 3.4.1.6. The worth of all CEAs, including shutdown CEAs, covers the reactivity variations and provides adequate shutdown with the most reactive CEA stuck in the fully withdrawn position. Margin is provided between the calculated CEA worth and the reactivity variations to account for uncertainties in the calculations. Only the 45 full length trippable CEAs are considered in Table 3.4-4.

Table 3.4.4 - "CEA Reactivity Allowances, %Δρ"

Control Rod Worth (%Δρ)	BOC		EOC (15,500 MWD/MTU)	
	HZP	HFP	HZP	HFP
Total Full Length Rod Worth	7.64	8.07	8.32	8.70
Stuck Rod Worth	1.26	1.22	1.23	1.26
Total Minus Stuck Rod Worth	6.38	6.85	7.09	7.44
PDIL Rod Insertion	1.06	0.26	1.14	0.34
Gross PDIL Scram Worth	5.32	6.59	5.95	7.10
Uncertainty (6.52%)	0.35	0.43	0.39	0.46
Net PDIL Rod Worth	4.97	6.16	5.56	6.64
Required Shutdown Margin	4.0	4.0	4.0	4.0
Excess Margin	0.97	2.16	1.56	2.64

3.4.1.1 Doppler Defect and Moderator Temperature Defect

The increase in reactivity associated with the change from full to zero power from both the Doppler effect in U-238 and the moderator temperature effect is 1.1%Δρ at beginning-of-cycle and 2.1%Δρ at end-of-cycle. This change in reactivity is compensated by CEA movement.

3.4.1.2 Axial Flux Redistribution

A change in reactivity occurs due to axial flux redistribution over a cycle as a result of the localized burn-out and redistribution of Xenon. This is conservatively estimated to be 0.2%Δρ at beginning-of-cycle and 0.4%Δρ at end-of-cycle.

3.4.1.3 Moderator Voids

A change in reactivity results from the formation of voids due to local boiling. The average void content in this core is very small and is estimated to be one-fourth of 1 percent at full power. As with the moderator temperature effect, the maximum increase in reactivity from full to zero power occurs at end-of-cycle when the least amount of dissolved boron is present. The maximum reactivity variation due to one-fourth of 1 percent voids is conservatively estimated to be 0.1 percent Δρ.

3.4.1.4 CEA Power Dependent Insertion Limit (PDIL)

The PDIL rod insertion is a measure of the rod worth associated with the permissible CEA configurations (see Figure 3.4-3) for both hot zero power and hot full power.

3.4.1.5 Maneuvering Band

An allowance is made in the reactivity worth of the CEA's to compensate for variations in xenon, dissolved boron concentration, and moderator temperature. When the CEA's reach the limits imposed on CEA motion, additional reactivity changes will be made by changing the boron concentration. The allowance made for the maneuvering band is 0.60 percent Δρ at BOC.

3.4.1.6 Shutdown Margin and Safeguards Allowance

An allowance of 4.0 percent Δρ at both the beginning-of-cycle (BOC) and at the end-of-cycle (EOC), respectively, has been made for the shutdown margin and safeguards allowances at hot, zero power conditions with the most reactive CEA stuck in the withdrawn position.

3.4.2 Reactivity Coefficients

The factors which contribute to the reactivity of a reactor, such as the thermal utilization, resonance escape probability, and nonleakage probabilities, are dependent upon certain parameters, such as moderator temperature and pressure and fuel temperature. Reactivity coefficients, denoted by α, relate changes in the core reactivity to variations in these parameters.

3.4.2.1 Moderator Temperature Coefficient

The reactivity worth of 1054 ppm of boron (amount of reactivity needed to maintain the reactor just critical at BOC full power conditions) increases from 8.16%$\Delta\rho$ to 8.61%$\Delta\rho$ as the moderator temperature decreases from operating to zero power temperature. The interaction of these temperature effects (along with the temperature coefficient of the unborated core) results in a net moderator temperature coefficient of reactivity, α_{mod}, at operating temperature which ranges from strongly negative to slightly positive, depending on the moderator temperature, the soluble boron content, and the fuel burnup.

In a core which is controlled by chemical shim dissolved in the moderator, there are two factors which cause the moderator temperature coefficient to become less negative as the fraction of reactivity controlled by the dissolved boron increases (i.e., at higher boron concentrations). First, an increase in moderator temperature reduces the effective density of the chemical poison and hardens the thermal neutron spectrum, thereby decreasing neutron absorption in the boron. Secondly, the effective reactivity worth of a solid poison such as the CEA's increases as the moderator temperature increases; thus, since there are fewer solid poison control elements than would be required in a reactor without chemical shim, the magnitude of this effect is reduced.

The calculated moderator temperature coefficient for various core conditions is given in Table 3.4-5. As shown in the table, the most positive value occurs at the beginning of cycle (HZP) when the dissolved boron content is at its maximum.

Table 3.4-5 - "Moderator Temperature Coefficients"

Conditions	(α_{mod}, $\Delta\rho/°F$)
Beginning of Cycle	
Hot, Full Power, CEAs Out	-0.60×10^{-4}
Hot, Zero Power, CEAs Out	$+0.49 \times 10^{-4}$
End of Cycle	
Hot, Full Power, CEAs Out, Zero ppm	-3.04×10^{-4}

The moderator coefficient becomes more negative with burnup, due mainly to the reduction in the dissolved boron content with burnup. The effects of plutonium and fission products are small when compared to the above; however, the buildup of xenon supplies a positive contribution to the coefficient for a constant boron concentration. Equilibrium xenon raises α_{mod} by 0.05×10^{-4}. However, when the dissolved boron concentration is reduced by the reactivity equivalent of xenon, the α_{mod} becomes more negative by 0.3×10^{-4} per °F.

The change in moderator temperature coefficient as a function of boron concentration is linear, being $+0.20 \times 10^{-4}$ per 100 ppm soluble boron.

3.4.2.2 Moderator Pressure Coefficient

The moderator pressure coefficient, α_p, is the change in reactivity per unit change in reactor coolant system pressure. Since an increase in pressure increases the water density, the pressure coefficient is opposite in sign to the temperature coefficient. The reactivity effect of increasing the pressure is reduced in the presence of dissolved boron because an increase in water density adds boron to the core. The calculated pressure coefficients for the beginning and end of the first cycle at full power were $+0.4 \times 10^{-6}$ $\Delta\rho$/psi and $+2.1 \times 10^{-6}$ $\Delta\rho$/psi, respectively.

3.4.2.3 Moderator Void Coefficient

During full power operation, some local boiling occurs resulting in a predicted average void fraction in the moderator of about one-fourth of 1 percent. Changes in reactivity are associated with the appearance of these voids in the moderator and are reflected in the void coefficient of reactivity, α_{void}. The presence of boron has a positive effect on the coefficient since an increase in voids results in a reduction in the boron content in the core. The calculated values at BOC and EOC are -0.18×10^{-3} $\Delta\rho$/% void and -1.46×10^{-3} $\Delta\rho$/% void, respectively.

3.4.2.4 Fuel Temperature Coefficient

The fuel temperature coefficient, α_{fuel} (commonly called the Doppler coefficient), reflects the change of core reactivity with fuel temperature. The effect may be broken into two parts, the thermal and the epithermal (Doppler) contributions. The thermal contribution is due to hardening of the spectrum as the temperature increases. The epithermal contribution is the temperature dependence of the resonance escape probability, which in turn is physically due to Doppler broadening of the resonances in U-238.

The variation in fuel coefficient over the fuel cycle is small. The hot full power coefficient is -1.61×10^{-5} $\Delta\rho$/°F at BOC and -1.67×10^{-5} $\Delta\rho$/°F at EOC.

3.4.2.5 Power Coefficient

The power coefficient, α_{power}, is the change in core reactivity per unit change in core power level. All of the previously mentioned coefficients contribute to the α_{power}, but only the moderator temperature coefficient and the fuel temperature coefficient are significant due to the relative magnitudes. To determine the change in reactivity with power, it is necessary to know the change in the weighted average fuel temperature with power. However, the determination of average fuel pellet temperatures is extremely complex. An "effective fuel temperature" may be defined as that temperature which gives the correct fuel temperature and power coefficients when used in a standard design calculation. The method used is contained in Reference 3-1.

This correlation which is a function of moderator temperature, fuel burnup and local power is incorporated into the standard design calculations.

3.4.3 Control Element Assembly Worths

Figure 3.4-4 is a schematic of one quadrant of the core cross section, showing the location and the groupings of the 45 trippable and 4 full length non-trippable CEA's. The total worth available from the trippable CEA's and the worth with the highest worth CEA stuck out are given in Table 3.4-6 for beginning and end-of-cycle.

Table 3.4-6 - "Calculated CEA Worths, %Δρ (@ 570°F)"

	Beginning-of-Cycle	End-of-Cycle
All 45 Standard CEA's Inserted	8.07	8.70
44 CEA's Inserted; Highest Worth CEA Stuck Out	6.85	7.44
Net Shutdown Worth	6.40	6.95

Table 3.4-7 gives the worth of each group of CEA's relative to the full power condition. These worths are from full out to full in. The CEA withdrawal procedure, meeting the minimum requirements of the PDIL, as shown in Figure 3.4-3 is as follows:

a. With the reactor subcritical, Shutdown Group A is fully withdrawn and then Shutdown Group B is withdrawn;

b. Regulating Groups 1 and 2 are fully withdrawn and Group 3 is withdrawn to at least 20% to take the core critical. Adjustments in dissolved boron concentration are made to maintain Group 3 above the PDIL;

c. Withdrawal of Groups 3 and 4 is made sequentially with the prescribed overlaps and within the specified range until the desired power level and power distribution is achieved.

Table 3.4-7 - "Worth of CEA Groups, %Δρ (@ BOC 532°F)"

When Sequenced as Listed Above*	Beginning-of-Cycle
Shutdown CEA's	
Group A	1.76
Group B	3.06
Regulating CEA's	
Group 1	0.86
Group 2	1.31
Group 3	0.64
Group 4	0.45

* The worth listed assumes that Group N is withdrawn after Group B in the sequence.

Adherence to the relationship of power to CEA insertion ensures that acceptable peaking factors are maintained within the bounds assumed for the LSSS and LCO shutdown margin is maintained, and that the potential consequences of a CEA ejection accident are limited to acceptable levels. Operation with the CEA's inserted beyond the PDIL is prevented by the rod block system.

3.4.4 Reactivity Insertion Rates

The maximum rate of reactivity insertion of the regulating groups at full power is 0.02%Δρ/sec. Analyses of CEA withdrawal incidents (Section 14.2) show that no core thermal limits would be exceeded for rates considerably in excess of the above values.

The maximum rate of reactivity insertion due to boron removal by operation of the chemical and volume control system is covered by the low end of the CEA insertion rate spectrum. Adequate time is available to take corrective measures as described in the analysis of the boron dilution incident in Section 14.3.

3.4.5 Power Distribution

The power distribution in the core, and in particular the peak heat flux and enthalpy rise, is of major importance in determining core thermal margin. The maximum expected peaking factors for Cycle 19 are 1.732 for F_r and 1.816 for F_{xy}. The corresponding COLR limits for the tilted values are 1.732 and 1.816. The maximum expected total peaking factor F_a^T, is 2.27 including uncertainties.

The behavior of the gross radial power distribution in the unrodded core through the 19th burnup cycle is shown in Figures 3.4-5, 3.4-6, and 3.4-7. The trend of the overall radial power distribution is to start at the center of the core at BOC, and remain toward the center through EOC.

CEA's are used to a minimum extent and in configurations that will result in a combined radial and axial peaking factor which is within the design limits stated above.

3.4.5.1 Malpositioned CEA's

The two worst cases of a malpositioned CEA were evaluated with respect to permissible operating modes and current administrative guidelines. The two cases evaluated are:

a. The worst case of a CEA left in the core and

b. Insertion of the CEA bank permissible at full power with one CEA left out.

Worst Case of a CEA Left in the Core

Startup or operation with the most reactive CEA left in the core would result in a large distortion in the radial power distribution and consequently excessive peaking. It is not necessary to analyze this event, because the rod block system will prevent its occurrence. This system prevents rod group motion with one CEA position deviating from the groups position by more than 12 inches. Group motion will stop before the 12 inch deviation limit is reached, preventing leaving one rod in and the rest of the group withdrawn. This limit is referenced in Technical Specification 2.10.2(4). In addition, the CEA position sensing system provides alarms from the synchros for deviations of four and eight inches.

If a CEA was not coupled to the rack and pinion drive system, which could occur after refueling, rack movement could indicate no deviation. This condition is assured not to exist by measuring the coupled and uncoupled weights of the letdown tool used in the CEDM coupling procedure. The weight difference between the coupled and uncoupled weights of 55 pounds for a single CEA and 110 pounds for a dual CEA proves that the CEAs are properly coupled. The administrative requirement for performing this test ensures that the CEA's are coupled, eliminating this concern.

Even if a rod was totally inserted with all others withdrawn, the inserted rod would be detected by the incore monitoring system which would cause alarms due to excessive flux peaking in detector locations across the core from the rodded and depressed neutron flux area. A detectable change in flux tilt would also be detected by both the incore and excore detector systems.

