Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

November 23, 2004
LIC-04-0117

U.S. Nuclear Regulatory Commission
ATTN.: Document Control Desk
Washington, DC 20555-0001

Reference: Docket No. 50-285

SUBJECT: Fort Calhoun Station Unit No. 1 License Amendment Request, "Revisions of Technical Specifications Table 1-1 and Section 4.0"

Pursuant to 10 CFR 50.90, Omaha Public Power District (OPPD) hereby transmits an application for amendment to the Fort Calhoun Station Unit 1 (FCS) Operating License.

The proposed amendment (1) revises the descriptive wording of Technical Specifications Table 1-1, RPS Limiting Safety System Settings, for the Reactor Trip setpoint for Low Steam Generator Water Level to relocate unnecessary detail, and (2) converts Technical Specifications Section 4.0, Design Features, to the content of NUREG-1432, Revision 3, *Standard Technical Specifications for Combustion Engineering Plants*. These changes will be needed to support operation of FCS after major components (steam generators, pressurizer, and reactor vessel head) are replaced in 2006.

The proposed Technical Specification changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c); it has been determined that these changes involve no significant hazards considerations. The bases for these determinations, information supporting the change, a no significant hazards consideration, and an environmental consideration are included in the attached submittal.

Attachment 1 provides the Description of Changes and Justification. Attachment 2 provides the No Significant Hazards Evaluation and the technical bases for this requested change to the Technical Specifications. Attachment 3 contains the marked-up pages and Attachment 4 contains the clean version reflecting the requested Technical Specification and Basis changes.

OPPD requests approval of the proposed amendment by November 30, 2005. Although the proposed changes are needed to support operation after major component replacements in 2006, these changes are considered to be administrative and consistent with the current plant configuration and licensing basis; they can therefore be implemented before 2006. OPPD requests a 60 day implementation period. No commitments are made to the NRC in this letter.

I declare under penalty of perjury that the forgoing is true and correct. (Executed on November 23, 2004.)

If you have any questions or require additional information, please contact Tom Matthews of my staff at 402-533-6938.

Sincerely,

D. J. Bannister
Manager – Fort Calhoun Station

DJB/RLJ/rlj

Attachments:
1. Description of Changes and Justification
2. Fort Calhoun Station's Evaluation
3. Markup of Technical Specification Pages
4. Proposed Technical Specifications (clean)

cc: Division Administrator - Public Health Assurance, State of Nebraska

ATTACHMENT 1

Description of Changes and Justification For Revisions of Technical Specifications Table 1-1 and Section 4.0

TABLE 1: LESS RESTRICTIVE REQUIREMENTS – REMOVAL OF DETAIL

Change No.	Affected FCS Technical Specification	Summary of Change	New Location	Change Control	Characterization
1	1.0 Table 1-1 Item 3	FCS TS 1.0 Table 1-1, Item 3, Trip Setpoints description is revised to relocate the statement "(Top of feedwater ring; 4'10" below normal water level)." This phrase is descriptive information which does not meet 10 CFR 50.36 criteria. This information for the replacement steam generators will be moved to the USAR to relocate unnecessary detail from the FCS Tech Specs under the change controls of 10 CFR 50.59. This information is not contained in the comparable table for Standard Technical Specifications (STS) NUREG-1432.	USAR	10 CFR 50.59	Relocation of descriptive information which does not meet 10 CFR 50.36 criteria.
2	4.2	FCS Section 4.2, Containment Design Features, is not included in the STS because these features are addressed in the USAR.	USAR	10 CFR 50.59	Relocation of descriptive information which does not meet 10 CFR 50.36 criteria.
3	4.3.1 & 4.3.3	FCS Sections 4.3.1, Reactor Coolant System, and 4.3.3, Emergency Core Cooling, are not included in the STS because these features do not	USAR	10 CFR 50.59	Relocation of descriptive information which does not meet 10 CFR

