Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

December 1, 2003
LIC-03-0132

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

Reference: Docket No. 50-285

SUBJECT: Fort Calhoun Station Unit No. 1 License Amendment Request, "Various Administrative and Editorial Changes"

Pursuant to 10 CFR 50.90, Omaha Public Power District (OPPD) hereby proposes to make administrative and editorial changes to the Fort Calhoun Station (FCS) Technical Specifications. The proposed change consists primarily of typographical changes and relocation of material not required to be in Technical Specifications.

The proposed Technical Specification change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c); it has been determined that this change involves 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, No Significant Hazards Evaluation and the technical bases for this requested change to the Technical Specifications. Attachment 2 contains the marked-up pages and Attachment 3 contains the clean version reflecting the requested Technical Specification and Basis changes.

OPPD requests approval of the proposed amendment by June 30, 2004. OPPD requests 120 days to implement this amendment. No commitments are made to the NRC in this letter.

In accordance with 10 CFR 50.91, a copy of this application is being submitted to the designated Nebraska State Official.

I declare under penalty of perjury that the foregoing is true and correct. (Executed on November 25, 2003)

If you have any questions or require additional information, please contact Dr. R. L. Jaworski at (402) 533-6833.

Sincerely,

R. T. Ridenoure
Vice President

RTR/TRB/trb

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

c: B. S. Mallett, NRC Regional Administrator, Region IV
A. B. Wang, NRC Project Manager
J. G. Kramer, NRC Senior Resident Inspector
Division Administrator - Public Health Assurance, State of Nebraska

ATTACHMENT 1

Fort Calhoun Station's Evaluation
for
Amendment of Operating License

Fort Calhoun Station's Evaluation For Amendment of Operating License

1.0 INTRODUCTION

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

The proposed amendment proposes to make administrative and editorial changes to the FCS Technical Specifications (TS). The proposed changes consist primarily of typographical changes and relocation of material not required to be in the TS.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

FCS proposes to change the TS conditions, specifications, and requirements as follows:

The requirements of Item 14 of Table 3-3 regarding testing of Nuclear Detector Well Cooling Annulus Exit Air Temperature Detectors are proposed to be relocated to the FCS Updated Safety Analysis Report (USAR). In Amendment 214 dated January 6, 2003, the corresponding Limiting Condition for Operation (LCO) for Detector Well Cooling was deleted. The Surveillance for this LCO remained in the TS. OPPD proposes to relocate this Surveillance from the TS to the USAR for the same reasons already presented in Amendment 214, i.e., this Surveillance does not meet any of the criteria in 10 CFR 50.36(c)(2)(ii). This change is administrative since the relocation of Detector Well Cooling requirements has already been approved by the NRC in Amendment 214.

The title of Item of 10a.2 of Table 3-5 is being corrected from "Labaratory Testing" to "Laboratory Testing". This is an obvious typographical error. Therefore this change is administrative.

TS Section 3.17(5)(ii) are proposed to be changed to delete the words, "pursuant to Section 5.6 of the Technical Specifications," from the TS. The associated Bases are also being updated with the same change. This is being done for consistency with the relocation of TS Section 5.6 described below. Therefore this change is administrative.

The details of TS Section 5.5, "Review and Audit", are proposed to be relocated to the FCS USAR. The review and audit activities performed by the Plant Review Committee (PRC) and the Safety Audit and Review Committee (SARC) are required by ANSI N18.1-1976. Additional requirements are contained in 10 CFR 50.54(p), 10 CFR 50.54(t), 10 CFR 50 Appendix B Criterion XVIII, 10 CFR 73, and ANSI N45.2-1971. The current review and audit activities can be addressed in adequate detail in the FCS USAR and need

not be repeated in the TS. Changes to the FCS USAR are controlled by 10 CFR 50.59 to ensure proper reviews are performed. Therefore this change is administrative.

The details of TS Section 5.6, "Reportable Event Action" are proposed to be relocated to FCS Plant Procedures. The requirements and actions for Reportable Events are contained in 10 CFR 50.72 and 10 CFR 50.73. Repeating these requirements in the TS is unnecessary and creates a potential burden when regulations change. The requirements for performing reviews of Reportable Events are information that can be adequately controlled in plant procedures under 10 CFR 50.59 control. Therefore this change is administrative.

