Omaha Public Power District
1946 - 2006

Fort Calhoun Station
P.O. Box 550
Fort Calhoun, NE 68023

July 2, 2008
LIC-08-0078

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555-0001

Reference: Docket No. 50-285

SUBJECT: Fort Calhoun Station, Unit No. 1, License Amendment Request, "Administrative Revisions to Fort Calhoun Station, Unit No. 1, Technical Specifications"

Pursuant to 10 CFR 50.90, the Omaha Public Power District (OPPD) submits the attached administrative license amendment request (LAR) proposing to correct several typographical errors and make administrative clarifications to the Fort Calhoun Station, Unit No. 1 (FCS) Technical Specifications (TS).

The enclosure contains a description of the proposed changes, the supporting technical analyses, and the significant hazards consideration determination. Attachment 1 provides the existing TS pages marked-up to show the proposed changes. Attachment 2 provides the retyped (clean) TS pages.

OPPD has determined that this LAR does not involve a significant hazard consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

There are no regulatory commitments associated with this proposed change.

OPPD requests approval by June 1, 2009 with a 6-month implementation period.

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

If you should have any questions regarding this submittal, please contact Mr. Bill R. Hansher at (402) 533-6894.

I declare under penalty of perjury that the foregoing is true and correct. (Executed on July 2, 2008.)

R. P. Clemens
Division Manager
Nuclear Engineering

RPC/MLE/mle

Enclosure: OPPD's Evaluation of the Proposed Changes

c: Director of Consumer Health Services, Department of Regulation and Licensure, Nebraska Health and Human Services, State of Nebraska

Omaha Public Power District's Evaluation
for
Amendment of Operating License

Administrative Revisions

ATTACHMENTS:

1. Technical Specification Pages (Mark-up of Proposed Changes)
2. Proposed Technical Specification Pages (Clean)

1. SUMMARY DESCRIPTION

The Omaha Public Power District (OPPD) hereby requests an amendment to Fort Calhoun Station (FCS), Unit No. 1, Renewed Facility Operating License No. DPR-40 to correct typographical errors and make administrative clarifications to the FCS Technical Specifications (TS). As described below, the proposed changes are administrative in nature.

2. DETAILED DESCRIPTION

TS 2.1.1(1)

The FCS TS do not contain a Table 1.1; reactor coolant flow is found in Table 2-11. TS 2.1.1(1) is revised to correct this typographical error.

TS 2.1.1(12)(a)

TS 2.1.1(12)(a) incorrectly references a Table 2.9 but there is no such table. TS 2.1.1(12)(a) is revised to reference the correct table, which is Table 2-9.

TS 2.1.2c

TS 2.1.2c states that the heatup rate of the pressurizer shall not exceed "I"00 degrees Fahrenheit in any one hour period. The letter "I" is replaced with a "1" to correct this typographical error.

TS 2.10.4(1)(d)

Figure 1-2 was relocated to the Core Operating Limits Report (COLR) in Amendment 170 (Reference 6.1). However, the license amendment request (LAR) (Reference 6.2) that proposed the relocation overlooked a reference to Figure 1-2 in TS 2.10.4(1)(d) and thus the reference remained in the TS. OPPD proposes to replace the reference to Figure 1-2 with a reference to the applicable figure in the COLR, which is Figure 7, "Axial Power Distribution LSSS for 4 Pump Operation." This is the figure that was relocated to the COLR by Amendment 170.

The last paragraph of TS 2.10.4(1)(d) contains an extraneous period in the reference to TS 2.10.4(1). OPPD proposes to delete this period.

TS 2.13, Table 2-11, RPS Limiting Safety System Settings

OPPD proposes to make several administrative clarifications to Table 2-11, which lists the reactor protective system (RPS) trip setpoints. First, OPPD proposes to rename the column titled "Trip Setpoints" to "Allowable Value," which is consistent with the title of the table and with Improved Standard Technical Specifications (ISTS) (Reference 6.3).

Second, OPPD proposes to place a greater than or equal to sign in front of the trip setpoint for low steam generator water level. The designation of this setpoint at a specified value is inconsistent with the other RPS setpoints of Table 2-11. Third, OPPD proposes to correct the title of the Axial Power Distribution Figure, which is "Axial Power Distribution LSSS for 4 Pump Operation." Finally, OPPD proposes to make a grammatical correction to Footnote F by inserting a comma to improve its readability.