		meet 10 CFR 50.36(c)(4) criteria and are contained in the USAR.			50.36 criteria.
4	4.4.1	FCS TS 4.4.1 states that, "The new unirradiated fuel bundles will normally be stored in the dry new fuel storage rack with an effective multiplication factor of less than 0.9." This information is not included in the STS because these features are addressed in the USAR.	USAR	10 CFR 50.59	Relocation of descriptive information which does not meet 10 CFR 50.36 criteria.
5	4.4.1	FCS TS 4.4.1 states that, "The new fuel storage rack is located 18'-9" above the main floor of Room 25A which provides for adequate drainage and precludes flooding of the new fuel storage rack." This information is not included in the STS because these features are addressed in the USAR.	USAR	10 CFR 50.59	Relocation of descriptive information which does not meet 10 CFR 50.36 criteria..
6	4.4.1	FCS TS 4.4.1 states that, "New fuel may also be stored in shipping containers or in the spent fuel pool racks which have a maximum effective multiplication factor of 0.95 with Fort Calhoun Type C fuel and unborated water." This statement is not relevant to the discussion of the new fuel storage racks. Therefore, it is not included in the STS.	USAR	10 CFR 50.59	Removal of descriptive information which is already located in Tech. Spec Bases 2.8.3(1) and contained in USAR Reference 9.5-6, EA-FC-96-001, Rev 0 and does not meet 10 CFR 50.36 criteria.
7	4.4.1	FCS TS 4.4.1 states that, "The new fuel storage racks are designed as a Class I structure." This statement is not included in the STS because it is contained in the USAR.	USAR	10 CFR 50.59	Relocation of descriptive information which does not meet 10 CFR 50.36 criteria.

8	4.4.2	FCS TS 4.4.2 states that, "Irradiated fuel bundles will be stored prior to off-site shipment in the stainless steel lined spent fuel pool. The spent fuel pool is normally filled with borated water with a concentration of at least the refueling boron concentration." This statement is not included in the STS because it is contained in the USAR	USAR	10 CFR 50.59	Relocation of descriptive information which does not meet 10 CFR 50.36 criteria.
9	4.4.2	FCS TS 4.4.2 states that, "The spent fuel racks are designed as a Class I structure." This statement is not included in the STS because it is contained in the USAR	USAR	10 CFR 50.59	Relocation of descriptive information which does not meet 10 CFR 50.36 criteria.
10	4.4.2	FCS TS 4.4.2 states that, "Normally the spent fuel pool cooling system will maintain the bulk water temperature of the pool below 120°F. Under other conditions of fuel discharge, the fuel pool water temperature is maintained below 140°F." This statement is not included in the STS because it is contained in the USAR	USAR	10 CFR 50.59	Relocation of descriptive information which does not meet 10 CFR 50.36 criteria.
11	4.4.2	FCS TS 4.4.2 states that, "The spent fuel racks are designed and will be maintained such that the calculated effective multiplication factor is no greater than 0.95 (including all known uncertainties) assuming the pool is flooded with unborated water. The racks are divided into 2 regions. Storage in Region 1 and Region 2 of the spent	USAR	10 CFR 50.59	Removal of descriptive information which is already located in Tech. Spec Bases 2.8.3(1) and contained in USAR Reference 9.5-6, EA-FC-96-001, Rev 0 and does not meet 10 CFR

		fuel racks shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.5 weight percent of U-235. Region 1 and 2 cells are surrounded by Boral. Acceptance criteria for fuel storage in Regions 1 and 2 are delineated in Section 2.8 of these Technical Specifications. " These statements are not included in the STS because they are contained in the USAR and in some cases are restated by the STSs.			50.36 criteria.
12	4.5	FCS Section 4.5, Seismic Design for Class I Systems, is not included in the STS because these features are addressed in the USAR.	USAR	10 CFR 50.59	Relocation of descriptive information which does not meet 10 CFR 50.36 criteria.

