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 700°F. The volume of the reactor coolant system is approximately 6,616 cubic feet.

4.3.2 Reactor Core and Control

~~The reactor core shall approximate a right circular cylinder with an equivalent diameter of 106.5 inches and an active height of 128 inches. The reactor core shall normally consist of Zircaloy-4 clad fuel rods containing slightly enriched uranium in the form of sintered UO_2 pellets. The fuel rods shall normally be grouped into 133 assemblies.~~

~~The core excess reactivity shall be controlled by a combination of boric acid chemical shim, control element assemblies, and mechanically fixed boron rods where required. Forty-nine control element assemblies are distributed throughout the core as shown in Figure 3.4-4 of the USAR; four of the CEA's are full length non-trippable CEA's.~~
The reactor shall contain 133 fuel assemblies. Each assembly shall consist of a matrix of zircaloy 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.

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.

5.9.5 Core Operating Limits Report

a. Core Operating Limits shall be established and documented in the Core Operating Limits Report (COLR) before each reload cycle or any remaining part of a reload cycle.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as follows:

1. OPPD-NA-8301-P-A, "Reload Core Analysis Methodology Overview" (Latest NRC approved revision as stated in the COLR).

2. OPPD-NA-8302-P-A, "Neutronics Design Methods and Verification" (Latest NRC approved revision as stated in the COLR).

3. OPPD-NA-8303-P-A, "Transient and Accident Methods and Verification" (Latest NRC approved revision as stated in the COLR).

4. WCAP-12610, "VANTAGE + Fuel Assembly Report," June 1990 (Westinghouse Proprietary)

c. The core operating limits shall be determined so that all applicable limits of the safety analysis are met. The Core Operating Limits Report, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Region IV Administrator and Senior Resident Inspector.

ATTACHMENT B

DISCUSSION, JUSTIFICATION AND NO SIGNIFICANT HAZARDS CONSIDERATIONS

DISCUSSION AND JUSTIFICATION:

Omaha Public Power District (OPPD), the licensee for Fort Calhoun Station (FCS), Unit No. 1, proposes to amend the Technical Specifications (TS) contained in Appendix A of License No. DPR-40 as follows:

TS 4.3.2, Reactor Core and Control

TS 4.3.2 is proposed for revision to modify the description of fuel and control element assemblies. Following the proposed revision, TS 4.3.2 will be identical to improved STS 4.2 "Reactor Core" of NUREG-1432. The revision will clarify that the reactor may contain fuel assemblies consisting of a matrix of zircaloy or ZIRLO® clad fuel rods.

The change will also permit the limited substitution of zirconium alloy, stainless steel filler rods, or lead test assemblies for fuel rods in accordance with NRC-approved applications of fuel rod configurations that have been analyzed with NRC-approved methods. This will allow the timely removal of leaking fuel rods or those that are a probable source of future leakage. This change also makes provisions for the loading of lead test assemblies into the reactor without requiring a specific TS change.

TS 5.9.5, Core Operating Limits Report

The proposed revision of TS 4.3.2 is supported by Westinghouse Topical Report, WCAP-12610, "VANTAGE + Fuel Assembly Report," dated June 1990 (Westinghouse Proprietary). This topical report describes the fuel rod design bases, criteria and models, which are affected by the use of ZIRLO® cladding. Consequently, WCAP-12610 is proposed for addition to the list of analytical methods located in TS 5.9.5b that are used to determine the core operating limits.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATIONS:

The proposed Technical Specification (TS) changes do not involve significant hazards considerations because operation of Fort Calhoun Station (FCS) Unit No. 1 in accordance with these changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed revision to TS 4.3.2 is based on improved STS 4.2 of NUREG-1432. ZIRLO® is similar in chemical composition, physical and mechanical properties to Zircaloy-4, but features improved corrosion performance and dimensional stability. These characteristics ensure that fuel rod cladding integrity and fuel assembly structural integrity are maintained. Fuel assemblies manufactured with ZIRLO® clad fuel rods meet the same design bases requirements as fuel assemblies manufactured with Zircaloy-4 cladding and the regulatory requirements of 10 CFR 50.46 are applicable to either material.

No concerns have been identified pertaining to reactor operation with a core comprised of fuel assemblies manufactured with Zircaloy-4 clad rods and fuel assemblies manufactured with ZIRLO® clad rods. ZIRLO® clad fuel rods do not require a change to the FCS reload design and safety analysis limits. Radiological consequences of previously evaluated accidents are not increased because the safety analysis dose predictions are not sensitive to the type of cladding material used. The proposed limited substitution of zirconium alloy or stainless steel filler rods in accordance with NRC-approved fuel rod configurations will allow leaking fuel rods (or potential leakers) to be removed. Therefore, the radiological consequences of accidents previously evaluated in the FCS Updated Safety Analysis Report (USAR) are not increased by this change.

The revisions to TS 4.3.2 listed above will not result in a change to any of the process variables that might initiate an accident or affect the radiological release for an accident. The operating limits will not be changed and the analysis methods to demonstrate operation within the limits will remain in accordance with NRC-approved methodology. There are no physical changes to the plant associated with the change to TS 4.3.2 other than the changes to the fuel assemblies. Therefore, this revision does not involve a significant increase in the probability or consequences of an accident previously evaluated because the safety analysis to be performed for each cycle will continue to demonstrate compliance with all fuel safety design bases.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATIONS (CONTINUED):

The proposed revision of TS 4.3.2 is supported by Westinghouse Topical Report, WCAP-12610, "VANTAGE + Fuel Assembly Report," dated June 1990 (Westinghouse Proprietary). This topical report describes the fuel rod design bases, criteria and models, which are affected by the use of ZIRLO® cladding. Consequently, WCAP-12610 is proposed for addition to the list of analytical methods located in TS 5.9.5b that are used to determine the core operating limits.

Based on the above discussion, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

Fuel assemblies manufactured with ZIRLO® clad fuel rods must meet original design criteria and thus they will not be an initiator for any new or different kind of accident. All design and performance criteria will continue to be met by fuel assemblies manufactured with ZIRLO® clad fuel rods and no new single failure mechanisms have been found.

The use of fuel assemblies manufactured with ZIRLO® cladding does not involve any alterations to plant equipment or procedures that would introduce any new or unique operational modes or accident precursors. The substitution of zirconium alloy, stainless steel filler rods, or lead test assemblies for fuel rods will be limited to NRC-approved fuel rod configurations. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created by this change.

(3) Involve a significant reduction in a margin of safety.

The use of fuel assemblies manufactured with ZIRLO® clad rods does not change the proposed FCS reload design and safety analysis limits. The normal operating conditions allowed for in the Technical Specifications will be taken into consideration for the use of these fuel assemblies. For each cycle reload core, the fuel assemblies will be evaluated using NRC-approved reload design methods to include consideration of the core physics analysis peaking factors and core average linear heat rate effects.

NRC-approved methods will also be used to analyze each configuration of zirconium alloy or stainless steel filler rods in fuel assemblies to demonstrate continued safe operation within the limits that assure acceptable plant response to accidents and transients. Therefore, this change does not involve a significant reduction in a margin of safety.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATIONS (CONTINUED):

Based on the above considerations, it is OPPD's position that this proposed amendment does not involve significant hazards considerations as defined by 10 CFR 50.92. The proposed changes will not result in a condition that significantly alters the impact of the Station on the environment. Thus, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), and pursuant to 10 CFR 51.22(b) no environmental assessment need be prepared.