TS 2.14, Table 2-1, Engineered Safety Features System Initiation Instrument Setting Limits

OPPD proposes to make two administrative clarifications to Table 2-1, which lists the engineered safety features (ESF) system trip setting limits. First, OPPD proposes to rename the column titled "Setting Limit" to "Allowable Value," which is consistent with the title of the table and with ISTS. Second, OPPD proposes to correct the setting for Item 6.a (4.16 KV Emergency Bus Low Voltage), which contains formatting errors in the superscript for time delay. The -20.8 (volts) pertains to the trip setting for loss of voltage. It does not pertain to the time delay setting. Thus, OPPD proposes to relocate the -20.8 to the line above its current position to clarify that it applies to the loss of voltage trip. For added clarity, OPPD proposes to relocate the superscript (4) away from the numeric value for time delay such that it follows the word "seconds," which is a more appropriate location for the footnote reference.

TS 5.8.1d

OPPD proposes to revise TS 5.8.1d to reflect the addition of the "Diesel Fuel Oil Testing Program," the "Steam Generator (SG) Program," and the "Control Room Habitability Program." These programs are described in TS 5.22, TS 5.23, and TS 5.24, respectively; therefore, TS 5.8.1d is revised to apply through TS 5.24.

TS 5.19

OPPD proposes to revise TS 5.19 to correct a punctuation error, renumber its provisions for clarity, eliminate a one-time exception stating that the Type A test can be performed no later than November 2008, and correct a typographical error in the acceptance criteria for containment purge valve leakage rate.

3. TECHNICAL EVALUATION

TS 2.1.1(1)

The objective of TS 2.1.1 is to specify certain conditions of the reactor coolant system components. As such, TS 2.1.1(1) states that all four (4) reactor coolant pumps shall be in operation. The exceptions for TS 2.1.1(1) allow the limitations of this specification to be suspended during low power physics testing provided the reactor coolant flow requirements of Table 1.1 are met. However, the FCS TS do not contain a Table 1.1.

Prior to Amendment 252 (Reference 6.4), the FCS TS contained reactor coolant flow requirements in Table 1-1. Amendment 252 relocated Table 1-1 to TS 2.13 and renamed it Table 2-11. Thus, reactor coolant flow requirements are now found in Table 2-11 and OPPD is correcting this typographical error.

TS 2.1.1(12)(a)

TS 2.1.1(12)(a) incorrectly references a Table 2.9 but there is no such table. The correct table is Table 2-9 and TS 2.1.1(12)(a) is revised to correct this typographical error.

TS 2.1.2c

TS 2.1.2c states that the heatup rate of the pressurizer shall not exceed "I"00 degrees Fahrenheit in any one hour period. The letter "I" is replaced with a "1" to correct this typographical error.

TS 2.10.4(1)(d)

In a letter dated April 7, 1995, (Reference 6.2), OPPD submitted an LAR proposing to delete Figure 1-2 titled "Axial Power Distribution LSSS for 4 Pump Operation" from the FCS TS and relocate it to the COLR in accordance with Generic Letter 88-16 (Reference 6.5). However, the LAR overlooked a reference to Figure 1-2 contained in TS 2.10.4(1)(d) and thus the reference, which should have been deleted, was not. The NRC approved the relocation of Figure 1-2 to the COLR in Amendment 170 (Reference 6.1). To correct this oversight, the reference to Figure 1-2 is deleted and replaced with a reference to the Axial Power Distribution LSSS for 4 Pump Operation Figure, which is Figure 7 in the COLR. This change is consistent with the changes requested by Reference 6.2 and approved by the NRC in Amendment 170 and thus is administrative in nature to remove an obsolete reference.

Deletion of an extraneous period in the reference to Specification 2.10.4(1) corrects a typographical error and thus is also administrative in nature.

TS 2.13, Table 2-11, RPS Limiting Safety System Settings

Table 2-11 lists each reactor trip and its associated setting. The title of the column containing the setting (*Trip Setpoints*) implies that it contains a specific value for each reactor trip. However, the presence of greater than (or less than) or equal to signs in front of the settings indicates that a range of values is allowed. Renaming this column to "Allowable Value" is a more accurate description of the column and is consistent with the title of Table 2-11. This change also makes Table 2-11 more similar to Table 3.3.1-1, *Reactor Protective System Instrumentation* of ISTS (Reference 6.3).