TABLE 2: ADMINISTRATIVE CHANGES TO STANDARD TECHNICAL SPECIFICATIONS

Affected FCS Technical Specification	Summary of Change*
4.3.1.1.c and 4.3.1.1.d	FCS Technical Specifications 4.3.1.1.c and 4.3.1.1.d are revised from the STS to include "Region 2" and "Region 1", respectively, to assure consistent nomenclature of the high and low density fuel storage racks to be consistent with the nomenclature presented on existing FCS TS Figure 2-10.
4.3.1.e and 4.3.1.f	FCS TS 4.3.1.e and 4.3.1.f are revised from the STS wording of "unacceptable range" to "unacceptable domain" to be consistent with the wording "domain" used in existing FCS TS Figure 2-10.
4.3.1.e	FCS TS 4.3.1.e is revised from the STS wording as follows: 1) "Figure[3.7.18.1]" is changed to "Figure 2-10 for "Region 2 Unrestricted" or "Peripheral Cells"", and 2) "or in peripheral cells if acceptable for peripheral storage" has been added following "fuel storage racks(s)", to address the additional restrictions imposed on loading of peripheral cells by existing FCS TS Figure 2-10.
4.3.1.f	FCS TS 4.3.1.f is revised from the STS wording "Figure[3.7.18.1]" to "Figure 2-10 for "Region 2 Unrestricted" or "Peripheral Cells"" to address the additional restrictions imposed on loading of peripheral cells by existing FCS TS Figure 2-10. Following "will be stored in", "Region 1" has been added to specify: 1) where the spent fuel falling within this burnup domain will be stored and 2) incorporate the reference prescribing the storage requirement.
4.3.1.2.b	FCS TS 4.3.1.2.b is revised from the STS wording of "keff ≤0.98 if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR]" to "keff ≤0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Reference (2)." because the licensed FCS limit of the new fuel storage racks is 0.95.
4.3.1.2	FCS TS 4.3.1.2 is revised from the STS 4.3.1.2.c wording of "keff ≤0.98 if moderated by aqueous foam, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR]." to the deletion of an aqueous foam related specification because FCS has no system or capability of injecting foam into the new fuel storage racks.

* - The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples in Reference 10.2 (48 FR 14864) of amendments that are considered not likely to involve significant hazards consideration. The above proposed changes are consistent with example (vii) which relates to a change to make a license conform to changes in the regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations.

ATTACHMENT 2

Fort Calhoun Station's Evaluation
For
Revisions of Technical Specifications Table 1-1 and Section 4.0

1.0 DESCRIPTION

This letter is a request to amend Operating License DPR-40 for Fort Calhoun Station Unit No. 1.

The proposed changes would revise the operating License through administrative changes to the Fort Calhoun Station (FCS) Technical Specifications, specifically: (1) revises the descriptive wording of TS Table 1-1, RPS Limiting Safety System Settings, for the Reactor Trip setpoint for Low Steam Generator Water Level to relocate unnecessary detail, and (2) converts Section 4.0, Design Features, to the Design Features content of NUREG-1432, Revision 3, *Standard Technical Specifications for Combustion Engineering Plants.*

2.0 PROPOSED CHANGE

The proposed changes will be needed to support operation of FCS after major components (steam generators, pressurizer, and reactor vessel head) are replaced in 2006.

(1) Deletion of extra setpoint description from Item 3 of Technical Specifications Table 1-1, RPS Limiting Safety System Settings,

The existing Trip Setpoint description for Item 3, Low Steam Generator Water Level Reactor Trip, contains a parenthetical descriptive phrase below the main setpoint description "31.2% of Scale." OPPD proposes to relocate the parenthetical phrase "(Top of feedwater ring: 4'10" below normal water level)" to the USAR.

The low water level trip setpoint for each replacement steam generator will remain at 31.2% of scale; however, the corresponding physical location will not be consistent with the existing parenthetical description due to sizing and geometry differences.