The details of TS Section 5.7.1.b, 5.7.1.c, and 5.7.1.d regarding the preparation, review and processing of reports of Safety Limit violations are proposed to be relocated to plant procedures. The requirements and actions for Reportable Events are contained in 10 CFR 50.72 and 10 CFR 50.73. The requirements for preparation, review and processing of reports of Safety Limit violations are information that can be adequately controlled in plant procedures under 10 CFR 50.59 control. Therefore this change is administrative.

The details of TS Section 5.9.1.a, "Startup Report," are proposed to be relocated to the FCS USAR. The Startup Report provides the NRC a mechanism to review the appropriateness of licensee activities after-the-fact, but provides no regulatory authority once the report is submitted (i.e., no requirement for NRC approval). The Quality Assurance requirements of 10 CFR 50, Appendix B and the Startup Test Program provisions contained in the USAR provide assurance the listed activities will be adequately performed and that appropriate corrective actions, if required, are taken. The placement of these TS requirements in the USAR also ensures that change control is performed in accordance with 10 CFR 50.59. Therefore this change is administrative.

The details of TS Section 5.9.4.c, "Fire Protection Program Deficiency Report," are proposed to be relocated to plant procedures. Fire Protection Program requirements are specified in License Condition 3.D and need not be placed in the TS. The placement of these TS requirements in the USAR also ensures that change control is performed in accordance with 10 CFR 50.59. Therefore this change is administrative.

3.0 BACKGROUND

The proposed changes are a result of the correction of discrepancies or confusing statements identified by the FCS condition reporting system.

4.0 REGULATORY REQUIREMENTS & GUIDANCE

The proposed Technical Specifications changes are administrative in nature. The Commission has provided guidance concerning the application of the standards for

determining whether a significant hazards consideration exists by providing certain examples (Reference 10.2) of amendments that are considered not likely to involve significant hazards consideration. Section 2.0 for each of the proposed changes cites one or more of these NRC provided examples.

The proposed amendment relocates the requirements of Item 14c of Table 3-3 regarding testing of Nuclear Detector Well Cooling Annulus Exit Air Temperature Detectors to the FCS USAR.

FCS was licensed for construction prior to May 21, 1971, and at that time committed to the preliminary General Design Criteria (GDC). These preliminary design criteria are contained in the FCS USAR Appendix G.

This activity complies with FCS Design Criterion 10, "Containment," which is similar to 10 CFR 50 Appendix A GDC 16, "Containment design." FCS Design Criterion 10 states that containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

This activity also complies with FCS Design Criterion 40, "Missile Protection," which is similar to 10 CFR 50 Appendix A GDC 4, "Environmental and dynamic effects design bases." FCS Design Criterion 40 states that protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

This activity also complies with FCS Design Criterion 49, "Containment Design Basis," which is similar to 10 CFR 50 Appendix A GDC 50, "Containment design basis." FCS Design Criterion 49 states that the containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

This activity also complies with FCS Design Criterion 50, "NDT Requirement for Containment Material," which is similar to 10 CFR 50 Appendix A GDC 51, "Fracture prevention of containment pressure boundary." FCS Design Criterion 50 states that principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperature under normal operating and testing conditions are not less than 30°F above nil ductility transition (NDT) temperature.

All of these FCS Design Criteria will continue to be satisfied after the change to relocate the requirements of Item 14c of Table 3-3 to the FCS USAR. The FCS accident analyses do not assume operation of the nuclear detector cooling system.

5.0 TECHNICAL ANALYSIS

Evaluation

The proposed amendment relocates the requirements of Item 14c of Table 3-3 regarding testing of Nuclear Detector Well Cooling Annulus Exit Air Temperature Detectors to the FCS USAR. The accident analyses do not assume operation of the nuclear detector cooling system. Therefore, this system does not meet the criteria set forth in 10 CFR 50.36(c)(2)(ii) for inclusion in the TS, and the requirements will be relocated to the USAR. Any future changes to these requirements will be evaluated under 10 CFR 50.59.

Relocation of the requirements of Item 14c of Table 3-3 regarding testing of Nuclear Detector Well Cooling Annulus Exit Air Temperature Detectors to the USAR in conjunction with existing USAR information will continue to ensure that the biological shield structural concrete temperature is maintained below 150°F.

Comparison to Screening Criteria of 10 CFR 50.36(c)(2)(ii):

Criterion 1

The nuclear detector cooling system is not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

The nuclear detector cooling system is not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

The nuclear detector cooling system is not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4

The FCS Probabilistic Safety Assessment does not address the nuclear detector cooling system. This system is considered to be a non-risk contributor to the core damage frequency and offsite releases.