The RPS generates a reactor trip signal at 31.2% narrow-range steam generator water level, which is an abnormally low level. A water level this low indicates a loss of steam generator secondary water inventory, which if not corrected could result in the loss of the ability to remove heat from the reactor coolant system. Since the purpose of the column is to list the allowable value of the various trip settings, it is more accurate to precede the low steam generator water level allowable value (31.2%) with a greater than or equal to sign. This is consistent with the designation of allowable values in Table 3.3.1-1 of ISTS, as well as with the designation of other RPS settings in Table 2-11. This change does not allow less conservative settings (i.e., less than 31.2%) but would allow settings greater than 31.2%, which would provide additional margin for protecting steam generator water levels.

The change proposed to the title of the Axial Power Distribution LSSS for 4 Pump Operation Figure is consistent with the title of the applicable figure in the COLR (Figure 7) and with the changes made to TS 2.10.4(1)(d) as described above.

Insertion of a comma into Footnote F is a grammatical correction that improves readability of the footnote when bypassing the low reactor coolant flow and thermal margin/low pressure trips during low power physics testing.

These changes are administrative in nature, are consistent with ISTS and with the designations of other settings in Table 2-11, or otherwise improve accuracy or readability.

<u>TS 2.14, Table 2-1, Engineered Safety Features System Initiation Instrument Setting Limits</u>

Table 2-1 lists each ESF trip setting limit. Renaming the "Setting Limit" column to "Allowable Value" is a more accurate description of the column. This title is consistent with the title of Table 2-1 and with the change proposed for Table 2-11 as shown above. This change also makes Table 2-1 more similar to Table 3.3.4-1, *Engineered Safety Features Actuation System Instrumentation* of ISTS (Reference 6.3).

A review of regulatory correspondence associated with amendments to Table 2-1 determined that the loss of voltage setting limit error has been manifesting itself since Amendment 157 (Reference 6.6). Prior to Amendment 157, the -20.8 (volts) was shown immediately below the +104 volts near the superscript for the reference to Footnote (4). The Reference 6.7 LAR and a supplement thereto (Reference 6.8), associated with Amendment 157 were reviewed and did not request changes to Item 6. OPPD surmises that due to its location near the footnote reference, a typographical error related to the issuance of Amendment 157 caused the -20.8 to end up on the line below in the superscript to Footnote (4). This footnote is associated with the time delay for loss of voltage.

The proposed change corrects this discrepancy by placing both loss of voltage limits on the same line and clarifies the time delay setting by relocating the superscript (4) away from the numeric value to follow the word "seconds," which is a more appropriate location for the footnote.

These changes are limited to the correction of typographical errors and the relocation of text for legibility, and thus are administrative in nature.

TS 5.8.1d

TS 5.8.1d lists additional programs for which written procedures and administrative policies are required. In recent years, several new programs have been added to TS 5.0, "Administrative Controls" and the need to update TS 5.8.1d to reflect the addition of these programs was overlooked. Specifically, the new programs are:

TS 5.22, Diesel Fuel Oil Testing Program,
TS 5.23, Steam Generator (SG) Program, and
TS 5.24, Control Room Habitability Program.

Accordingly, TS 5.8.1d is revised to apply through TS 5.24. OPPD has instituted an administrative control to evaluate TS 5.8.1d whenever new programs are added to TS 5.0. Since the evaluation occurs prior to submitting an LAR to the NRC, this error should not recur.

TS 5.19

OPPD proposes the following changes to TS 5.19.

The first sentence of TS 5.19 is revised to correct a punctuation error where a period, which should be a comma, separates the citation of 10 CFR 50, Appendix J, Option B.

In order to set the provisions of TS 5.19 apart from each other, OPPD proposes to format TS 5.19 using an alpha-numeric numbering scheme that is similar to the format of TS 5.5.16, *Containment Leakage Rate Testing Program* of ISTS (Reference 6.3). Because this change is limited to segregating the provisions of TS 5.19 for clarity, it is administrative in nature.

OPPD proposes to delete Exception (4), which was added to TS 5.19 by Amendment 220 (Reference 6.9). In Reference 6.10 as supplemented by References 6.11, and 6.12, OPPD had requested a one-time five-year extension to the ten-year test interval for the Type A, integrated leak rate test (ILRT). OPPD successfully completed the Type A test during the 2008 refueling outage, which ended in June 2008. With the deletion of Exception (4), OPPD reverts to a 10-year test interval based on performance history in accordance with Option B of 10 CFR 50, Appendix J as approved by the NRC in

Amendment 185 (Reference 6.13). The proposed change is administrative in nature as the exception no longer serves a purpose as the Type A test has been completed.