(2) Conversion of Technical Specifications Section 4.0, Design Features, to the Design Features content of NUREG-1432, Revision 3, *Standard Technical Specifications for Combustion Engineering Plants*

The existing Technical Specification Section 4.0, Design Features, is being changed to be identical to the Design Features content of NUREG-1432, Revision 3, *Standard Technical Specifications for Combustion Engineering Plants,* through addition of standardized descriptions of the new and spent fuel storage facilities, and removal of descriptions of the containment, reactor coolant system, reactor core and control, emergency core cooling, and seismic design for Class I systems.

The 2006 component replacements affect the RCS volume value currently in TS 4.3.1; however, rather than revising just that value, OPPD has elected to convert the whole section to Standard Technical Specifications as part of a long term Technical Specification conversion plan. The RCS volume value is not included in STS.

Fort Calhoun Station (FCS) proposes to change the Technical Specification (TS) conditions, specifications, and requirements, as described in Attachment 1, "Description of Changes and Justification."

In summary, the proposed amendment relocates redundant steam generator setpoint description from the Technical Specifications and updates the existing Technical Specifications Section 4.0, Design Features, with text and content identical to the Standard Technical Specifications.

3.0 BACKGROUND

OPPD plans to replace major components (steam generators, pressurizer, and reactor vessel head) at FCS in 2006, using the provisions of 10 CFR 50.59 to the extent possible. OPPD also has a long term plan to convert the FCS Technical Specifications to Standard Technical Specifications, using a phased approach. Based on a review to determine what Technical Specification changes are needed to support operation of FCS after the major components are replaced, OPPD has determined that the changes requested in this letter are necessary. These changes are considered to be administrative and consistent with the current plant configuration; they can therefore be implemented before 2006. Engineering and safety analyses specific to the new components are ongoing; any future License Amendment Requests resulting from these analyses will be contained in separate correspondence.

4.0 TECHNICAL ANALYSIS

The proposed amendment affects two separate and distinct sections of the Technical Specifications which have been evaluated for technical significance.

(1) <u>Deletion of extra setpoint description from Item 3 of Technical Specifications Table 1-1, RPS Limiting Safety System Settings</u>

The existing Trip Setpoint description for Item 3, Low Steam Generator Water Level Reactor Trip, contains a parenthetical descriptive phrase below the main setpoint description "31.2% of Scale." This parenthetical phrase is "(Top of feedwater ring: 4'10" below normal water level)." The specific dimension for the distance between the feedwater ring and normal water level is a historical carryover from the initial 1972 Technical Specifications which define the steam generator level water inventory that is sufficient "to provide a twelve minute margin before the auxiliary feedwater is required" as stated in the existing Technical Specification 1.3(5) bases section. The setpoint analyses for the replacement steam generator, currently in progress, confirm the continued use of the existing Technical Specification Low Steam Generator trip setpoint value of "31.2% of Scale". However, the physical geometry of the replacement steam generators, when installed, will be different than the original steam generators. Elimination of the parenthetical descriptive statement relocates unnecessary and duplicative detail from the Technical Specifications, has no effect on the safety functions or requirement of the reactor protection system, and has no impact upon the USAR accident analysis. Corresponding changes to the Basis and inclusion of the information in the USAR will be appropriately controlled under 50.59 criteria.

Because the value/dimension for the normal water level to feedwater ring distance is not currently required by 10 CFR 50.36 criteria nor is the dimension utilized in Standard Technical Specifications, the proposed relocation of the parenthetical statement is acceptable.

(2) Conversion of Technical Specifications Section 4.0, Design Features, to content of NUREG-1432, Revision 3, *Standard Technical Specifications for Combustion Engineering Plants*

The existing Technical Specifications Section 4.0, Design Features, format and content is being changed to be identical to the Design Features content of NUREG-1432, Revision 3, Standard Technical Specifications for Combustion Engineering Plants through addition of standardized descriptions of the new and spent fuel storage facilities, and proposed removal of descriptions of the containment, reactor coolant system, reactor core and control, emergency core cooling, and seismic design for Class I systems. An evaluation of each of the less restrictive requirements for removal of detail and administrative changes, tabulated in Attachment I, concludes that these changes are administrative in nature because they follow the form of NRC examples for determining whether a significant hazards consideration exists. These changes do not impact the USAR accident analyses.