Insertion of the CEA Bank Permissible at Full Power
With One CEA Left Out

Insertion of Regulating Group 4 with one of the CEA's left out is prevented from occurring by the rod block circuitry. As in the case of leaving one CEA inserted, group motion will automatically be stopped and prevented prior to the rod-group deviation exceeding 12 inches. In addition, deviation alarms at four and eight inches provide warning of the asymmetry. Therefore, due to the rod block circuitry preventing the occurrence of this condition, no further analysis is required.

3.4.6 Pressure Vessel Fluence

The design of the reactor internals and of the water annulus between the active core and vessel wall is such that for NSSS operations at 1500 MWt and an 80 percent plant capacity factor, the integrated fast neutron fluence at the vessel/clad interface (E>1 MeV) is 3.54×10^{19} n/cm^2 over the 40-year design life of the vessel and corresponds to a fluence of 2.55×10^{19} n/cm^2 at the critical reactor vessel beltline weld (Reference 3-68). The fluence was determined using the threshold detector analysis for the surveillance capsule removed at the end of Cycle 7 (Reference 3-2). The SAND-II and ANISN computer codes were used to calculate the fast fluence at the reactor vessel clad interface.

The SAND-II computer code is used to calculate a neutron flux spectrum from the measured activities of the flux monitors. SAND-II requires an initial flux spectrum estimate; this is calculated using ANISN. The measured activities must be adjusted before they can be put into SAND-II. The various steps of the procedure are described below.

The measured activities must be decay corrected to reactor shutdown. Before being used by SAND-II, the foil activities must be converted to saturated activity with units of disintegrations per second per target atom (dps/a).

For U-238 fission product activities, the required SAND input has dimensions of fissions per second per U-238 atom (fps/a). This is obtained by dividing the saturated activity by the fractional fission yield of the fission product whose activity was measured.

The uranium foil is shielded with cadmium to prevent thermal fissioning in any U-235 impurities. However, the cadmium cover does not prevent fast fissioning in U-235. Therefore, an unshielded uranium foil is included in the flux monitor set. The activity of the unshielded foil can be used to determine the amount of fissioning in the shielded uranium foil caused by U-235. As a result of this calculation, the U-238 fission rate was determined to be 75% of the shielded uranium foil activity.

SAND-II requires an initial estimate of the neutron flux spectrum. This initial estimate was calculated using ANISN, a one-dimensional discrete ordinate code.

The peak fluence at the vessel/clad interface was determined to be 8.8 x 10^{18} n/cm^2 at the end of Cycle 7. To credit the azimuthal flux distribution and low radial leakage symmetric core loading patterns initiated in Cycle 8, conservative flux reduction factors were determined from DOT 4.3 azimuthal flux distribution plots.

No credit was assumed for axial flux distribution or the asymmetric core loading pattern of Cycle 10. Based on these assumptions the peak end of life reactor vessel/clad interface fluence was determined to be 3.54 x 10^{19} n/cm^2. The corresponding fluence to the critical reactor vessel beltline weld material is 2.55 x 10^{19} n/cm^2.

The SAND-II code will give fluxes that are accurate to within ±10% to ±30% if the errors in the measured activities are within similar limits. The 2-sigma uncertainties in the measured activities were less than ±12%. Therefore, it is estimated that the uncertainty in the measured fluence at the surveillance capsule location is ±20% to ±30%. The extrapolated fluence in the vessel will be slightly higher and is estimated to be ±30%.

3.4.7 Nuclear Evaluation

3.4.7.1 Nuclear Design Methods

The nuclear analysis design package developed for use in the design of low enrichment PWR cores is based on a combination of multigroup spectrum calculations, over which cross sections are appropriately averaged to obtain few group constants, and few group, one- two- and three-dimensional diffusion theory calculations of integral and differential reactivity effects and power distributions. The multigroup calculations include spatial effects in those portions of the neutron energy spectrum where volume homogenization is inappropriate, e.g., the thermal neutron energy range. The majority of the calculations are performed with the aid of computer programs embodying analytical procedures and fundamental nuclear data consistent with the current state-of-the-art.

Current design methods involve the use of OPPD's design methodology (as outlined in References 3.10.15 and 3.10.16) and Yankee Atomic Electric Company's Axial Shape/Setpoint Methodology (as outlined in Reference 3-74 and 3-75).

The OPPD design methodology has incorporated the Studsvik of America ICFM software package into the ABB-Combustion Engineering design methodology. OPPD has updated its methods utilizing CASMO-3 (References 3-76, 3-77 and 3-78) for generating cross-sections and SIMULATE-3 (References 3-79, 3-80 and 3-81) for X-Y power distribution depletions, reactivities and three dimensional analyses.

3.4.7.2 Comparisons With Experiments

Reactivity

The Combustion Engineering nuclear design package has been checked against a variety of critical and subcritical experiments. Table 3.4-8 summarizes the properties of the fuel rods employed in the lattices analyzed; Tables 3.4-9 and 3.4-10 summarize certain pertinent characteristics of the lattice and the eigenvalues calculated with the design package.

Table 3.4-8 - "Fuel Rod Description"

Laboratory	Clad OD (in.)	Clad Thickness (in.)	Clad Mat.	Fuel Pellet OD (in.)	Fuel Density (gm/cc)	Fuel Enrichment w/o U-235	Fuel Enrichment w/o PuO_2
B&W	0.4755	0.016	SS 304	0.4440	9.46	4.020	0
B&W	0.4748	0.032	A1 6061	0.4054	10.24	2.459	0
Yankee	0.3383	0.0161	SS 304	0.3000	10.18	2.700	0
Winfrith	0.4301	0.01051	SS 304	0.3984	10.44	3.003	0
Brookhaven	0.499	0.02743	SS 304	0.4441	9.30	3.006	0
Bettis	0.453	0.028	A1	0.3830	10.53	1.311	0
Hanford	0.426	0.027	Zr-2	0.372	9.646*	0.22	1.50
Battelle N. W. Westinghouse	0.568	0.030	Zr-4	0.508	9.869*	0.72	2.20

* effective fuel density

Table 3.4-9 - "Results of Analysis of Critical and Subcritical UO_2 Systems"

	Lattice	Pitch w/o U-235	(in.)	Boron $H_2O/UO2$	(ppm)	K_{eff}	Ref
B&W-1273	1	4.020	0.595	1.137	0	0.9998	3-7
	2	4.020	0.595	1.137	3390	1.0018	3-7
	3	4.020	0.571	0.956	0	0.9963	3-7
	4	2.459	0.595	1.371	0	1.0009	3-7
	5	2.459	0.595	1.371	1675	1.0016	3-7
B&W-3467	6	2.459	0.644	1.846	0	1.0004	3-8
	7	2.459	0.644	1.846	864	1.0014	3-8
	8	2.459	0.644	1.846	1536	0.9997	3-8
Yankee	9	2.700	0.405	1.048	0	0.9965	3-9
	10	2.700	0.435	1.405	0	0.9979	3-9
	11	2.700	0.470	1.853	0	0.9990	3-9
	12	2.700	0.493	2.166	0	1.0004	3-10
Winfrith	13 (20°C)	3.003	0.520	1.001	0	0.9987	3-11
	14 (80°C)	3.003	0.520	1.001	0	0.9977	3-11
	15	3.003	0.735	3.164	0	1.0009	3-11
	16	3.003	0.492	0.779	0	0.9992	3-11
Bettis	17	1.311	0.6133=	1.429	0	0.9963	3-12
	18	1.311	0.6133=	1.429	0	0.9963	3-12
	19	1.311	0.6133=	1.429	0	0.9970	3-12
	20	1.311	0.6504=	1.781	0	0.9962	3-12
	21	1.311	0.6504=	1.781	0	0.9975	3-12
	22	1.311	0.7110=	2.401	0	0.9968	3-12
	23	1.311	0.7110=	2.401	0	0.9975	3-12
BNL[(a)]	24	3.006	0.6767=	1.319	0	0.9997	3-13
	25	3.006	0.6767=	1.319	1363	0.9932	3-13
	26	3.006	0.7163=	1.632	0	0.9964	3-13
	27	3.006	0.7163=	1.632	470	0.9950	3-13
	28	3.006	0.7163=	1.632	992	0.9931	3-13
	29	3.006	0.7163=	1.632	1345	0.9940	3-13
	30	3.006	0.7706=	2.091	0	0.9981	3-13
	31	3.006	0.7706=	2.091	1141	0.9931	3-13
B&W[(a)]	32 (66°F)	4.020	0.595	1.137	0	1.0046	3-14
	33 (103°F)	4.020	0.595	1.137	0	1.0036	3-14
	34 (203°F)	4.020	0.595	1.137	0	1.0003	3-14
	35 (308°F)	4.020	0.595	1.137	0	0.9992	3-14
	36 (406°F)	4.020	0.595	1.137	0	1.0010	3-14

= Triangular Pitch

(a) Subcritical Measurements

Table 3.4-10 - "Results of Analysis of PuO_2-U02 Fueled Lattices"

Lattice	w/o U-235	Pitch w/o Pu-02	(in.)	Boron H_2O/Fuel	(ppm)	K_{eff}	Ref
Hanford	0.22	1.50	0.55=	1.099	0	1.0027	3-15
			0.60=	1.557	0	1.0056	3-15
			0.71=	2.705	0	1.0108	3-15
			0.80=	3.788	0	1.0094	3-15
BNWL	0.72	2.2(1)	0.85=	1.837	0	1.0056	3-16
			0.93=	2.445	0	1.0099	3-16
WCAP 0.72	2.2(1)	0.69	1.099	0	0.9994	3-17	
			0.75	1.525	0	1.0058	3-17
			0.69	1.099	261	0.9998	3-17
			0.9758	3.448	261	1.0122	3-17
			0.69	1.099	526	1.0005	3-17
			0.9758	3.448	526	1.0099	3-17
BNWL	0.72	2.2(2)	0.93=	2.445	0	1.0112	3-16
			1.05=	3.461	0	1.0068	3-16
BNWL	0.72	2.2(3)	0.85=	1.837	0	1.0113	3-16
			0.93=	2.445	0	1.0123	3-16
WCAP 0.72	2.2(3)	0.9758	3.448	0	1.0206	3-16	

= Triangular Pitch
(1) 7.654 w/o Pu-240 in Pu
(2) 16.54 w/o Pu-240 in Pu
(3) 23.503 w/o Pu-240 in Pu

The average eigenvalue for the critical uranium lattices in Table 3.4-9 (numbers 1 through 23) is 0.9987 ± 0.0019 and for the mixed oxide lattices of Table 3.4-10 the corresponding number is 1.00799 ± .0053. The $U0_2$ experiments cover a wide range of core dimensions, boron concentrations, temperature, enrichment, water-to-fuel ratios, and clad materials, thus giving confidence in the validity of the design package to predict beginning-of-life fuel properties with an acceptable accuracy. The analysis of the mixed oxided lattices exhibits larger deviations than for the $U0_2$ lattices; this result is not surprising in view of the limited amount of data compared with $U0_2$ systems, the relatively large experimental bucklings, and uncertainties in the same.

The rods-out, beginning-of-cycle, cold and hot zero power reactivities of the Obrigheim (Reference 3-13) and Connecticut Yankee (Reference 3-14) reactors were also calculated to demonstrate the validity of the model in large multiregion cores.

The results are summarized here:

	Reactor	Temperature	Boron (ppm)	K_{eff}
(a)	Obrigheim	cold	1727	0.9964
		hot	1962	0.9989
(b)	Connecticut Yankee	260°F	2040	1.0025
		560°F	2305	1.0002

Table 3.4-11 summarizes the predicted and measured values of the BOC hot zero power critical boron concentrations, with all-rods-out (ARO), for Cycles 1 through 18. This table shows excellent agreement between the two values for each cycle.

Table 3.4-11 - "BOC HZP Critical Boron Concentration (ARO)"

Cycle	Predicted C_B (ppm)	Measured C_B (ppm)
1	911	933
2	1248	1240
3	964	996
4	1023	1027
5	1235	1242
6	1230	1230
7	1240	1241
8	1239	1240
9	1520	1518
10	1481	1474
11	1502	1502
12	1510	1507
13	1560	1568
14	1201	1182
15	1392	1411
16	1534	1546
17	1592	1621
18	1581	1608

Depletion Calculation

Over 50 spent fuel samples from Yankee Core I were subjected to isotopic and radio-chemical analyses which were performed in the Tracerlab Laboratory at Richmond, California and by the Vallecitos Atomic Laboratory of the General Electric Company (Reference 3-15). Depletion calculations were performed on the Yankee core for comparison with the above measurements. Figure 3.4-9 compares measured and calculated values of the Pu/U mass ratio versus exposure, and Figure 3.4-10 shows a comparison for the relative isotopic composition of plutonium as a function of fractional U-235 depletion.

The inventory changes for the 74 fuel assemblies from Yankee Core I are compared with measured results (Reference 3-16) in Table 3.4-12; the calculations were carried out using both one-dimensional and three- dimensional (RZ) representations.