Conclusion

Since the screening criteria have not been satisfied, the nuclear detector cooling system requirements may be relocated to licensee controlled documents outside the Technical Specifications.

This change and the remaining proposed Technical Specifications changes are administrative in nature. Section 2.0 provides a description of each proposed change and a justification as to why this change is considered administrative.

6.0 REGULATORY ANALYSIS

The proposed amendment relocates the requirements of Item 14c of Table 3-3 regarding testing of Nuclear Detector Well Cooling Annulus Exit Air Temperature Detectors to the FCS USAR. The accident analyses do not assume operation of the nuclear detector cooling system. Therefore, this system does not meet the criteria set forth in 10 CFR 50.36(c)(2)(ii) for inclusion in the TS, and the requirements will be relocated to the USAR. This complies with the regulatory requirements in FCS Design Criteria 10, 40, 49, and 50 by continuing to prevent damage to the containment structure.

The proposed Technical Specifications changes are administrative in nature. Section 2.0 provides a description of each proposed change and a justification as to why this change is considered administrative.

It should also be noted that the proposed changes do not alter, degrade, or prevent actions described or assumed in any accident analysis. They 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 FCS 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.

7.0 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 change relocates requirements for Nuclear Detector Cooling that do not meet the criteria for inclusion in the TS set forth in 10 CFR 50.36(c)(2)(ii). The requirements for Nuclear Detector Cooling are being relocated from TS to the USAR, which will be maintained pursuant to 10 CFR 50.59, thereby reducing the level of regulatory control. The level of regulatory control has no impact on the probability or consequences of an accident previously evaluated. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The correction of typographical errors and relocation of specifications is not an initiator of any previously evaluated accident. The proposed changes 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 change relocates requirements for Nuclear Detector Cooling that do not meet the criteria for inclusion in TS set forth in 10 CFR 50.36(c)(2)(ii). The proposed change only affects the technical specifications and does not involve a physical change to the plant. Modifications will not be made to existing components nor will any new or different types of equipment be installed. The proposed change corrects typographical errors and relocates information that is unnecessary in the TS. This change 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 change relocates requirements for Nuclear Detector Cooling that do not meet the criteria for inclusion in TS set forth in 10 CFR 50.36(c)(2)(ii). The change will not reduce a margin of safety since the location of a requirement has no impact on any safety analysis assumptions. In addition, the relocated requirements for Nuclear Detector Cooling remain the same as the existing TS. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, there will be no reduction in a margin of safety.

The additional proposed changes correct typographical errors and relocate redundant information not required to be in the TS.

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.

8.0 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 FCS 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.

9.0 PRECEDENTS

Similar amendment requests that would support this proposed change are cited in Section 2.0 where required to support the fact that these changes are administrative changes.

10.0 REFERENCES

10.1 NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants"

10.2 Standards for Determining Whether License Amendments Involve No Significant Hazards Considerations, 48 FR 14864

ATTACHMENT 2

Markup of Technical Specification Pages

TABLE 3-3 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS

Channel Description	Surveillance Function	Frequency	Surveillance Method
14. Not Used ~~Nuclear Detector Well Cooling Annulus Exit Air Temperature Detectors~~	~~a. Check~~	~~S~~	~~a. CHANNEL CHECK~~
	~~b. Calibrate~~	~~R~~	~~b. CHANNEL CALIBRATION~~
15. Reactor Coolant System Flow	a. Check	R(1)	a. Calculation of reactor coolant flow rate.
16. Pressurizer Pressure	a. Check	S	a. CHANNEL CHECK
17. Reactor Coolant Inlet Temperature	a. Check	S	a. CHANNEL CHECK
18. Low-Temperature Set-point Power-Operated Relief Valves	a. Test	PM	a. CHANNEL FUNCTIONAL TEST (excluding actuation)
	b. Calibrate	R	b. CHANNEL CALIBRATION

(1) Required to be performed within 24 hours after ≥95.00% reactor thermal power following power escalation.