The design of the facility is required to be described in the Updated Safety Analysis Report (USAR) by 10 CFR 50.34. In addition, the quality assurance (QA) requirements of Appendix B to 10 CFR Part 50 require that the plant design be documented in controlled procedures and drawings, and maintained in accordance with an NRC-approved QA plan (referenced in the USAR). Controls are specified in 10 CFR 45.59 for changing the facility as described in the USAR. Removing details of system design from the custom technical specifications is acceptable because this information will be adequately controlled in the USAR.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

Omaha Public Power District has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed changes are not related to an initiator of any previously evaluated accident. The proposed changes revise descriptive information only, and will not prevent safety systems from performing their accident mitigation function as assumed in the safety analysis.

   Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed changes only relocate descriptive information in the Technical Specifications to the USAR. Modifications will not be made to existing equipment nor will any new or different types of equipment be installed. The proposed changes to the Technical Specifications will not alter assumptions made in safety analysis and licensing bases.

   Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed administrative changes only relocate descriptive information in the FCS Technical Specifications to the USAR, and have no effect on safety margins.

   Therefore, this technical specification change does not involve a significant reduction in the margin of safety.

Based on the above, Omaha Public Power District concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.1 Applicable Regulatory Requirements/Criteria

The technical analysis of the proposed amendment addressed the two separate and distinct sections of the Technical Specifications. Due to their administrative nature, the proposed changes move information from the Technical Specifications and present the same or equivalent information in the USAR. This change in information location does not change or affect commitments to Fort Calhoun Station design criteria presented in the FCS USAR Appendix G, the USAR accident analyses, approved methodologies, Regulatory Guides, or NUREGs.

For the changes listed in Attachment I, Table 1: "Less Restrictive Requirement – Removal of Detail," specific technical information which does not meet the criteria of 10 CFR 50.36 for inclusion in the Technical Specifications is being moved to the USAR under the guidance of 50.59. Changes listed in Table 2: "Administrative Changes" meet the Commission's guidance example concerning the application of the standards for determining whether a significant hazards consideration exists. The "Description of

Changes and Justification," for the administrative changes for the proposed changes cites the appropriate NRC provided example.

The proposed changes do not alter, degrade, or prevent actions described or assumed in any accident analysis. The proposed changes will not change any assumptions previously made in evaluating radiological consequences or affect any fission product barriers, nor do they increase any challenges to safety systems. Therefore, the proposed change does not increase or have any impact on the consequences of events described and evaluated in Chapter 14 of the Fort Calhoun USAR.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

Based on the above considerations, the proposed amendment does not involve and will not result in a condition which significantly alters the impact of Fort Calhoun Station on the environment. Thus, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR Part 51.22(c)(9), and, pursuant to 10 CFR Part 51.22(b), no environmental assessment need be prepared.

7.0 REFERENCES

7.1 NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants"
7.2 Standards for Determining Whether License Amendments Involve No Significant Hazards Considerations, 48 FR 14864

Attachment 3

Markup of
Technical Specification Pages

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE 1-1

RPS LIMITING SAFETY SYSTEM SETTINGS

No.	**Reactor Trip**	**Trip Setpoints**
1	High Power Level (A) 4-Pump Operation	≤109.0% of Rated Power
2	Low Reactor Coolant Flow (B)(F) 4-Pump Operation	≥95% of 4 Pump Flow
3	Low Steam Generator Water Level	31.2% of Scale ~~(Top of feedwater ring; 4'10" below normal water level)~~
4	Low Steam Generator Pressure (C)	≥500 psia
5	High Pressurizer Pressure	≤2400 psia
6	Thermal Margin/Low Pressure (B)(F)	1750 psia to 2400 psia (depending on the reactor coolant temperature as shown in the Thermal Margin/Low Pressure 4 Pump Operation Figure provided in the COLR)
7	High Containment Pressure (D)	≤5 psig
8	Axial Power Distribution (E)	(as shown in the Axial Power Distribution for 4 Pump Operation Figure provided in the COLR)
9	Steam Generator Differential Pressure	≤135 psid