Table 3.4-12 - "Inventory Change Comparison"

	U-235 Dep. (kg)	Total Pu. (kg)	Fiss. Pu. (kg)	Fissile Consumption (g/MWd)
NFS Meas.	171.0±4.7	91.1±1.0	80.27±0.88	0.535±0.028
1-D	170.8	91.0	80.88	0.530
3-D (RZ)	169.0	89.9	79.48	0.528

Doppler and Power Coefficient

The Doppler coefficient of reactivity is due to Doppler broadening of the U-238 resonances with increasing fuel temperature. The power coefficient of reactivity is the change in reactivity associated with the Doppler and moderator coefficients as a function of power. The fuel temperature used to calculate the Doppler coefficient as a function of the core average power level and coolant temperature is determined on the basis of the Reference 3-1 model. Table 3.4-13 shows a comparison between the predicted and measured power coefficients, beginning with Cycles 2 through 18. All of the pairs of measured and predicted values show good agreement.

Table 3.4-13 - "Comparison of Predicted and Measured Power Coefficients"

Cycle	Burnup MWD/MTU	Percent of Rated Power	Critical Boron Concentration	Predicted Power Coefficient (Δρ/% Power)	Measured Power Coefficient (Δρ/% Power)
2	10877	46(1)	104	-1.70×10^{-4}	-1.95×10^{-4}
3	157	46(1)	720	-1.60×10^{-4}	-1.47×10^{-4}
3	1513	90(1)	535	-1.20×10^{-4}	-1.12×10^{-4}
3	4183	90(1)	309	-1.26×10^{-4}	-1.31×10^{-4}
3	7208	90(1)	62	-1.74×10^{-4}	-1.48×10^{-4}
4	267	92(1)	690	-1.06×10^{-4}	-1.04×10^{-4}
4	4690	94(1)	288	-1.31×10^{-4}	-1.12×10^{-4}
4	8027	95(1)	44	-1.52×10^{-4}	-1.10×10^{-4}
5	426	93(1)	876	-0.65×10^{-4}	-1.05×10^{-4}
5	6815	94(1)	296	-1.33×10^{-4}	-1.25×10^{-4}
6	400	95(1)	848	-1.18×10^{-4}	-1.11×10^{-4}
6	6467	96(2)	307	-1.53×10^{-4}	-1.45×10^{-4}
7	450	96(2)	817	-1.20×10^{-4}	-0.98×10^{-4}
7	6900	95(2)	283	-1.45×10^{-4}	-1.30×10^{-4}
7	7800	95(2)	192	-1.52×10^{-4}	-1.57×10^{-4}
8	459	79(2)	817	-1.18×10^{-4}	-1.18×10^{-4}
8	6150	95(2)	292	-1.41×10^{-4}	-1.70×10^{-4}
9	420	95(2)	1036	-0.86×10^{-4}	-1.64×10^{-4}
9	9663	96(2)	300	-1.35×10^{-4}	-1.57×10^{-4}
10	583	95(2)	1017	-0.90×10^{-4}	-1.24×10^{-4}
10	9261	95(2)	302	-1.39×10^{-4}	-1.40×10^{-4}
11	433	93(2)	1073	-0.91×10^{-4}	$-.95 \times 10^{-4}$
11	9765	95(2)	301	-1.41×10^{-4}	-1.52×10^{-4}
12	425	93(2)	1050	-0.95×10^{-4}	-1.42×10^{-4}
12	9691	95(2)	309	-1.43×10^{-4}	-1.63×10^{-4}
13	373	94(2)	1113	-0.91×10^{-4}	-1.26×10^{-4}
13	10694	95(2)	325	-1.52×10^{-4}	-1.51×10^{-4}
14	355	90(2)	768	-1.25×10^{-4}	-1.55×10^{-4}
14	10559	94(2)	319	-1.73×10^{-4}	-1.77×10^{-4}
15	340	95(2)	948	-1.05×10^{-4}	-1.57×10^{-4}
15	10180	94(2)	303	-1.48×10^{-4}	-2.13×10^{-4}
16	400	95(2)	1088	-0.95×10^{-4}	-1.54×10^{-4}
16	11,350	95(2)	315	-1.46×10^{-4}	-2.08×10^{-4}
17	237	92(2)	1104	-1.06×10^{-4}	-1.70×10^{-4}
17	10,550	95(2)	311	-1.53×10^{-4}	-2.27×10^{-4}
18	420	95(2)	1102	-0.99×10^{-4}	-1.66×10^{-4}
18	11,806	97(2)	287	-1.40×10^{-4}	-2.14×10^{-4}

(1) Full Rated Power = 1420 MWt
(2) Full Rated Power = 1500 MWt

Moderator Temperature Coefficient

The moderator temperature coefficient (MTC) for Fort Calhoun has been measured at both BOC and EOC (approximate), for full power conditions, beginning with Cycle 1. Table 3.4-14 summarizes the measurements and predictions which show good agreement for all cycles.

Power Distributions

Comparisons between predicted and measured power distributions using the design methodology of Combustion Engineering are contained in Reference 3-17.

Table 3.4-14 - "Comparison of Predicted and Measured Moderator Temperature Coefficients"

	BOC				EOC			
	Percent of Rated	Critical Boron	Predicted MTC	Measured MTC	Percent of Rated	Critical Boron Concentration	Predicted MTC	Measured MTC
1	-	-	-	-	75[1]	239	-1.02×10^{-4}	-0.98×10^{-4}
2	69[1]	927	-0.27×10^{-4}	-0.28×10^{-4}	46[1]	104	-1.66×10^{-4}	-1.62×10^{-4}
3	46[1]	720	-1.04×10^{-4}	-0.41×10^{-4}	90[1]	62	-2.04×10^{-4}	-1.65×10^{-4}
4	92[1]	690	-0.66×10^{-4}	-0.42×10^{-4}	95[1]	44	-2.16×10^{-4}	-1.41×10^{-4}
5	93[1]	876	-0.64×10^{-4}	-0.19×10^{-4}	94[1]	296	-1.33×10^{-4}	-0.97×10^{-4}
6	95[1]	848	-0.61×10^{-4}	-0.34×10^{-4}	96[2]	307	-1.79×10^{-4}	-1.38×10^{-4}
7	96[2]	817	-0.61×10^{-4}	-0.40×10^{-4}	95[2][3]	192	-1.79×10^{-4}	-1.79×10^{-4}
8	80[2]	817	-0.84×10^{-4}	-0.74×10^{-4}	95[2]	292	-1.66×10^{-4}	-1.76×10^{-4}
9	95[2]	1036	-0.32×10^{-4}	-0.29×10^{-4}	96[2]	300	-1.53×10^{-4}	-1.37×10^{-4}
10	95[2]	1017	-0.41×10^{-4}	-0.36×10^{-4}	95[2]	302	-1.63×10^{-4}	-1.41×10^{-4}
11	93[2]	1073	-0.39×10^{-4}	-0.31×10^{-4}	95[2]	301	-1.65×10^{-4}	-1.47×10^{-4}
12	93[2]	1050	-0.39×10^{-4}	-0.41×10^{-4}	95[2]	309	-1.68×10^{-4}	-1.66×10^{-4}
13	94[2]	1113	-0.27×10^{-4}	-0.39×10^{-4}	95[2]	325	-1.71×10^{-4}	-1.58×10^{-4}
14	90[2]	768	-0.81×10^{-4}	-0.74×10^{-4}	94[2]	319	-1.80×10^{-4}	-1.67×10^{-4}
15	95[2]	939	-0.57×10^{-4}	-0.68×10^{-4}	94[2]	303	-1.77×10^{-4}	-1.89×10^{-4}
16	95[2]	1088	-0.44×10^{-4}	-0.55×10^{-4}	95[2]	315	-1.804×10^{-4}	-2.08×10^{-4}
17	92[2]	1104	-0.66×10^{-4}	-0.79×10^{-4}	95[2]	311	-2.15×10^{-4}	-2.23×10^{-4}
18	95[2]	1102	-0.63×10^{-4}	-0.67×10^{-4}	97[2]	287	-2.15×10^{-4}	-2.08×10^{-4}

[1] Full Rated Power = 1420 MWt
[2] Full Rated Power = 1500 Mwt
[3] Predicted

3.4.8 Reactor Stability

Xenon stability analyses on the Fort Calhoun core indicate that any radial and azimuthal xenon oscillations induced in the core will be damped, but that the core could exhibit instabilities with respect to axial xenon oscillations during certain portions of the burnup cycle, in the absence of appropriate control action. Before discussing the methods of analysis employed to obtain these predictions, it is appropriate to reiterate several important aspects of the xenon oscillation problem.

a. The time scale on which the oscillations occur is long, and any induced oscillations typically exhibit a period of 30 to 50 hours;

b. Xenon oscillations are detectable as discussed below;

c. As long as the initial power peak associated with the perturbation initiating the oscillation is acceptable, the operator has time in the order of from hours to days to decide upon and to take appropriate remedial action prior to the time when allowable peaking factors would be exceeded.

3.4.8.1 Method of Analysis

The classic method for assessing spatial xenon oscillations is that developed by Randall and St. John (Reference 3-18) which consists of expanding small perturbations of the flux and xenon concentrations about equilibrium values in eigenfunctions of the system with equilibrium xenon present. While the Randall-St. John technique is correct only for a uniform unreflected system, its use of the separations between the eigenvalues of the various excited states of the system and the eigenvalue of the fundamental state is helpful in directing attention to which of the various excited states are the most likely to occur. As indicated in Figure 3.4-11, the first axial mode, which has the minimum eigenvalue separation from fundamental mode, is the most likely to occur, and the higher modes would have, on the basis of this simple theory, the indicated relative likelihoods of occurrence.

However, it is necessary to extend this simpler linear analysis to treat cores which are non-uniform because of fuel zoning, depletion and CEA patterns, for example. Such extensions have been worked out and are reported in References 3-19 and 3-20. In this extension, the eigenvalue separations between the excited state of interest and the fundamental are computed numerically for symmetrical flux shapes. For nonsymmetrical flux shapes, the eigenvalue separation can usually be obtained indirectly from the dominance ratio $\lambda 1/\lambda 0$, computed during the iteration cycle of the machine spatial calculation.

In making the analysis, numerical space-time calculations are performed in the required number of spatial dimensions for the various modes as checkpoints for the predictions of the extended Randall-St. John treatment described above.

3.4.8.2 Radial Mode Oscillations

From the remote position of the first radial excited eigenvalue in Figure 3.4-11 (over 4 percent in λ), it is expected that such oscillations would be rapidly damped even in a core whose power was flattened for example, by enrichment zoning. To confirm that this mode is extremely stable, a space-time calculation was run for a reflected, zone core 11 feet in diameter without including the damping effects of the negative power coefficient. The initial perturbation was a poison worth 0.4 percent in reactivity placed in the central 20 percent of the core for 1 hour. Following removal of the perturbation, the resulting oscillation was followed in 4-hour time steps for a period of 80 hours. As shown in Figure 3.4-12, the resulting oscillation died out very rapidly with a damping factor of about -0.06 per hour. If this damping coefficient is corrected for a finite time mesh by the formula in Reference 3-21, it would become even more strongly convergent. On this basis, it is concluded that radial oscillations are highly unlikely.

This conclusion is of particular significance because it means that there is no type of oscillation where the inner portions of the core act independently of the peripheral portions of the core whose behavior is most closely followed by the out-of-core flux detectors. As will be noted later, primary reliance is placed on these for the detection of any xenon oscillations.

3.4.8.3 Azimuthal Mode Oscillations

Azimuthal oscillations in an unreflected uniform reactor are less likely than axial mode oscillations as indicated in Figure 3.4-11. The situation is quite different in a radially power-flattened reflected core even at beginning of cycle, as shown in Figure 3.4-13. Here, the eigenvalue separations for the actual core are predicted by the modified Randall-St. John treatment and include the effects of power flattening. On the basis of this information, it appears that the azimuthal mode is the most easily excited at beginning of life even though the axial mode becomes the most unstable later.

With reference to Figure 3.4-13, it is indicated that the eigenvalue separation between the first azimuthal harmonic and the fundamental is about 1.2 percent in λ. Although the axial oscillations were found to be relatively insensitive to the moderator temperature feedback because of the constant power condition, the azimuthal modes should be stabilized appreciably by the negative moderator coefficient. Furthermore, the Doppler coefficient applicable to the Fort Calhoun reactor is calculated to be approximately -1.35×10^{-3} $\Delta\rho$/kW-ft^{-1}, which is more than enough to ensure stability of all the azimuthal modes.

3.4.8.4 Axial Mode Oscillations

As checkpoints for the predictions of the modified Randall-St John approach, numerical spatial time calculations have been performed for the axial case at both beginning and end of cycle. The fuel and poison distributions were obtained by depletion with soluble boron control so that, although the power distribution was strongly flattened, it was still symmetric about the core midplane. Spatial Doppler feedback was included in these calculations. In Figure 3.4-14 the time variation of the thermal neutron flux is shown for two points along the core axis near end of life with Doppler feedback. The initial perturbation used to excite the oscillations was a 20 percent insertion into the top of the reactor of a 1.5 percent reactivity CEA bank for 1 hour. As is indicated, the damping factor for this case was about +0.02 per hour. When corrected for finite time mesh by the methods of Reference 3-21, however, the damping factor is approximately +0.05.