TABLE 3-5 (continued)

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

	Test	Frequency	USAR Section Reference
10a. (continued)	2. Laboaratory Testing** Verify, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows methyliodide penetration less than 0.175% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.	On a refueling frequency or every 720 hours of system operation or after any structural maintenance on the HEPA filter or charcoal adsorber housing or following significant painting, fire or chemical release in a ventilation zone communicating with the system.	
	3. Overall System Operation		
	a. Each circuit shall be operated.	Ten hours every month.	
	b. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 9 inches of water at system design flow rate.	R	
	c. Fan shall be shown to operate within ± 10% design flow.	R	
	4. Automatic and manual initiation of the system shall be demonstrated.	R	

**Tests shall be performed in accordance with applicable section(s) of ANSI N510-1980.

3.0 **SURVEILLANCE REQUIREMENTS**
3.17 Steam Generator Tubes (Continued)

(iii) Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported ~~pursuant to Section 5.6 of the Technical Specifications~~ prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

3.0 **SURVEILLANCE REQUIREMENTS**
3.17 Steam Generator Tubes (Continued)

Basis

The surveillance requirements for inspection of the steam generator tubes and tube sleeves ensure that the structural integrity of this portion of the RCS will be maintained. The program for in-service inspection of the steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1, dated July 1975. The program for in-service inspection of steam generator tube sleeves is based on a modification of EPRI PWR Steam Generator Examination Guidelines, Revision 5, Dated September 1997. In-service inspection of steam generator tubing and tube sleeves is essential in order to maintain surveillance of the conditions of the tubes and sleeves in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or in-service conditions that lead to corrosion.

In-service inspection of steam generator tubing and tube sleeves also provides a means of characterizing the nature and cause of any tube or sleeve degradation so that corrective measures can be taken.

Tubes with defects may be repaired by a Combustion Engineering, Inc. Leak Tight Sleeve. The technical bases for sleeving repair are described in the Proprietary Combustion Engineering, Inc. Report CEN-630-P, Revision 02, "Repair of 3/4" O.D. Steam Generator Tubes Using Leak Tight Sleeves," June 1997.

Whenever the results of any steam generator tubing in-service inspection fall into Category C-3, these results will be promptly reported to the Commission ~~pursuant to Section 5.6 of the Technical Specifications~~ prior to the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

5.4 Training

5.4.1 A retraining and replacement training program for the plant staff shall be maintained under the direction of the Manager - Training and shall meet or exceed the requirements of Section 6 of ANSI/ANS 3.1-1993, as modified by Regulatory Guide 1.8, Revision 3, dated May 2000 and 10 CFR Part 55.

5.5 Not Used ~~Review and Audit~~

~~5.5.1 Plant Review Committee (PRC)~~

~~A committee composed of key management personnel designated as the PRC acts in an advisory capacity on all matters related to nuclear safety to the plant manager and serves in accordance with Quality Assurance Program requirements, USAR Section 12.5, and plant Standing Orders.~~

~~5.5.2 Safety Audit and Review Committee (SARC)~~

~~The Safety Audit and Review Committee (SARC) is a committee composed of highly qualified and experienced OPPD management personnel and consultants, which functions to provide independent review and audit of activities in accordance with Quality Assurance Program requirements, USAR Section 12.5, and the SARC Charter. The SARC reports to and advises the corporate officer responsible for overall plant nuclear safety.~~

5.6 Not Used ~~Reportable Event Action~~

~~5.6.1 The following actions shall be taken in the event of a REPORTABLE EVENT:~~

~~a. The Commission shall be notified pursuant to the requirements of 10 CFR 50.72, if applicable.~~

~~b. Each Reportable Event shall be reviewed by the Plant Review Committee and submitted to the Chairperson of the Safety Audit and Review Committee and the corporate officer responsible for overall plant nuclear safety.~~

~~c. Submit reports of Reportable Events pursuant to the requirements of Specification 5.9.2.~~

5.7 Safety Limit Violation

5.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT SHUTDOWN within 1 hour.

~~b. The Safety Limit Violations shall be reported to the corporate officer responsible for overall plant nuclear safety and the Chairperson of the Safety Audit and Review Committee (SARC) within 24 hours.~~

~~c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Plant Review Committee. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.~~

~~d. The Safety Limit Violation Report shall be submitted to the Chairperson of the Safety Audit and Review Committee and the corporate officer responsible for overall plant nuclear safety within 14 days of the violation.~~

5.8 Procedures

5.8.1 Written procedures and administrative policies shall be established, implemented and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Fire Protection Program implementation; and

d. All programs specified in Specification 5.11 through 5.21.

5.8.2 Temporary changes to procedures of 5.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant supervisory staff, at least one of whom holds a Senior Reactor Operator's License.