4.0 DESIGN FEATURES

4.1 Site

The site for Fort Calhoun Station Unit No. 1 is in Washington County, Nebraska, on the west bank of the Missouri River and approximately nineteen miles north, northwest of the city of Omaha, Nebraska. The exclusion area, as defined in 10 CFR Part 100, Section 100.3(a), consists of approximately 1242 acres. The exclusion area boundary extent includes approximately 660 acres in Washington County, Nebraska, owned by the Omaha Public Power District (OPPD), and 582 acres in Harrison County, Iowa, on the east bank of the river directly opposite the facility, on which the District retains perpetual easement rights. The minimum exclusion area boundary point is located approximately at the 187.0 degree radial from the outer wall of the containment building and at a distance of 910 meters.

4.2 Reactor Core

4.2.1 Fuel Assemblies

[redacted]

4.2.2 Control Element Assemblies

[redacted]

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.5 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.5 of the USAR.

4.0 **DESIGN FEATURES** (Continued)

c. A nominal 8.6 inch center to center distance between fuel assemblies placed in Region 2, the high density fuel storage racks,

d. A nominal 9.8 inches (East-West) by 10.3 inches (North-South) center to center distances between fuel assemblies placed in Region 1, the low density fuel storage racks,

e. New or partially spent fuel assemblies with a discharge burnup in the "acceptable domain" of Figure 2-10 for "Region 2 Unrestricted" or "Peripheral Cells" may be allowed unrestricted storage in either fuel storage racks or peripheral cells if acceptable for peripheral cell storage, and

f. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable domain" of Figure 2-10 for "Region 2 Unrestricted" or "Peripheral Cells" will be stored in Region 1 in compliance with the Reference (1).

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent,

b. keff ≤ 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Reference (2).

c. A nominal 16 inch center to center distance between fuel assemblies placed in the storage racks.

<u>References:</u>

(1) Letter from R. Wharton (NRC) to T. Patterson (OPPD), Amendment 174 to Facility Operating License No. DPR-49, (TAC NO. M94789) Dated July 30, 1996, NRC-96-0126.

(2) Ft. Calhoun USAR, Reference 9.5-1

4.0 **DESIGN FEATURES** (Continued)

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 23 ft.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1083 fuel assemblies.

~~4.0 DESIGN FEATURES~~

~~4.2 Containment Design Features~~

~~4.2.1 Containment Structure~~

~~The containment structure completely encloses the reactor coolant system to minimize release of radioactive material to the environment should a failure of the reactor coolant system occur. The prestressed, post tensioned concrete structure provides adequate biological shielding for both normal operation and accident situations and is designed for low leakage at a design pressure of 60 psig and 305°F.~~

~~The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a design basis loss of coolant accident. In this event, the total energy contained in the water of the reactor coolant system is assumed to be released into the containment through a double-ended break of the largest reactor coolant pipe coincident with a loss of normal and off-site electrical power. Subsequent pressure behavior is determined by the engineered safety features and the combined influence of energy sources and heat sinks.~~

~~The external design pressure of the containment shell is 2.5 psig; this is the positive differential pressure that would result if the containment were sealed during a period of low barometric pressure and high internal temperature and subsequently, the containment atmosphere were cooled with a concurrent major rise in barometric pressure. Vacuum breakers are therefore not provided.~~

~~The containment is designed as a seismic Class I structure.~~

~~4.2.2 Penetrations~~

~~All penetrations through the steel-lined concrete structure for electrical conductors, pipe, ducts, air locks and hatch doors are of the double barrier design.~~

~~The automatically actuated containment isolation valves are designed to close upon low pressurizer pressure or high pressure in the containment structure.[1] No single component failure in the actuation system will prevent the isolation valves from functioning as designed.~~