When this damping factor is plotted on Figure 3.4-13 at the appropriate eigenvalue separation for this mode at end of cycle, it is apparent that good agreement is obtained with the modified Randall-St. John prediction.

At beginning of cycle, the space-time calculations indicated a positive damping coefficient of about +0.04 per hour in the absence of spatial Doppler feedback, and a negative damping coefficient of -0.05 per hour results with a power coefficient of -1.35×10^{-3} $\Delta\rho$/kW-ft^{-1}. Again, these space-time results are in excellent agreement with the predictions of the modified Randall-St. John technique.

Calculations performed with both Doppler and moderator feedback have resulted in damping factors which were essentially the same as those obtained with Doppler feedback alone. This result suggests that the constant power condition which applies to the axial oscillations results in a very weak moderator feedback since the moderator density is fixed at the top and bottom of the core and only the density distribution in between can change.

For the estimated Doppler coefficient of -1.35×10^{-3} $\Delta\rho$/kW-ft^{-1} (see Section 3.4.2.4) it can be seen from Figure 3.4-12 that the damping factor toward end of the burnup cycle is positive; thus within the uncertainties in predicting power coefficients and uncertainties in the analysis, there is a possibility of unstable axial xenon oscillations in the absence of any control action. These oscillations, however, are sufficiently slow, (doubling time of 14 hours with a damping factor of +0.05 hr^{-1}, detected as outlined below), that there would be sufficient time to institute corrective action.

3.4.8.5 Detection of Xenon Oscillations

Primary reliance for the detection of any xenon oscillations is placed on the out-of-core flux monitoring instrumentation, one channel of which per quadrant is an axially split ionization detector. As indicated earlier, oscillations in modes such as the radial, which would allow the center of the core to behave independently from the peripheral portions of the core, are highly unlikely and this lends support to reliance on the out-of-core detectors for this purpose.

Furthermore, as an example of the ability of the axially split out-of-core detectors to respond to axial flux tilts (i.e., axial shape index (ASI)) in the core, Figure 3.4-15 indicates the ratio of the lower half of the axially split detector signal to the signal from the upper half for two different power distributions; one is axially symmetric, the other contains a strong contribution from the first axial harmonic and has a peaking factor of about 1.8. In the latter case, the signal from the lower half of the detector is 50 percent higher than that from the upper half.

Considering that the primary response of these detectors will be to the power in the peripheral fuel assemblies, but noting that the lower modes of any induced oscillations will affect the power shapes in these peripheral assemblies, it has been concluded that any flux tilts can be observed and identified by the use of out-of-core instrumentation to provide data upon which appropriate remedial action can be based.

The incore or core average ASI, Y_I is related to the excore detector ASI, Y_E, by the Shape Annealing Factor (SAF) in the following equation.

$$Y_I = SAF*Y_E + bias$$

The SAF is a function of excore detector geometry and is determined by reactor physics testing. The incore detectors (see Section 3.6.1) were utilized to compute the core average ASI, the excore detector signals were used to calculate excore ASI and the equation was solved for the SAF during either a power change or a controlled axial xenon oscillation. The core average ASI is computed by the incore detectors during normal operation. This value is compared to Y_I computed by the excore detectors on a periodic basis and if the difference between the two exceeds a prescribed limit the split excore detectors are calibrated to assure that the "correct" value of Y_I is calculated.

3.4.8.6 Control of Xenon Oscillations

The split detectors of the power range safety and control channels are used to calculate the Axial Shape Index, ASI, which is defined as the ratio of the difference and sum of the signals from the lower and upper detectors respectively. Three separate limits have been established for allowable ASI as a function of reactor power.

These functions allow the axial peaks to increase as reactor power decreases. The first and most restrictive of these limits is the maintenance of the ASI around an Equilibrium Shape Index (ESI). The ESI is defined as the ASI when the core is at a constant power level with an equilibrium xenon concentration and all CEA's removed from the core. The operator is to maintain the ASI within a given band using the Bank 4 CEA's for fuel performance considerations.

The second limit is a Limiting Condition for Operation (LCO) based either on DNBR or peak linear heat generation rate. Since the peak linear heat generation rate is usually monitored by the incore detectors the DNBR LCO defines the ASI limit during normal operation. The Technical Specifications state that if the ASI exceeds the DNBR LCO, it is to be restored to within the limits in two hours or take the reactor to less than 15% of rated power in the next eight hours. The operator is to utilize the Bank 4 CEA's to maintain the ASI within limits.

The last and least restrictive limit on ASI as a function of power is the Reactor Protective System Axial Power Distribution protection channels. Each independent channel compares the observed ASI with the ASI limit. A trip is initiated on two out of four logic if the ASI exceeds the limit. The limit is derived through consideration of the DNBR and the peak linear heat generation rate for various ASI's.

3.4.8.7 Xenon Oscillation Operating Experience

Section 3.4.8.4 discusses the theoretical possibility of unstable axial xenon oscillations in the absence of any control action. During the operation of the Fort Calhoun Station from August, 1973 through December, 1986, no unstable axial xenon oscillations have been observed. Near end of cycle stable axial xenon oscillations with slightly positive damping factors have been observed. Half cycle damping techniques utilizing Bank 4 CEA's have been successfully utilized to control these oscillations.

3.6 THERMAL AND HYDRAULIC DESIGN EVALUATION FOR LIMITING TRANSIENTS AND SETPOINT ANALYSIS FOR CYCLE 19

3.6.1 General

The thermal and hydraulic design of the reactor has as its primary objective the assurance that the core can meet normal steady state and transient performance requirements without exceeding thermal and hydraulic design limits. This section, therefore, discusses the thermal and hydraulic characteristics that relate reactor performance to the margin of design limits.

The following transients were either re-evaluated for validity or reanalyzed for Cycle 19. The criteria of this section was applied to the Cycle 19 setpoint analysis and the following transients:

14.2 CEA Withdrawal Incident [1]
14.3 Boron Dilution Incident [2]
14.4 CEA Drop Incident [1]
14.6 Loss of Coolant Flow
-Loss of Coolant Flow [2]
-Seized Rotor [1]
14.11 Excess Load Increase [1]
14.12 Main Steam Line Break Accident [1]
14.13 CEA Ejection Accident [1]
14.15 Loss of Coolant Accident [2]
14.22 Reactor Coolant System Depressurization Incident [1]

[1] Reanalyzed
[2] Evaluated Cycle 19 inputs to analysis to demonstrate existing analysis remains bounding

The thermal and hydraulic design is based on a limiting minimum departure from nucleate boiling ratio of 1.18 as calculated using the CE-1 correlation. To ensure that this limit is not exceeded the reactor protective system is designed to trip the reactor before this condition can be achieved.

This section also discusses the fuel pellet performance characteristics that relate the reactor performance to the margin to design limits. The fuel pellet performance design limit ensures that fuel pellet centerline melt does not occur. The fuel centerline melt design criterion is based on maintaining the peak linear heat rate below prescribed limit of 22 kW/ft. To ensure that this limit is not exceeded, the reactor protective system is designed to trip the reactor before this condition can be achieved.

A summary of thermal and hydraulic parameters used for limiting Cycle 19 analyses is presented in Table 3.6-1.

Table 3.6-1 - "Thermal and Hydraulic Parameters for Cycle 19 Analyses"

General Characteristics at Full Power

Total Heat Output, MWt	1500
Btu/hr	5119×10^6
Heat Generated in Fuel, Fraction (i.e., fuel and cladding)	0.975
Pressurizer Pressure	
Nominal, psia	2,100
Minimum in Steady State, psia	2,075
Maximum in Steady State, psia	2,150
Design Inlet Temperature, Steady State, °F	545
Total Reactor Coolant Volumetric Flow,gpm	202,500
lb/hr	76.47×10^6
Coolant Flow Through Core, lb/hr	73.06×10^6
Hydraulic Diameter Nominal Channel, ft	0.0444
Average Mass Velocity, lb/hr-ft^2	2.2247×10^6
Average Coolant Velocity In-Core, ft/sec	13.13
Core Average Heat Flux, Btu/hr-ft^2	178,378
Total Heat Transfer Area, ft^2	28,700*
Average Linear Heat Rate of Rod, kW/ft	6.02
Design Overpower, %	112

*Ten fuel rods assumed to be replaced with stainless steel rods.

3.6.2 Coolant Flow

3.6.2.1 Total Coolant Flow Rate and Bypass Flow

The credited nominal total coolant flow rate at full power is 76.47×10^6 lb/hr. The coolant flow path can be traced in Figure 3.1-1. Coolant enters the four inlet nozzles and flows into the annular plenum between the reactor vessel and the core support barrel. It then flows down on both sides of the thermal shield and through the flow skirt to the plenum below the core lower support structure. Pressure losses in the skirt and lower support structure help to even out the inlet flow distribution to the core. The coolant passes through the openings in the lower flow plate and flows axially upward through the fuel assemblies. A portion flows through the lower core plate and into the guide tubes in the fuel assemblies. Flow limiting devices have been placed in the guide tubes of fuel assemblies located under spare CEDM locations to prevent asymmetric bypass flows. After passing through the core, the coolant flows into the region outside the control element shrouds. From this region the coolant flows across the control element assembly shrouds and passes out through the outlet sleeves on the core barrel to the outlet nozzles.

The principal core bypass routes are direct inlet-to-outlet coolant flow at the joint between the core support barrel sleeve and the outlet nozzle and the flow in the reflector region in excess of that required for cooling. The design limits the total guide tube flow and core bypass to a maximum of 3.41 x 10^6 lb/hr, yielding a core flow rate of 73.06 x 10^6 lb/hr. Some internal leakage occurs within the core and is included in the 73.06 x 10^6 lb/hr flow rate.

The coolant required to cool the control elements flows in the annulus between the control element and the guide tube and then into the region outside the control element assembly shrouds. A similar but smaller leakage will occur at the upper end of those guide tubes without control elements.

3.6.2.2 Inlet Flow Distribution

The Cycle 19 core is a mixed core design which includes five ABB/CE fuel assemblies from previous cycles and 128 Westinghouse fuel assemblies. The inlet flow distribution was revised to account for the difference in inlet pressure losses between the ABB/CE and Westinghouse fuel. Methodologies to calculate the new inlet flow distribution used in the Cycles 14, 15 and 16 Thermal Hydraulics analyses were used, (Ref. 3-72, 3-73, and 3-74). Westinghouse assemblies used a new inlet grid design which reduces debris flow into the fuel assemblies, thereby reducing the chance of debris induced fuel damage.

3.6.2.3 Exit Pressure Distribution

The exit pressure distributions were derived in conjunction with the inlet flow distribution. For each inlet flow distribution there is a corresponding exit pressure distribution. They are also input directly into the hydraulic code. The exit pressures allow the code to more accurately predict the delta pressure across each assembly. This in turn leads to more accurate modeling of the enthalpy rise in each assembly.

3.6.2.4 Partial Flow Loop Operations

There are two steam generators and four reactor coolant pumps which give rise to six possible configurations of operation. At present the Fort Calhoun Nuclear Station is only licensed for the normal four-pump configurations.

One three pump configuration was analyzed (Ref. 3-82). This analysis assumed flow in three loops and no flow (forward or reverse) in the fourth loop. Inlet flow distribution and its associated exit pressure for the three pump configuration served as input to the thermal-hydraulics code. The results were then used to evaluate the Seized Rotor Incident (Section 14.6) and its impact on thermal margin degradation.

3.6.3 Peak Linear Heat Rate

The peak linear heat rate (PLHR), in the limiting fuel pin in the core shall not exceed that corresponding to the onset of fuel centerline melt. This fuel melt limit was calculated by the FATES computer code (Reference 3-67) to be 22 kW/ft.

The parameters affecting this fuel design limit are as follows:

- Core Power;
- Axial Power Distribution;
- Radial Power Distribution;
- Azimuthal Tilt Magnitudes

3.6.4 Peak Linear Heat Rate Protection

The axial power distribution (APD) trip is provided to ensure that excessive axial peaking will not cause fuel damage. The APD trip performs the following two functions:

- It provides a reactor trip before the peak kW/ft exceeds the power to fuel centerline melt value of the fuel (22 kW/ft), by working in combination with the variable high power trip, rod block system and the LCO's shown in Figure 3.6-4.

- It provides a reactor trip before the axial power distribution becomes more severe than that assumed to exist by the thermal margin/low pressure trip.

The maximum radial power peak that can occur for power levels up to the variable high power trip limit (in the event of a design basis AOO) is factored into the axial power distribution LSSS. The radial power peak that is allowed at any steady-state or transient core power is specified through the Power Dependent Insertion Limit (PDIL). The rod block system assures that no single electrical component failure in the control element drive system (other than a dropped CEA) can result in CEA group insertion in violation of the PDIL. The rod block system also controls CEA group sequencing, deviation and overlap. A penalty is factored into the LSSS to allow the existence of a 3% azimuthal tilt in core power.

3.6.5 Thermal Margin Analysis

The basic objective of thermal margin analysis is to identify the combinations of steady state operating conditions which satisfy the Specified Acceptable Fuel Design Limit on minimum DNBR. To meet this objective, calculations are performed over a range of operating parameters to determine the power levels that would be required to reach the DNBR design limit.