5.8 Procedures (Continued)

c. The change is documented, reviewed by a qualified reviewer and approved by either the plant manager or the department head designated by Administrative Controls Standing Orders as the responsible department head for that procedure within 14 days of implementation.

5.8.3 Written procedures shall be implemented which govern the selection of fuel assemblies to be placed in Region 2 of the spent fuel racks (Technical Specification 2.8). These procedures shall require an independent verification of initial enrichment requirements and fuel burnup calculations for a fuel bundle to assure the "acceptance" criteria for placement in Region 2 are met. This independent verification shall be performed by individuals or groups other than those who performed the initial acceptance criteria assessment, but who may be from the same organization.

5.9 Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the appropriate NRC Regional Office unless otherwise noted.

5.9.1 Routine Reports

a. Not Used ~~Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufacture by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the USAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.~~

~~Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.~~

5.9 Reporting Requirements (Continued)

b. Annual Occupational Exposure Report. An annual occupational exposure report shall be submitted on or before April 30 of each year. The report shall consist of a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,[3/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling outages. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, with a copy to the appropriate Regional Office, no later than the fifteenth of each month following the calendar month covered by the report.

5.9.2 Not Used ~~Reportable Event~~

~~A Licensee Event Report (LER) shall be submitted to the U.S. Nuclear Regulatory Commission for any event meeting the requirements of 10 CFR Part 50.73.~~

5.9.3 Special Reports

Special reports shall be submitted to the appropriate NRC Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification where appropriate:

a. In-service inspection report, reference 3.3.
b. Tendon surveillance, reference 5.21.
c. Containment structural tests, reference 3.5.
d. DELETED
e. DELETED
f. DELETED
g. Materials radiation surveillance specimens reports, reference 3.3.
h. DELETED
i. Post-accident monitoring instrumentation, reference 2.21
j. Electrical systems, reference 2.7(2).

3/ This tabulation supplements the requirements of § 20.2206 of 10 CFR Part 20.

5.9 Reporting Requirements (Continued)

5.9.4 Unique Reporting Requirements

a. Annual Radioactive Effluent Release Report

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year of operation shall be submitted before May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be 1) consistent with the objectives outlined in the ODCM and PCP, and 2) in conformance with 10 CFR 50.36a. and Section IV.B.1 of Appendix I to 10 CFR 50.

b. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Section IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR 50.

c. Not Used ~~Fire Protection Program Deficiency Report~~

~~Deficiencies in the Fire Protection Program described in the Updated Safety Analysis Report which meet the reportability criteria of 10 CFR 50.73 shall be reported pursuant to Section 5.9.2 of the Technical Specifications.~~

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" approved version as specified in the COLR.

   2. OPPD-NA-8302-P-A, "Neutronics Design Methods and Verification", approved version as specified in the COLR.

ATTACHMENT 3

Proposed Technical Specification Pages

TABLE 3-3 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS

Channel Description	Surveillance Function	Frequency	Surveillance Method
14. Not Used			
15. Reactor Coolant System Flow	a. Check	R[1]	a. Calculation of reactor coolant flow rate.
16. Pressurizer Pressure	a. Check	S	a. CHANNEL CHECK
17. Reactor Coolant Inlet Temperature	a. Check	S	a. CHANNEL CHECK
18. Low-Temperature Set-point Power-Operated Relief Valves	a. Test	PM	a. CHANNEL FUNCTIONAL TEST (excluding actuation)
	b. Calibrate	R	b. CHANNEL CALIBRATION

(1) Required to be performed within 24 hours after ≥95.00% reactor thermal power following power escalation.

TABLE 3-5 (continued)

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

	Test	Frequency	USAR Section Reference
10a. (continued)	2. Laboratory Testing** Verify, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows methyliodide penetration less than 0.175% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.	On a refueling frequency or every 720 hours of system operation or after any structural maintenance on the HEPA filter or charcoal adsorber housing or following significant painting, fire or chemical release in a ventilation zone communicating with the system.	
	3. Overall System Operation		
	a. Each circuit shall be operated.	Ten hours every month.	
	b. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 9 inches of water at system design flow rate.	R	
	c. Fan shall be shown to operate within ± 10% design flow.	R	
	4. Automatic and manual initiation of the system shall be demonstrated.	R	

**Tests shall be performed in accordance with applicable section(s) of ANSI N510-1980.

3.0 **SURVEILLANCE REQUIREMENTS**
3.17 Steam Generator Tubes (Continued)