~~4.0 **DESIGN FEATURES**~~

~~4.2 Containment Design Features (Continued)~~

~~4.2.3 Containment Structure Cooling Systems~~

~~The containment air recirculation, cooling, and iodine removal system includes four separate self-contained units which cool the containment air during normal operation and limit the pressure rise in the event of a design accident. A cooling water flow of 4,680 gpm with an inlet temperature of 120°F will remove 280 x 10^6 Btu/hr.~~[(2)]

~~The containment spray system is capable of removing 280 x 10^6 Btu/hr (2 pumps) from the containment atmosphere at 288°F by spraying cool borated water from the 314,000 gallon SIRW tank. Recirculation of spray water from the containment sump and through the shutdown cooling heat exchangers into the containment atmosphere is also provided. Under this mode of operation, the heat removal capability per heat exchanger is 87.5 x 10^6 Btu/hr based upon 4,000 gpm of cooling water at 114°F inlet temperature.~~[(3)]

~~Two of the four containment air cooling units are equipped with particulate filters and with activated impregnated charcoal absorbers for iodine removal.~~

~~References~~

~~(1) FSAR, Table 5.9-1~~

~~(2) FSAR, Table 6.4-5~~

~~(3) FSAR, Table 6.3.1~~

~~4.0 DESIGN FEATURES~~

~~4.3 Nuclear Steam Supply System (Continued)~~

~~4.3.1 Reactor Coolant System (Continued)~~

~~The reactor coolant system is designed and constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Vessels including all addenda through the winter of 1967 and the Code for Pressure Piping USAS B31.1. The reactor coolant system is designed for a pressure of 2500 psia and a temperature of 650°F except for the pressurizer which has a design temperature of 700EF. The volume of the reactor coolant system is approximately 6,616 cubic feet.~~

~~4.3.2 Reactor Core and Control~~

~~The reactor shall contain 133 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or ZIRLO7 fuel rods with an initial composition of natural, depleted, or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.~~

~~The reactor core shall contain 49 control element assemblies (CEAs). The control material shall be silver indium cadmium, boron carbide, or hafnium metal as approved by the NRC.~~

~~4.3.3 Emergency Core Cooling~~

~~Emergency core cooling is provided by the Safety Injection System which consists of various subsystems, each with internal redundancy. Included in the Safety Injection System are four safety injection tanks, three high-pressure and two low-pressure safety injection pumps, a safety injection and refueling water storage tank, and interconnecting piping as shown in USAR Section 6.~~

~~4.0 DESIGN FEATURES~~

~~4.4 Fuel Storage~~

~~4.4.1 New Fuel Storage~~

~~The new unirradiated fuel bundles will normally be stored in the dry new fuel storage rack with an effective multiplication factor of less than 0.9. The new fuel storage rack is located 18'-9" above the main floor of Room 25A which provides for adequate drainage and precludes flooding of the new fuel storage rack.~~

~~New fuel may also be stored in shipping containers or in the spent fuel pool racks which have a maximum effective multiplication factor of 0.95 with Fort Calhoun Type C fuel and unborated water.~~

~~The new fuel storage racks are designed as a Class I structure.~~

~~4.4.2 Spent Fuel Storage~~

~~Irradiated fuel bundles will be stored prior to off-site shipment in the stainless steel lined spent fuel pool. The spent fuel pool is normally filled with borated water with a concentration of at least the refueling boron concentration.~~

~~The spent fuel racks are designed as a Class I structure.~~

~~Normally the spent fuel pool cooling system will maintain the bulk water temperature of the pool below 120°F. Under other conditions of fuel discharge, the fuel pool water temperature is maintained below 140°F.~~

~~The spent fuel racks are designed and will be maintained such that the calculated effective multiplication factor is no greater than 0.95 (including all known uncertainties) assuming the pool is flooded with unborated water. The racks are divided into 2 regions. Storage in Region 1 and Region 2 of the spent fuel racks shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.5 weight percent of U-235. Region 1 and 2 cells are surrounded by Boral. Acceptance criteria for fuel storage in Regions 1 and 2 are delineated in Section 2.8 of these Technical Specifications.~~