These calculated powers to DNB or overpower margins are represented in curves which are used to assess and quantify available thermal margin. These curves, which constitute the Thermal Margin information, provide the upper limits on core power over the specified range of operating conditions. This information is used to establish the DNB related Limiting Safety System Settings (LSSS) (Core Operating Limits Report (COLR) Figure 1) and Limiting Conditions for Operation (LCO) (COLR Figure 5).

3.6.5.1 Engineering Factors

Engineering Heat Flux Factor

The effect on local heat flux of deviations from nominal design dimensions and specifications is accounted for by the engineering heat flux factor. Design variables that contribute to this factor are fuel density, fuel enrichment, pellet diameter, and clad outside diameter. These variables may be combined statistically to obtain the engineering heat flux factor. A design value of 1.03 (Ref. 3-83) is used for the engineering heat flux factor. This factor has been incorporated into the statistical combination of uncertainties (Ref. 3-55).

Engineering Enthalpy Rise Factor

The engineering enthalpy rise factor accounts for the effects of deviations in fuel fabrication from nominal dimensions or specifications on the enthalpy rise in the hot channel. Tolerance deviations (averaged over the length of the four fuel rods that enclose the hot channel) for fuel density, fuel enrichment, pellet diameter, and clad outside diameter, contribute to this factor.

The engineering enthalpy rise factor accounts for increased heat input resulting from higher-than-nominal U-235 content. Because of the difficulty in evaluating average pellet tolerance variations for groups of four fuel rods, the enthalpy rise factor is conservatively assumed to be equal to the engineering heat flux factor.

This factor has been incorporated into the statistical combination of uncertainties program (Ref. 3-55).

Fuel Rod Bowing Effects

Fuel rod bowing effects on DNB margin for Fort Calhoun Unit 1 have been evaluated with the guidelines set forth in Reference 3-64. The Cycle 19 core is comprised of both ABB/CE and Westinghouse fuel. The rod bow DNBR effects applied in the OPPD safety analysis were found to be bound by the new fuel design (Ref. 3-71).

A fuel rod penalty was applied to the statistically derived limit. This factor would account for any adverse impacts on DNBR due to fuel rod bowing (Ref. 3-55).

3.6.6 Departure From Nucleate Boiling

The margin to departure from nucleate boiling (DNB) at any point in the core is expressed in terms of the departure from nucleate boiling ratio (DNBR). The DNBR is defined as the ratio of the heat flux required to produce departure from nucleate boiling at specific local coolant conditions to the actual local heat flux. At some point in the core the DNBR is a minimum and it is at this point that the margin to DNB for the core is evaluated.

Because of the uncertainties associated with predicting DNB there is a finite probability that if a channel is operated at a specified DNB ratio greater than one based on a particular correlation, it will be at or above its DNB heat flux. Therefore, the proper interpretation of DNB ratio is that it is a measure of the probability that DNB would occur in the particular design situation to which the DNB correlation is applied. It is customary to establish the relationship between DNB ratio and probability of DNB by statistically evaluating the scatter between actual values of DNB heat flux, as measured experimentally for many test geometries and operating conditions, and the corresponding values that are predicted by the correlation. Uncertainties associated with prediction of the operating conditions in the channel are subject to separate statistical interpretation. The approach used in design is to select core operating conditions and analytical methods in such a way that there is a very small probability that the actual hot channel coolant conditions are more severe than the calculated conditions used as input to the DNB correlation.

The CE-1 DNB correlation presented in References 3-53 and 3-54 is used for the Fort Calhoun design. The correlation limit of 1.15 (Ref. 3-84) has been changed to 1.18 to reflect the Statistical Combination of Uncertainties (Ref. 3-55) effort. This effort was completed prior to the Cycle 9 analysis. A supplement to this analysis was issued in conjunction with the Cycle 10 analysis. The analysis was performed again for Cycle 19 and no change was required to the CE-1 correlation limit of 1.18. This analysis statistically combined the uncertainties associated with the generation of linear heat rate and thermal margin LSSS's and LCO's.

3.6.7 Vapor Fraction

The high operating pressure of the reactor minimizes vapor formation. A calculation, assuming 2 percent overpower, maximum inlet temperature and design coolant flow rate, shows the core vapor fraction is less than 0.1 percent. A conservative value of 0.25 percent is assumed in assessing the effect of voids on reactivity (Section 3.4.1.3).

To avoid the possibility of departure from nucleate boiling as the result of local flow oscillations, a conservative limit has been established to prevent flow instabilities. The limits to assure stable flow are based on avoiding flow regime changes in the hot channel that could affect the flow-pressure drop characteristics so as to cause an instability. Figure 3.6-3 shows flow regimes and regions of stable flow as a function of local mass flow rate and void fraction based on the data in Reference 3-25.

An automatic reactor shutdown (thermal margin trip) will occur before the flow instability limit is reached; thus departure from nucleate boiling resulting from flow oscillations is prevented.

3.6.8 Thermal and Hydraulic Evaluation

Steady state DNBR analyses were performed using the TORC computer code (Reference 3-56), the CE-1 critical heat flux correlation (Reference 3-54 and 3-55), and the CETOP modeling and tuning methodology (Reference 3-57). CETOP was used to develop the thermal margin model used for the DNBR analysis (Reference 3-57). Additional discussions of CETOP methodology are contained in the Arkansas Nuclear One Unit 2 (ANO-2) docket in Reference 3-58 and on the Calvert Cliffs docket in Reference 3-59 and Reference 3-63. CETOP was approved for use on ANO-2 and on Calvert Cliffs (References 3-60 and 3-61). CETOP differs from TORC in that enthalpy transport coefficients are used to improve modeling of coolant conditions in the vicinity of the hot subchannel and in that more rapid equation-solving routines are used. CETOP models are tuned to always give conservative MDNBR results relative to detailed TORC. CETOP is used only because it reduces computer time significantly; no margin gain is realized. TORC was used for the Cycle 19 thermal margin model since it can be run now at low cost (Ref. 3-83).

TORC

The TORC (Reference 3-56) thermal hydraulics computer code solves conservation equations for a 3-dimensional representation of the open lattice core to determine local coolant conditions at all points within the core. The code performs these calculations in three separate stages with the output of the previous stage coupled to the next stage. This type of model permits greater accuracy in predicting localized conditions.

CETOP

The CETOP thermal hydraulics code (Reference 3-57) is utilized in the setpoint analysis. It is benchmarked against TORC to always give conservative results relative to TORC. In other words, CETOP always calculates smaller MDNBR's than does TORC for the same operating conditions.

3.6.9 Fuel Temperature Conditions

Steady state fuel temperatures are determined by the FATES computer program. The calculational procedure considers the effect of linear rate, fuel relocation, fuel swelling, densification, thermal expansion, fission gas release and clad deformations. The model for predicting fuel thermal performance is discussed in detail in Reference 3-65.

Two sets of burnup and axially dependent linear heat rate distributions are considered in the calculation. One is the hot rod, time averaged, distribution expected to persist during long-term operation, and the other is the envelope of the maximum linear heat rate at each axial location. The long-term distributions are integrated over selected time periods to determine burnup, which is in turn used for the various burnup dependent behavioral models in the FATES computer program. The envelope accounts for possible variations in the peak linear heat rate at any elevation which may occur for short periods of time and is used exclusively for fission gas release calculations.

The power history used assumes continuous 100 percent reactor power operation from beginning-of-life. Using this history, the highest fuel temperatures occur at beginning of life. It has been shown that fuel temperatures for a given power level at any burnup are insensitive to the previous history used to arrive at the given power level.

Fuel thermal performance parameters are calculated for the hot rod. These parameters for any other rod in the core can be obtained by using the axial location in the hot rod, whose local power and burnup corresponds to the local power and burnup in the rod being examined. This procedure will yield conservatively high stored energy in the fuel rod under consideration.

3.6.10 Flow Stability

Flow oscillations of significant amplitude may be sustained in some channels when heat is added to two-phase flow in parallel channels. This possibility results from two conditions that exist within the core:

a. The pressure drop flow characteristics with two-phase flow are such that large changes in flow can occur for small changes in pressure drop;

b. With parallel channels, the flow has an alternate path.

The flow regimes may be classed as separated or homogeneous. Homogeneous flow is bubbly or froth flow. Separate flow is annular or slug. Reference 3-30 describes these flow regimes in detail. For homogeneous flow, the channel pressure drop continuously increases with increasing flow rate or increasing vapor fraction. A change in the flow regime to separated flow results in a change in the flow characteristics and flow oscillations in the parallel channels are then possible. Figure 3.6-3 shows flow regimes as a function of mass flow rate and void fraction based on the data of Reference 3-25. In general, increasing void fraction results in a transition to an annular type flow and decreasing void fraction results in a transition to slug flow. A comparison of this flow regime map with data of observed flow regimes reported in References 3-26, 3-27, 3-28 and 3-29 has been made to verify the effect of variation of such parameters as channel length, diameter and pressure and to check the consistency of the data. Good agreement was obtained, and the limits for stable flow as shown in Figure 3.6-3 are considered a reasonable and conservative representation of flow regime changes for core hot channel conditions.

The limit to ensure flow stability as applied to core conditions is conservative since the "openness" of the channels to crossflow tends to damp any flow oscillations. This is explained by the basic requirement that in order for flow oscillations to occur, a feedback effect (from the channel outlet to the channel inlet region) on channel flow and pressure loss is necessary. Crossflow tends to damp the feedback effect and tends to make the open channel array stable even when parallel closed channels would not be stable. This conclusion is supported by the observation of the absence of DNB conditions in the open array experiments reported in Reference 3-30. The experimental results of Reference 3-25 to 3-29 are all for closed channels.

3.7 MECHANICAL DESIGN AND EVALUATION

The reactor core and internals are shown in Figure 3.1-1. A cross section of the reactor core and internals is shown in Figure 3.1-2. Mechanical design features of the reactor internals, the control element drive mechanisms and the reactor core are described below. Mechanical design parameters are listed in Table 3.7-1.

Table 3.7-1 - "Mechanical Design Parameters"

Fuel Assemblies

Type	Number of Assemblies	Number of Fuel Rods per Assembly	Number of Fuel Displacing Poison Rods per Assembly	IFBA Rods per Assembly	Spacers Per Assembly	Weight of Contained Uranium per Assembly,kg	Spacer Material
M/	5	168	8	N/A	9	349	Zircaloy-4
U1	4	176	N/A	0	9	376	Zircaloy-4 & Inconel
U2	8	176	N/A	0	9	375	Zircaloy-4 & Inconel
U3	4	176	N/A	20	9	376	Zircaloy-4 & Inconel
U4	8	176	N/A	40	9	376	Zircaloy-4 & Inconel
U5	8	176	N/A	48	9	375	Zircaloy-4 & Inconel
U6	8	176	N/A	56	9	375	Zircaloy-4 & Inconel
U7	4	176	N/A	64	9	375	Zircaloy-4 & Inconel
W1	12	176	N/A	0	9	374	ZIRLO & Inconel
W2	8	176	N/A	48	9	371	ZIRLO & Inconel
W3	16	176	N/A	56	9	372	ZIRLO & Inconel
W4	4	176	N/A	84	9	370	ZIRLO & Inconel
W5	4	176	N/A	64	9	371	ZIRLO & Inconel
X1	4	176	N/A	84	9	370	Inconel
X2	8	176	N/A	0	9	374	Inconel
X3	8	176	N/A	48	9	372	Inconel
X4	4	176	N/A	56	9	371	Inconel
X5	4	176	N/A	64	9	371	Inconel
X6	12	176	N/A	84	9	370	Inconel

Fuel Rod Array, square 14 x 14

Table 3.7-1 (Cont'd)

Fuel Rod	M/	U1 - U7, W1 - W5, X1 - X6
Fuel Material (Sintered Pellets)	UO_2	UO_2
Pellet Diameter, inches	0.3765	0.3765
Pellet Length, inches	0.450	0.452*
Pellet Density, g/cc	10.38	
Clad Material	Zircaloy-4	ZIRLO
Clad ID, inches	0.384	0.384
Clad OD, inches (nominal)	0.440	0.440
Clad Thickness, inches (nominal)	0.028	0.028
Clad Thickness, inches (minimum)	0.026	
Diametral Gap, inches	9.0075	
Active Length, inches	128	128
Total Length Between End Plates, inches	138.28	137.08 (137.10 for W and X fuel)
Maximum Fuel Rod Average Burnup (MWD/MTU)	60,000	60,000

* Axial Blanket Pellets on Batches W and X Fuel are 0.545 inches in length.

Table 3.7-1 (Cont'd)

Control Element Assemblies (CEA's)

Type	Original Design*	Revised Design	Improved Design
CEA Description	Five(5) non-reconstitutable Full-Length Poison Rods Attached to a Spider	Five(5) non-reconstitutable Full-Length Poison Rods Attached to a Spider	Five(5) reconstitutable Full Length Poison Rods Attached to a Spider
Poison Material	B_4C Pellets	B_4C Pellets Ag-In-Cd Tips	B_4C Pellets Ag-In-Cd Tips
Sheath Material	Inconel 625	Inconel 625	Inconel 625
Number of CEAs in the Core	0	8	41
Poison Length	125"	125"	125"
Overall Length	148"	148"	152"
Corner Element Pitch	4.64"	4.64"	4.64"
Pre-Irradiated CEA Rod Diameter	0.95"	0.95"	0.95"
CEA Weight	75± 1 lbs.	75± 1 lbs.	75± 1 lbs.
Total Operating CEA weight	260 lbs.	260 lbs.	260 lbs.