(iii) Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

3.0 **SURVEILLANCE REQUIREMENTS**
3.17 Steam Generator Tubes (Continued)

Basis

The surveillance requirements for inspection of the steam generator tubes and tube sleeves ensure that the structural integrity of this portion of the RCS will be maintained. The program for in-service inspection of the steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1, dated July 1975. The program for in-service inspection of steam generator tube sleeves is based on a modification of EPRI PWR Steam Generator Examination Guidelines, Revision 5, Dated September 1997. In-service inspection of steam generator tubing and tube sleeves is essential in order to maintain surveillance of the conditions of the tubes and sleeves in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or in-service conditions that lead to corrosion.

In-service inspection of steam generator tubing and tube sleeves also provides a means of characterizing the nature and cause of any tube or sleeve degradation so that corrective measures can be taken.

Tubes with defects may be repaired by a Combustion Engineering, Inc. Leak Tight Sleeve. The technical bases for sleeving repair are described in the Proprietary Combustion Engineering, Inc. Report CEN-630-P, Revision 02, "Repair of 3/4" O.D. Steam Generator Tubes Using Leak Tight Sleeves," June 1997.

Whenever the results of any steam generator tubing in-service inspection fall into Category C-3, these results will be promptly reported to the Commission prior to the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

5.0 ADMINISTRATIVE CONTROLS

5.4 Training

5.4.1 A retraining and replacement training program for the plant staff shall be maintained under the direction of the Manager - Training and shall meet or exceed the requirements of Section 6 of ANSI/ANS 3.1-1993, as modified by Regulatory Guide 1.8, Revision 3, dated May 2000 and 10 CFR Part 55.

5.5 Not Used

5.6 Not Used

5.7 Safety Limit Violation

5.7.1 The following action shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT SHUTDOWN within 1 hour.

5.8 Procedures

5.8.1 Written procedures and administrative policies shall be established, implemented and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Fire Protection Program implementation; and

d. All programs specified in Specification 5.11 through 5.21.

5.8.2 Temporary changes to procedures of 5.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant supervisory staff, at least one of whom holds a Senior Reactor Operator's License.

5.8 Procedures (Continued)

c. The change is documented, reviewed by a qualified reviewer and approved by either the plant manager or the department head designated by Administrative Controls Standing Orders as the responsible department head for that procedure within 14 days of implementation.

5.8.3 Written procedures shall be implemented which govern the selection of fuel assemblies to be placed in Region 2 of the spent fuel racks (Technical Specification 2.8). These procedures shall require an independent verification of initial enrichment requirements and fuel burnup calculations for a fuel bundle to assure the "acceptance" criteria for placement in Region 2 are met. This independent verification shall be performed by individuals or groups other than those who performed the initial acceptance criteria assessment, but who may be from the same organization.

5.9 Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the appropriate NRC Regional Office unless otherwise noted.

5.9.1 Routine Reports

a. Not Used

5.9 Reporting Requirements (Continued)

b. Annual Occupational Exposure Report. An annual occupational exposure report shall be submitted on or before April 30 of each year. The report shall consist of a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,[3/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling outages. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, with a copy to the appropriate Regional Office, no later than the fifteenth of each month following the calendar month covered by the report.

5.9.2 Not Used

5.9.3 Special Reports

Special reports shall be submitted to the appropriate NRC Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification where appropriate:

a. In-service inspection report, reference 3.3.
b. Tendon surveillance, reference 5.21.
c. Containment structural tests, reference 3.5.
d. DELETED
e. DELETED
f. DELETED
g. Materials radiation surveillance specimens reports, reference 3.3.
h. DELETED
i. Post-accident monitoring instrumentation, reference 2.21
j. Electrical systems, reference 2.7(2).

3/ This tabulation supplements the requirements of § 20.2206 of 10 CFR Part 20.

5.9 Reporting Requirements (Continued)

5.9.4 Unique Reporting Requirements

a. Annual Radioactive Effluent Release Report

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year of operation shall be submitted before May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be 1) consistent with the objectives outlined in the ODCM and PCP, and 2) in conformance with 10 CFR 50.36a. and Section IV.B.1 of Appendix I to 10 CFR 50.

b. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Section IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR 50.

c. Not Used

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" approved version as specified in the COLR.

   2. OPPD-NA-8302-P-A, "Neutronics Design Methods and Verification", approved version as specified in the COLR.