~~4.0 DESIGN FEATURES~~

~~4.5 Seismic Design for Class I Systems~~

~~Class I systems and equipment including piping (excluding the reactor coolant system) are designed to the criteria for load combinations and stresses shown in Table F-1 of the FSAR.~~

~~Design criteria for the reactor coolant are as shown in Table 4.2-3 of the FSAR.~~

Attachment 4

Clean-Typed Technical Specification Pages

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TABLE OF CONTENTS (Continued)

TABLE 1-1

RPS LIMITING SAFETY SYSTEM SETTINGS

No.	Reactor Trip	Trip Setpoints
1	High Power Level (A) 4-Pump Operation	≤109.0% of Rated Power
2	Low Reactor Coolant Flow (B)(F) 4-Pump Operation	≥95% of 4 Pump Flow
3	Low Steam Generator Water Level	31.2% of Scale
4	Low Steam Generator Pressure (C)	≥500 psia
5	High Pressurizer Pressure	≤2400 psia
6	Thermal Margin/Low Pressure (B)(F)	1750 psia to 2400 psia (depending on the reactor coolant temperature as shown in the Thermal Margin/Low Pressure 4 Pump Operation Figure provided in the COLR)
7	High Containment Pressure (D)	≤5 psig
8	Axial Power Distribution (E)	(as shown in the Axial Power Distribution for 4 Pump Operation Figure provided in the COLR)
9	Steam Generator Differential Pressure	≤135 psid

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4.0 DESIGN FEATURES

4.1 Site

The site for Fort Calhoun Station Unit No. 1 is in Washington County, Nebraska, on the west bank of the Missouri River and approximately nineteen miles north, northwest of the city of Omaha, Nebraska. The exclusion area, as defined in 10 CFR Part 100, Section 100.3(a), consists of approximately 1242 acres. The exclusion area boundary extent includes approximately 660 acres in Washington County, Nebraska, owned by the Omaha Public Power District (OPPD), and 582 acres in Harrison County, Iowa, on the east bank of the river directly opposite the facility, on which the District retains perpetual easement rights. The minimum exclusion area boundary point is located approximately at the 187.0 degree radial from the outer wall of the containment building and at a distance of 910 meters.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 133 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or ZIRLO fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Element Assemblies

The reactor core shall contain 49 control element assemblies (CEAs). The control material shall be silver indium cadmium, boron carbide, or hafnium metal as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.5 weight percent,

b. k_{eff} $\leq$0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.5 of the USAR,

4.0 **DESIGN FEATURES** (Continued)

c. A nominal 8.6 inch center to center distance between fuel assemblies placed in Region 2, the high density fuel storage racks,

d. A nominal 9.8 inches (East-West) by 10.3 inches (North South) center to center distances between fuel assemblies placed in Region 1, the low density fuel storage racks,

e. New or partially spent fuel assemblies with a discharge burnup in the "acceptable domain" of Figure 2-10 for "Region 2 Unrestricted" or "Peripheral Cells" may be allowed unrestricted storage in either fuel storage racks or peripheral cells if acceptable for peripheral cell storage, and

f. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable domain" of Figure 2-10 for "Region 2 Unrestricted" or "Peripheral Cells" will be stored in Region 1 in compliance with the Reference (1).

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent,

b. keff ≤ 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Reference (2).

c. A nominal 16 inch center to center distance between fuel assemblies placed in the storage racks.

References:

(1) Letter from R. Wharton (NRC) to T. Patterson (OPPD), Amendment 174 to Facility Operating License No. DPR-40, (TAC NO. M94789) Dated July 30, 1996, NRC-96-0126.

(2) Ft. Calhoun USAR, Reference 9.5-1

4.0 **DESIGN FEATURES** (Continued)

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 23 ft.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1083 fuel assemblies.