* All original design CEAs discharged. Data for information only.

Table 3.7-1 (Cont'd)

Core Arrangement

Number of Fuel Assemblies in Core, Total	133
Number of CEAs	49
Number of Active Fuel Rods (Design)	23,368*
CEA Pitch, min, inches	11.57
Spacing Between Fuel Assemblies, Fuel Rod Surface to Surface, inches	0.198
Spacing, Outer Fuel Rod Surface to Core Shroud, inches	0.179
Hydraulic Diameter, Nominal Channel, Feet	0.04444
Total Flow Area (Excluding Guide tubes), sq ft	32.84
Total Core Area, sq ft	62
Core Equivalent Diameter, inches	106.448
Core Circumscribed Diameter, inches	116.484
Core Volume, liters	18,726
Total Heat Transfer Area, sq ft	28,713*

(*Excludes the 40 Fuel Displacing shims in 5 of the CE Fuel Assemblies. The design provides an allowance for replacement of 10 or less fuel rods with stainless steel rods.)

3.7.1 Reactor Internals

The reactor internals are designed to support and orient the reactor core fuel assemblies and control element assemblies, absorb the CEA dynamic loads and transmit these and other loads to the reactor vessel flange, provide a passageway for the reactor coolant, and support in-core instrumentation.

The internals are designed to safely perform their functions during all steady state conditions and during normal operating transients. The internals are designed to safely withstand the forces due to deadweight, handling, system pressure, flow impingement, temperature differential, shock and vibration. All reactor components are considered Class 1 for seismic design. The reactor internals design limits deflection where required by function. The structural components satisfy stress values given in Section III of the ASME Boiler and Pressure Vessel Code. Certain components have been subjected to a fatigue analysis. Where appropriate, the effect of neutron irradiation on the materials concerned is included in the design evaluation.

The components of the reactor internals are divided into three major parts consisting of the core support barrel (including the lower core support structure, the core shroud and the thermal shield), the upper guide structure (including the CEA shrouds and the in-core instrumentation guide tubes) and the flow skirt. These components are shown in Figure 3.1-1. The in-core instrumentation is described in Sections 7.5.1 and 7.5.2.

3.7.1.1 Core Support Assembly

The major support member of the reactor internals is the core support assembly. This assembled structure consists of the core support barrel, the core support plate and the support columns, the core shroud, the thermal shield, the core support barrel to pressure vessel snubbers and the core support barrel to upper guide structure guide pins. The major material for the assembly is Type 304 stainless steel.

The core support assembly is supported at its upper flange from a ledge in the reactor vessel flange. The lower end is restrained in its lateral movement by six core support barrel-to-pressure vessel snubbers. Within the core support barrel are axial shroud plates which are attached to the core support barrel wall by horizontal former plates and to the core support plate by anchor blocks. The core support plate is positioned within the barrel at the lower end and is supported both by a ledge in the core support barrel and by 44 columns. The core support plate provides support and orientation for the fuel assemblies. Also within the core support barrel just below the nozzles are four guide pins which align and prevent excessive motion of the lower end of the guide structure relative to the core support barrel during operation. The thermal shield is affixed to the outside of the core support barrel.

The effect of neutron irradiation on the core support structure will be a reduction in the ductility of the structures in the areas of highest fluence. In the design of the structures in these areas, the deflections and resultant strains were determined and compared with the estimated ductility values at end-of-service life in order to ensure adequacy of the design. Table 3.7-2 shows this comparison for certain areas in the core support structure which experience high fluence.

Table 3.7-2 - "Comparison of Areas in the Core Support Structure Which Experience the Highest Fluence With Ductility at End of Life"

COMPONENT	LOCATION	FLUENCE Cm^2	CALCULATED STRAIN %	UNIFORM ELONGATION % AT OPERAT. TEMP.
Core Barrel	Opposite Center of Core	1.30×10^{21}	.01	2.5
Core Support Columns	Top of Column	7.50×10^{21}	.04	0.5
Core Barrel	Upper Flange	$< 10^{20}$	.08	> 23%
Upper Guide Structure	Grid Beams	$< 10^{20}$	.08	> 23%
Lower Support Beams	Top of Beam	$< 10^{20}$	.08	> 23%

3.7.1.2 Core Support Barrel

The core support barrel net weight (325,000 pounds) consists of the entire core and other internals. It is a right circular cylinder with a nominal inside diameter of 120-5/8 inches and a minimum wall thickness in the weld preparation area of 1 inch. It is suspended by a 4-inch thick flange from a ledge on the pressure vessel. The core support barrel in turn supports the core support plate upon which the fuel assemblies rest. Press fitted into the flange of the core support barrel are four alignment keys located 90 degrees apart. The reactor vessel, closure head and upper guide structure assembly flanges are slotted in locations corresponding to the alignment key locations to provide proper alignment between these components in the vessel flange region.

Since the core support barrel is 26 feet long and is supported only at its upper end, it is possible that coolant flow could induce vibrations in the structure. Therefore, amplitude limiting devices, or snubbers, are installed near the bottom outside end of the core support barrel. The snubbers consist of six equally spaced double lugs around the circumference and are the grooves of the "tongue-and-groove" assembly; the pressure vessel lugs are the tongues. Minimizing the clearance between the two mating pieces limits the amplitude of any vibration. At assembly, as the internals are lowered into the vessel, the pressure vessel tongues engage the core support grooves in an axial direction. With this design, the internals may be viewed as a beam with supports at the furthest extremities. Radial and axial expansions of the core support barrel are accommodated, but lateral movement of the core support barrel is restricted by this design. The pressure vessel tongues have bolted, lock welded Inconel X shims and the core support barrel grooves are hardfaced with Stellite to minimize wear.

3.7.1.3 Core Support Plate and Support Columns

The core support plate is a 120-inch diameter, 2-inch thick, Type 304 stainless steel plate into which the necessary flow distributor holes for the fuel assemblies have been machined. Fuel assembly locating holes (four for each assembly) are also machined into this plate.

Columns and support beams are placed between this plate and the bottom of the core support barrel in order to provide stiffness to this plate and transmit the core load to the bottom of the core support barrel.

3.7.1.4 Thermal Shield

The 3-inch thick, Type 304 stainless steel thermal shield is a cylindrical structure which reduces the neutron flux and radiation heating in the reactor vessel wall to an acceptable level. At the upper end, the shield is supported by eight equally spaced lugs on the outer periphery of the core support barrel. A 0.005-inch gap between the thermal shield and the lower portion of the lug is provided to permit assembly of the core support barrel and the thermal shield. The lower end of the thermal shield is positioned radially utilizing 16 equally placed positioning pins which pass through the shield and butt against the core support barrel.

3.7.1.5 Core Shroud Plates and Centering Plates

The core shroud provides an envelope for the perimeter of the core and limits the amounts of coolant bypass flow. The shroud consists of rectangular plates 5/8-inch thick, 142-3/8 inches long and of varying widths. The bottom edges of these plates are fastened to the core support plate by use of anchor blocks.

The critical gap between the outside of the peripheral fuel assemblies and the shroud plates is maintained by eight tiers of centering plates attached to the shroud plates and centered during initial assembly by adjusting bushings located in the core support barrel. The overall core shroud assembly, including the rectangular plates, the centering plates, and the anchor blocks, is a bolted and lock welded assembly. In locations where mechanical connections are used, bolts and pins are designed with respect to shear, binding and bearing stresses. All bolts and pins are lock welded. In addition, all bolts (bodies and heads) are designed to be captured in the event of fracture; the bolt heads are trapped by lock bars or lock welds, and the bodies are trapped by the use of non-thru holes or by incomplete tapping of thru holes. Holes are provided in the core support plate to allow coolant to flow upward between the core shroud and the core support barrel, thereby minimizing thermal stresses in the shroud plates and eliminating stagnant pockets.

3.7.1.6 Flow Skirt

The Inconel flow skirt is a perforated (2-1/4-in. diameter holes) right circular cylinder, reinforced at the top and bottom with stiffening rings. The flow skirt is used to reduce inequalities in core inlet flow distributions and to prevent formation of large vortices in the lower plenum. The skirt provides a nearly equalized pressure distribution across the bottom of the core support barrel. The skirt is fastened to the pressure vessel lower head by nine equally spaced welds.

3.7.1.7 Upper Guide Structure Assembly

This assembly (Figure 3.7-1) consists of a plate, 41 control element assembly shrouds (two of which have been modified to act as Heated Junction Thermocouple Probe holders), a fuel assembly alignment plate and a ring shim. The upper guide structure aligns and laterally supports the upper end of the fuel assemblies, maintains the CEA spacing, prevents fuel assemblies from being lifted out of position during a severe accident condition and protects the CEA's from the effect of coolant crossflow in the upper plenum. It also supports the in-core instrumentation guide tubing. The upper guide structure is handled as one unit during installation and refueling.

The upper end of the assembly is a flanged grid structure consisting of a grid array of 24-inch deep beams. The grid is encircled by a 24-inch deep cylinder with a 3-inch thick plate welded to the cylinder. The periphery of the plate contains four accurately machined and located alignment keyways, equally spaced at 90-degree intervals, which engage the core barrel alignment keys. The reactor vessel closure head flange is slotted to engage the upper ends of the alignment keys in the core barrel. This system of keys and slots provides an accurate means of aligning the core with the closure head. The grid aligns and supports the upper end of the CEA shrouds.

The control element assembly shrouds extend from the fuel assembly alignment plate to an elevation about 8 inches above the support plate. There are 29 single-type shrouds. These consist of centrifugally cast cylindrical upper sections welded to cast bottom sections, which are shaped to provide flow passages for the coolant passing through the alignment plate while shrouding the CEA's from crossflow. There are also 12 dual-type shrouds which in configuration consist of two single-type shrouds connected by a rectangular section shaped to accommodate the dual control element assemblies. The shrouds are bolted to the fuel assembly alignment plate. At the upper guide structure support plate, the single shrouds are connected to the plate by spanner nuts which permit axial adjustment. The spanner nuts are torqued in place and lockwelded. The dual shrouds are attached to the upper plate by welding.

The fuel assembly alignment plate is designed to align the upper ends of the fuel assemblies and to support and align the lower ends of the CEA shrouds. Precision machined and located holes in the fuel assembly alignment plate align the fuel assemblies. The fuel assembly alignment plate also has four equally spaced slots on its outer edge which engage with Stellite hardfaced pins protruding out from the core support barrel to prevent lateral motion of the upper guide structure assembly during operation. Since the weight of a fuel assembly under all normal operating conditions is greater than the flow lifting force, it is not necessary for the upper guide structure assembly to hold down the core. However, the assembly would capture the core and limit upward movement in the event of an accident.

A ring shim bears on the flange at the top of the assembly to resist axial upward movement of the upper guide structure assembly and to accommodate axial differential thermal expansions between the core barrel flange, upperguide structure flange and pressure vessel flange support edge and head flange recess.

The upper guide structure assembly also supports the in-core instrument guide tubes. The tubes are conduits which protect the in-core instruments and guide them during removal and insertion operations.

3.7.2 Control Element Drive Mechanism

The control element drive mechanism (CEDM) drives the CEA within the reactor core and indicates the position of the CEA with respect to the core. The speed at which the CEA is inserted or withdrawn from the core is consistent with the reactivity change requirements during reactor operation. For conditions that require a rapid shutdown of the reactor, the CEDM drive releases to allow the CEA and the supporting CEDM components to drop into the core by gravity. The reactivity is reduced during such a drop at a rate sufficient to control the core under any operating transient or accident condition. Since CEA speed is a direct function of drive motor power supply frequency, which is limited by the transmission frequency control system to 60 cycles/sec, CEA speed limiting features are not needed on Fort Calhoun and none are included as such.

The CEA is decelerated at the end of the drop by the CEDM which supports the CEA in the fully inserted position.

There are 37 CEDM's mounted on flanged nozzles on top of the reactor vessel closure head, located directly over the CEA's in the reactor core. Each CEDM is connected to a CEA by a locked coupling. The weight of the CEA's and CEDM's is carried by the vessel head. In order to provide lateral stability, particularly in resisting horizontal earthquake forces, the CEDM's are supported in the horizontal direction by a seismic support structure which is a cylindrical structure surrounding the CEDM's and attached to the reactor vessel head. This structure restricts bending deflection so as to limit stresses to allowable values in the lower housing and nozzle areas. Air is drawn through the structure for cooling (see Section 9.10).

The CEDM is designed to handle dual or single CEA's. The total stroke of the drive is 128 inches. The speed of the drive is 46 inches per minute. The time from receiving a trip signal to 90 percent of the fully inserted position of the CEA is less than 2-1/2 seconds under operating conditions. The CEA is allowed to accelerate to about 11 ft/sec and is decelerated to a stop at the end of the stroke.

The CEDM is of the vertical rack and pinion type with the drive shaft running parallel to the rack and driving the pinion gear through a set of bevel gears. The design of the drive is shown in Figure 3.7-2: The trippable CEA is driven by an electric motor operating through a gear reducer and a magnetic clutch. By de-energizing the magnetic clutch, the CEA drops into the reactor under the influence of gravity. The magnetic clutch incorporates an anti-reversing device which prevents upward CEA movement when the clutch is deenergized. The non-trippable CEA is driven by a CEDM that has been modified by replacing the magnetic clutch with a solid shaft assembly to eliminate the trip function so that the non-trippable CEA maintains its position during a reactor trip. Otherwise, this CEDM is the same as those attached to the other CEA's. The drive shaft penetration through the pressure housing is closed by means of a face-type rotating seal. The rack is connected to the CEA by means of a rack extension containing an external collet-type coupling which expands and locks into a mating shouldered bore on top of the CEA.

The rack extension is connected to the rack through a tie bolt by means of a nut and locking device at the upper end of the rack. A small diameter closure located at the top of the pressure housing provides tool access to this nut for releasing the CEA from the CEDM. The rack is guided at its upper end by a section having an enlarged diameter which operates in a tube extending the full length of the CEA travel. The final cushioning at the end of a CEA drop is provided by the dashpot action of the enlarged diameter of the rack entering a reduced diameter in the guide tube.

3.7.2.1 CEDM Pressure Housing

The pressure housing consists of a lower and an upper section joined near the top of the drive by means of a threaded autoclave type closure. The pressure housing design and fabrication conforms to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels. The housing is designed for steady state conditions as well as all anticipated pressure and thermal transients.

The lower housing section is a stainless steel tubular section welded to an eccentric reducer and flange piece at the lower end. This flange fits the nozzle flange provided on the reactor vessel closure head and is seal welded to it by an omega-type seal. Once seal welded and bolted into place, the lower pressure housing need not be removed since all servicing of the drive is performed from the top of this housing. The upper part of the lower housing is machined to form the autoclave-type closure and is provided with a recessed gasket surface for a gasket.

The upper part of the pressure housing has a flange which mates with the lower housing autoclave-type closure, a cavity which contains the drive rotating seal, and a tubular housing extension with a small flange closure which provides access for attaching and detaching the CEA. The shaft seal is a face-type rotating seal with mating surfaces of Graphitar and tungsten carbide. The two parts of the seal are fitted with O-rings to prevent leakage around the seal. The O-rings are static seals. A cooling jacket surrounds the seal area to maintain the temperature of the seal and O-rings at an acceptable level. This cooling water is from the component cooling system (see Section 9.7). A seal leak-off line is connected to the upper housing. A thermocouple in the seal leak-off connection at the upper housing monitors leak-off water temperature which provides an indication of seal leakage.

There are four spare CEDM penetrations in the reactor vessel head. Two have spare CEA pressure housings attached. These are used to house the Heated Junction Thermocouples. Each HJTC assembly consists of a Graylock adapter hub with a threaded autoclave-type closure that conforms to the requirement of the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels. Each adapter hub utilizes a seal plug to form a pressure boundary about each Heated Junction Thermocouple assembly. The other two spare penetrations are blind flanged.

3.7.2.2 Rack and Pinion Assembly

The rack and pinion assembly is an integrated unit which fits into the lower pressure housing and couples to the motor drive package through the upper pressure housing. This unit carries the bevel gears which transmit torque from the vertical drive shaft to the pinion gear. The vertical drive shaft has couplings at both ends and may be lifted out when the upper pressure housing is removed. Ball bearings are provided for supporting the bevel gears and the pinion gear. The rack engages the pinion, and is held in proper engagement with the pinion by the backup rollers which carry the load due to gear tooth reactions. The gear assembly is attached to a stainless steel tube supported by the upper part of the pressure housing. This tube also carries and positions the guide tube which surrounds the rack. The rack is a tube with gear teeth on one side of its outer surface and a flat on the opposite side which forms a contact surface for guide rollers. The upper end of the rack is fitted with an enlarged section which runs in the guide tube and provides lateral support for the upper end of the rack. It also acts as a piston in controlling water flow in the lower guide tube dashpot. The top section also carries a permanent magnet which is used to operate a reed switch position indicator outside the pressure housing. The guide tube is connected at its upper end to the support tube. The support for the guide tube contains an energy absorber at the top end of the tube which deforms to limit the stresses on the CEA, in case the mechanism is tripped without water in the dashpot. If such a "dry trip" should occur, the mechanism and CEA would not be damaged; however, it would be necessary to disassemble the CEDM and replace the energy absorber.

3.7.2.3 Motor Drive Package

Power to operate the drive is supplied by a fractional horsepower, 120-V, single-phase, 60-Hz motor. The output is coupled to the vertical drive shaft through a magnetic clutch and an anti-reverse clutch operating in parallel. When the magnetic clutch is energized, the drive motor is connected to the main shaft and can drive the CEA either up or down. When de-energized, the magnetic clutch separates and the CEA drops due to its own weight. The anti-reverse clutch prevents rotation of the drive in the up direction and holds the CEA inposition against upward forces. The action is completely mechanical and does not rely on any outside source of power. The motor, brake, clutches, position indicator and limit switches are all mounted on a common frame for maintaining position and alignment. This entire drive package is assembled and checked as a unit and can be removed and replaced without disturbing the other parts of the mechanism. The frame for the drive package is provided with a flange which is bolted to a flange on the upper pressure housing for positioning the drive assembly. The electrical connections are located on the top of the drive package and are readily accessible.

3.7.2.4 Position Readout Equipment

Two independent position readout systems are provided for indicating the position of the CEA. One (primary system) is a synchro transmitter geared to the main drive shaft with readout provided by synchro receivers connected to the transmitter. The other (secondary system) position indicator consists of a series of reed switches built into a subassembly which is fastened to the outside of the CEDM along the pressure housing. The permanent magnet built into the top of the rack actuates the reed switches one at a time as it passes by them. A resistor network in conjunction with these switches controls the readout to indicate position. Limit switches located in the motor drive package are geared to the drive shaft and are used to provide indication of CEA position at certain predetermined points. These switches are used in the CEDM control system. The systems are described in Section 7.5.

3.7.2.5 Control Element Assembly Disconnect

The CEA is connected to the CEDM by means of an extension shaft with an internal collet-type coupling at its lower end. A tie rod connects the extension shaft to the rack. In order to disengage the CEA from the drive, the flange access closure at the top of the CEDM is removed. A tool is then inserted through this opening and, with the CEA in the full down position, the tool is used to release the nut locking device and to unscrew the nut on top of the tie rod. By turning another handle on the tool, the tie rod is rotated about a quarter turn and lifted about 2 inches to unlock the collet coupling and disengage the CEDM from the CEA.

3.7.2.6 CEDM Evaluation

Development models of internal and external drive components, subassemblies of the CEDM, as well as a complete model CEDM have undergone accelerated life tests under reactor conditions and have demonstrated that the CEDM fulfills all drive, trip and endurance requirements (see Section 1.4.4).

In addition to these development tests, a prototype CEDM with a simulated reactor core module was accelerated-life tested in an autoclave under reactor conditions to provide the overall adequacy of the CEDM during its design life. Each CEDM was tested at design pressure to prove its functional adequacy.

3.7.3 Core Mechanical Design

The core approximates a right circular cylinder with an equivalent diameter of 106.4 inches and an active fuel height of 128 inches. It is made up of Zircaloy-4/ZIRLO clad fuel rods containing approximately 49 metric tons of slightly enriched uranium in the form of sintered $U0_2$ pellets. The fuel rods are grouped into 133 assemblies. Information in the following sections is based on Westinghouse (W) fuel and CE fuel. The data for CE fuel is shown in parantheses. Failed fuel rods are replaced with stainless steel rods per engineering analysis justified in nuclear engineering instructions.

Short term reactivity control is provided by 49 control element assemblies. Four of the CEA's are non-trippable. The CEA's are guided within the core by the guide tubes which are integral parts of the fuel assemblies.

3.7.3.1 Fuel Assembly

For Cycle 19, the 133 assemblies consist of 176 fuel rods in the 128 Westinghouse assemblies and 168 fuel rods in the five ABB/CE assemblies, five guide tubes, nine fuel spacer grids, and upper and lower end fittings. The structural frame of the assembly consists of the guide tubes, spacer grids and end fittings. All five guide tubes are mechanically attached to the stainless steel end fittings. The spacer grids contained in the assemblies are mechanically attached to the guide tubes. The spacer grids for the CE fuel are welded to the guide tubes.

The lower end fitting is a cast structure of 304 stainless steel. It is machined to accept alignment pins which fit in corresponding holes in the core support plate. The alignment pins provide lateral alignment of the lower end of the fuel assembly. The length of the alignment pin engagement ensures that the spacing between fuel assemblies will not be altered even during postulated accident conditions when a fuel assembly is lifted into contact with the upper guide structure. The lower end fitting contains flow holes and holes for positioning the fuel rods and guide tubes. The flow holes on the Westinghouse flowplate are of small diameter to limit the size of debris particles flowing into the fuel array. This reduces the potential for debris related fuel clad failures.

The fuel assembly upper end fitting is a cast structure of 304 stainless steel. It serves as an attachment for the guide tubes and as the lifting fixture. The pin-shaped protrusions serve as guide pins and mate with precision-drilled holes in the alignment plate to provide the alignment of the upper ends of the fuel assembly.

The fuel rod spacer grids maintain the fuel rod pitch over the full length of the fuel rods. The grids in the batch X assemblies are fabricated from Inconel strips interlocked in an egg crate fashion and welded together. The top and bottom grids in the Westinghouse batches U and W assemblies are fabricated from Inconel while the middle grids are fabricated from Zircaloy/Zirlo. The grids in the Batch M/ assemblies are fabricated from Zircaloy. Each batch U, W, and X fuel assembly rod is supported by two (2) springs and two (2) support dimples. Each batch M assembly rod is supported by two leaf springs and two arches which are opposite the leaf-springs. The springs press the rod against the arches to restrict relative motion between the grids and the fuel rods. The spring and arch positions are reversed from grid to grid to provide additional restriction to relative motion. The perimeter strips also contain springs and arches in addition to special features to prevent hang up of grids during a refueling operation. Spacer grids of this type have been hot-flow tested under a variety of conditions of velocity ranging from 15 to 25 feet per second, crossflow, and with simulated fully relaxed springs; no significant fretting has been detected. Analysis of loss coefficients for both Westinghouse and CE type grids have been performed, (Reference 3-62).

3.7.3.2 Fuel Rods

The fuel rods consist of $U0_2$ pellets, a compression spring and spacer discs, all encapsulated within a Zircaloy-4 tube. The $U0_2$ pellets have a density of 10.41 (10.38) g/cc and are dished at both ends. Dishing is used to accommodate the effects of thermal expansion and swelling and results in a pellet column density of approximately 10.29 (10.05) g/cc.

The fuel cladding is slightly cold worked Zircaloy-4/ZIRLO tubing. The cold nominal diametral gap between the pellet and clad ID is 0.0075 (0.0065) inches, and has been set taking into account clad stresses and strains and transfer of heat from the pellets. The compression spring located at the top of the fuel pellet column is of 302 stainless steel (X-750 Inconel) and maintains the column in its proper position during handling and shipping. It also provides support for the clad in the plenum region to prevent local buckling. The adequacy of the spring to perform its functions has been demonstrated in a series of long term creep buckling tests with plenum clad temperatures above those expected in the reactor.

There are two alumina spacers in each fuel rod for CE fuel and one spacer at the top of the pellet stack in Westinghouse fuel rods. The spacers are located at either end of the fuel pellet stack. The lower spacer reduces the lower end cap temperature and prevents local boiling in the spiral pin area. The upper spacer prevents $U0_2$ chips from entering the plenum region.

The plenum above the pellet column provides space for axial thermal expansion of the fuel column and for expansion of fission gas. The maximum internal pressure at the end of life for hot operation conditions is 2760 psia. Gas release calculations are based on the most adverse fuel temperature and burnup conditions.

Each fuel rod is internally pressurized with helium. The internal pressurization with helium improves the thermal conductance between the fuel pellets and the cladding, resulting in a decrease in fuel temperature with an attendant reduction in the release of fission products and an increase in the margins between operating temperatures and allowable thermal limits. In addition, by reducing the differential pressure across the clad, internal pressurization affords a substantial reduction in the adverse effects of fuel-clad interaction and ensures that the fuel clad will be free standing against the possibility of collapsing under the effects of long-term creep due to differential pressure. The initial helium pressure is sufficient to prevent cladding collapse at the peak power location for the full design life of the fuel rod as discussed in Appendix L.

All of the Westinghouse fuel assemblies residing in the current core contain axial blankets. Fuel rods containing axial blankets have a reduced U-235 enrichment in the top and bottom ends (5.995 inches for each end) of each fuel rod. Since normal power production in the fuel rod axial ends is significantly lower than in the central portion of a fuel rod, the reduced enrichment does not negatively impact rod power. In addition, axial blankets reduce the total U-235 inventory at discharge. The combination of these effects improves uranium utilization, which translates into direct fuel cycle cost savings.

3.7.3.3 Poison Rods

In some of the fuel assemblies, poison rods are included to make the beginning of cycle moderator coefficient more negative. The poison rods are mechanically similar to fuel rods except that they contain 116 (114) inch long burnable poison pellet columns instead of fuel pellets. For the Westinghouse fuel, the poison pellets are fuel pellets coated with ZrB_2 which is a burnable poison that does not displace fuel from the rod (Integral Fuel Burnable Absorber-IFBA). The pellet diameter is 0.353 inches. The ZIRLO clad thickness for the poison rod is 0.032 inches. Some of the axial blankets in the Westinghouse IFBA fuel assemblies are comprised of annular pellets. The annular pellet design increases the fuel rod plenum volume, thereby reducing the maximum rod internal pressure during operation. The annular axial blankets also provide an economic benefit in reduced fuel costs. For the CE fuel, the poison pellets are alumina with uniformly dispersed boron carbide particles. The pellet diameter is 0.362 inches. The Zircaloy-4 clad thickness for the poison rod is 0.028 inches. The outside diameter of all poison rods is identical to those of the fuel rods.

3.7.3.4 Part Length Poison Rods

The part length poison rods (PLPR), when used, reside in guide tubes of selected peripheral assemblies. Due to space limitations, the rods can not be attached to a spider section. Therefore, each individual rod has a gripper section at the upper end for handling purposes. The gripper section is also spring loaded to provide a holddown function when the upper guide structure is installed. The holddown load is transmitted over the entire length of the rod, causing the tip of the rod to be pressed against the bottom of the guide tube.

The poison rod cladding is fabricated of 0.04 inch wall Inconel 625, 0.948 inches in outer diameter. The central fifty percent of the rod in the core region contains B_4C pellets 0.860 inch in diameter, while a column of Alumina (AL203) pellets, 0.851 inches in length, encloses each end of the B_4C pellet stack. The initial atmosphere within the poison rods is unpressurized helium. Based on an anticipated burn-up of boron and considering swelling of B_4C and a clad strain limit of one percent, the PLPRs have an estimated lifetime of about ten years. The PLPR's were removed at the end of Cycle 10 and are currently stored in the spent fuel pool.

3.7.3.5 Hafnium Flux Suppression Rods

The Hafnium Flux Suppression Rods, when used, reside in guide tubes of selected peripheral assemblies. Due to space limitations, the rods can not be attached to a spider section. Therefore, each individual rod has a gripper section at the upper end for handling purposes. The gripper section is also spring loaded to provide a holddown function when the upper guide structure is installed. The holddown load is transmitted over the entire length of the rod, causing the tip of the rod to be pressed against the bottom of the guide tube.

The poison rods are composed of unclad hafnium 0.948 inches in outer diameter over the active length of the core. The absence of cladding prevents the localized hydriding and swelling of the rods. An oxide layer forms on the unclad hafnium during exposure to the primary coolant and is an effective barrier to hydrogen absorption. This avoids the concern for the localized hydriding phenomenon in this design extending the expected life of the rods to 15 years.

3.7.3.6 Clad Evaluation

The fuel rod cladding is designed to satisfy the design limits given in Section 3.2.3. The effects of irradiation of $U0_2$ and Zircaloy-4/ZIRLO have been considered in the design calculations. The predicted effects of anticipated transients have also been considered in the design process.

The design bases are conservative and the calculations used to demonstrate their compliance are conducted for limiting cases using limiting assumptions. This is considered advisable in the prediction of long term fuel behavior under irradiation.

A series of transverse and torsional deflection and thermal bow tests has been performed on a 12 x 12 fuel assembly to provide experimental support to the analytical effort in defining the structural action of a fuel assembly. The information gained from these tests has been used in the design of fuel assemblies and of lifting fixtures and shipping containers.

These tests show that the fuel assembly is sufficiently flexible to accommodate alignment tolerances, has adequate structural stability for reactor operation and maintains its as-fabricated dimensions during handling. All handling from a horizontal to vertical position was performed using an auxiliary support structure.

Based upon the thermal tests, the maximum thermal bow expected under adverse temperature conditions is 7 mils. This amount of thermal bowing has no significant effect on CEA operation.

Clad stress-strain behavior is based upon a triaxial stress analysis which includes the effect of creep. The loads considered are those due to fuel thermal and fission growth, fission gas pressure and external coolant pressure.

The fuel thermal and fission growth was calculated considering the fuel as a solid rod with unrestrained thermal expansion and a volumetric growth rate of 0.16 percent for 10^{20} fissions/cm^3 (Reference 3-31), an average clad temperature of 688°F and a linear heat rate of 17.6 kW/ft. The fission gas pressure was calculated for a 31.5 percent fission gas release which was derived from the data of Hoffmann and Coplin (Reference 3-32) considering the change in plenum volume due to thermal expansion and growth of the rod.

The analysis is based upon an incremental approach which divides the 3-year fuel life span into discrete time intervals and evaluates the clad stress and strain, including the effect of creep, during these intervals. The relation between the incremental creep and the actual stress state is expressed by the Prandtl-Reuss formula (Reference 3-33). The basis for creep is given by the von Mises criterion, (Reference 3-39) and the relation between creep rate and generalized stress is that given by Scott (Reference 3-34). A rapidly convergent iterative technique is employed to solve the resulting nonlinear equations.

For the nominal pellet-to-clad gap, at about 1000 hours after the beginning of life, the fuel has expanded to completely fill the fuel/clad gap and to restore the clad to a circular shape after its initial creep onto the fuel. The fuel is subsequently assumed to swell unrestrained with the clad following. Based upon this conservative assumption, the final strain after 3-years' service is 0.42 percent; that is, for average fuel-to-clad gap at peak power density the strain criterion is satisfied without credit for fuel strain under load.

For the most adverse initial condition, i.e., minimum clad ID, maximum pellet OD coincident with the point of maximum power density which is assumed to be sustained over lifetime, application of the unrestrained fuel growth model in a computed strain at end of life of about 0.91 percent. However, it has been shown (Reference 3-35, 3-36 and 3-37) that the effect of restraint from the exterior cooler regions of the fuel pellet, the clad and the external pressure results in a significant limitation on radial swelling with corresponding flow of pellet material into the dish provided. The assessment of this effect, using the methods of Reference 3-36, gives an upper limit strain for these adverse conditions of 0.73 percent.

These analyses have been conducted throughout with design beginning-of-life power density, although it is known that in fuel in its third burnup cycle, the local power density will be substantially below these values. Thus, the local power density increase which might be associated with overpower transients near end of fuel life has been conservatively considered. The maximum linear heat rating for the first core is 17.6 kW/ft at BOL (actual heat generated in fuel is 97.5 percent of the total heat generated in the core); therefore, actual peak linear heat rate is less than 17.6 kW/ft, and the maximum heat rating near EOC is estimated to be 14.9 kW/ft, resulting in a BOL/EOC ratio of 1.18. This is greater than the value of 1.12 for the ratio of maximum transient to steady state heat ratings. Thus, utilization of beginning-of-life power densities in these calculations for end-of-life transients has provided considerable margin.

Studies by Notley et al (References 3-36 and 3-37), in which 27 fuel elements were irradiated without failure, reported measured clad strains up to 3.33 percent. In a series of experimental element irradiations, Westinghouse (Reference 3-35) reported strain values at failure for Zr-4 clad fuel elements of 0.78 percent to 2.6 percent, depending on the fuel properties assumed. Also, Lustman (Reference 3-38) has noted that failures in-pile have occurred at strain values between 0.5 percent and 1.0 percent. However, these results are based on relatively low Zr-4 cladding temperatures as compared to current large commercial PWR's. It is known (Reference 3-39) that permissible strain values for zircaloy increase above 650°F. The average Zr-4 cladding temperatures of about 688°F in the Fort Calhoun reactor should result in increased ductility and thus higher strain limit to failure.

Westinghouse has conducted similar studies for the 14 x 14 fuel supplied for the Fort Calhoun Unit One reactor. A detailed discussion of this work can be found in WCAP-12977 (Reference 3-22).

3.7.3.7 Control Element Assembly

Each CEA is composed of five (5) full length control rods (i.e., fingers) assembled in a square array (i.e., spider), one (1) finger in the center capture tube (i.e., hub) and four (4) fingers at each corner of the spider. The overall length of each finger is approximately 150 inches and the active length is approximately the lower 125 inches.

The active length contains neutron absorbers (i.e., poison) in the Inconel tube and each finger is sealed by a welded end cap at the bottom and by a welded end fitting at the top. The upper 25 inches of nonactive length contains a gas expansion space to limit the maximum tube stress due to internal pressure buildup by the helium gas and moisture release from the B_4C pellets during the reactor operation. All CEA's in the core include an Ag-In-Cd pallet tip to minimize the strains due to swelling induced by high neutron exposure during reactor operation. The general description of the CEA's in the core is shown in Table 3.7-1.

The hub of the spider couples the CEA to the CEDM through the extension shaft and rack. A dashpot is provided in the CEDM to slow down the CEA in the last part of the insertion following a reactor trip as described in Section 3.7-1.

There are a total of 49 CEA's in the core and these are driven by 37 CEDM's on the reactor vessel head. Twenty-one CEDM's drive trippable single CEA's, twelve CEDM's drive trippable dual CEA's and four CEDM's drive non-trippable single CEA's. The dual CEA is made up of two single CEA's connected to separate grippers and carried by an extension shaft. The arrangement of the CEA's in the core is shown in Figure 3.4-1.

The CEA's in the core are grouped into seven different groups to provide flexible operational maneuvering. Groups A and B are shutdown groups; Groups 1, 2, 3 and 4 are regulating groups to achieve reactivity control and Group N is an additional group.

The CEA's can be controlled by the group or by the individual single/dual CEA's. Except for Group N, all of the CEA's will be automatically inserted following a reactor trip. Group N can be inserted manually following a reactor trip.

3.7.3.8 Control Element Assembly Evaluation

Several parallel experimental efforts (see Section 1.4) were pursued to assure that the CEA's will function under a wide variety of adverse conditions that could be encountered under normal and abnormal operation.

The results of the cold water tests indicate that the CEA's will operate satisfactorily and achieve acceptable scram times. Various hydraulic and friction forces developed in scramming a CEA have been ascertained. Measured drop times are less than those assumed in the accident analyses.

Additional cold water flow tests were conducted on CEA's to determine the effects on drop time of guide tube mechanical and thermal bow, core pressure drop, misalignments of all applicable components, guide structure clearance variations above the core, and CEA-to-guide tube clearance variation within the fuel assembly. These test conditions were more severe than the worst accumulations of tolerances and expected operating conditions.

Burst and collapse were also performed on the poison rod cladding. Reactor operating conditions for periods up to 10,000 hours were simulated using tubing which contained defects in the wall and tubes filled with B_4C pellets that have been water logged.

Full size and weight prototype CEA's were installed in a high temperature, high pressure test facility designed to simulate pressurized water reactor coolant conditions. It is an isothermal system with capabilities of operating at temperatures up to 625°F and pressures up to 2550 psig.

Hot flow tests exposing the fuel assemblies, CEA's and a CEDM to long term reactor conditions were also conducted. The main purpose of these tests was to proof test the CEA design. CEA scram time and operational characteristics, and wear and corrosion of CEA poison tube and guide tubes were evaluated.

3.7.4 Vibration Analysis and Monitoring

Design analyses were performed to verify the structural integrity of the Fort Calhoun reactor internals and fuel assemblies. Emphasis was placed on the dynamic analysis of those components which are particularly critical and vulnerable to vibratory excitation. Thermal shields on reactors built prior to Fort Calhoun Station experienced some vibrational problems; however, for the Fort Calhoun reactor, a more reliable design was achieved by using a top vs. a bottom support design, which eliminates a free edge in the flow path; increasing the number of supports to provide a stiffer structure; and using an all welded shield to eliminate local flexibilities and relative motion at bolted joints. Operating data have shown that the thermal shield is stable on its support system when exposed to the axial annular flow.

The response of the fuel assemblies to mechanical and flow excitation was evaluated. The calculated response (amplitude and frequency) of the fundamental mode of vibration of the core support barrel was used as the mechanical excitation of the fuel assemblies. The calculated fuel assembly response was then used to assure that test conditions of fuel assemblies were more severe than expected operating conditions. Vibration analysis of the fuel assemblies demonstrates that the most likely modes of vibration do not coincide in frequency with known excitations.

A digital vibration-loose parts monitoring system as described in Reference 3-40 was installed following Cycle 14. This system was designed to provide monitoring, recording, and analysis for vibration and/or loose parts on the primary coolant loop major components and neutron flux related motion of the core and its components. The system contains accelerometers placed on each of the following locations: lower reactor vessel (2), reactor vessel flange (2), steam generator primary manway (1 on each generator), steam generator secondary handhole (1 on each generator), and steam generator secondary manway (1 on each generator). Six channels of neutron flux signals generated from the excore detectors are routed to the monitoring system for measuring core internals vibration.

The core support barrel, the support structure for the core, was initially analyzed to provide assurance that this major structure does not exhibit excessive vibrations. Vibration analysis of the barrel based on inlet flow impingement forces and turbulent flow were performed to demonstrate that the anticipated RMS response of the barrel would be low. Spectral analyses of plant operations using the vibration-loose parts monitoring system have shown that although core barrel motion is present, it is within acceptable limits.

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