Finally, OPPD proposes to correct a typographical error in the leakage rate of the containment purge valves. In Reference 6.14, OPPD submitted an LAR to implement Option B of 10 CFR 50, Appendix J to allow the frequency of conducting the Type A ILRT and local leak rate testing (Type B and C) to be based on component performance. The LAR approved in Amendment 185 (Reference 6.13) resulted in the establishment of the Containment Leakage Rate Testing Program of TS 5.19.

The markup of TS 5.19 contained in the Reference 6.14 LAR specified that the leakage rate acceptance criteria for the containment purge valves is < 18,000 (eighteen thousand) standard cubic centimeters per minute (SCCM) when tested at ≥ Pa (containment design accident pressure). However, the TS page issued with Amendment 185 inadvertently changed the comma indicating thousand to a decimal point. This error was not discovered at the time the amendment was issued. TS 5.19 is revised to correct this typographical error and as such is administrative in nature.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.9 *Completeness and accuracy of information.* 10 CFR 50.9 states, "Information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee shall be complete and accurate in all material respects."

The requested changes revise the FCS TS to meet the requirements of this regulation.

10 CFR 50.36 Technical Specifications. 10 CFR 50.36(d) states:

Technical specifications will include items in the following categories:

(1) Safety limits, limiting safety system settings, and limiting control settings. (i)(A) Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down. . ."

The requested changes to Table 2-11 ensure that the FCS TS continue to meet the requirements of this regulation.

General Design Criteria

FCS was licensed for construction prior to May 21, 1971 and at that time committed to the draft General Design Criteria (GDC). Compliance with the draft GDC is documented in Appendix G of the FCS Updated Safety Analysis Report (USAR). The requested changes to the FCS TS are administrative in nature and continue to comply with USAR Appendix G.

4.2 Precedent

Because these changes are administrative in nature, the citing of precedent would add little value and therefore, none is provided. There are numerous administrative amendments correcting typographical errors and making editorial changes such as these.

4.3 Significant Hazards Consideration

The Fort Calhoun Station (FCS), Unit No. 1 Technical Specifications (TS) are modified to correct typographical and administrative errors, or make clarifications that more accurately reflect TS requirements.

The Omaha Public Power District (OPPD) 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 correct typographical and administrative errors, or make clarifications that more accurately reflect TS requirements. Administrative and editorial changes such as these are not an initiator of any accident previously evaluated. As a result, the probability of an accident previously evaluated is not affected. The consequences of an accident with the incorporation of these administrative and editorial changes are no different than the consequences of the same accident without these changes. As a result, the consequences of an accident previously evaluated are not affected by these changes.

   The proposed changes do not alter or prevent the ability of structures, systems, and components from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release

assumptions used in evaluating the radiological consequences of an accident previously evaluated.

Further, the proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with the safety analysis assumptions and resultant consequences. Therefore, the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed changes do not alter any assumptions made in the safety analysis. Therefore, the proposed changes do not create the possibility of a new or different accident from any accident previously evaluated.

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

   Response: No.

   The proposed changes consist of administrative and editorial changes to correct typographical or administrative errors and oversights or clarify the meaning of the TS. The changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by these changes. The proposed changes will not result in plant operation in a configuration outside of the design basis. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

4.4 CONCLUSIONS

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. Therefore, OPPD concludes that the proposed changes present 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.

5.0 ENVIRONMENTAL CONSIDERATION

OPPD has evaluated the proposed amendment and has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0. REFERENCES

6.1 Letter from NRC (S. D. Bloom) to OPPD (T. L. Patterson), "Fort Calhoun Station. Unit No. 1 - Amendment No. 170 to Facility Operating License No. DPR-40 (TAC No. M92057)," dated September 1, 1995 (NRC-95-0184)

6.2 Letter from OPPD (W. G. Gates) to NRC (Document Control Desk), "Application for Amendment of Operating License," dated April 7, 1995 (LIC-95-0083)

6.3 NUREG-1432, Rev 3.0, "Standard Technical Specifications Combustion Engineering Plants," published June 2004

6.4 Letter from NRC (M. T. Markley) to OPPD (D. J. Bannister, "Fort Calhoun Station, Unit No. 1 – Issuance of Amendment [252] Re: Adoption of Technical Specification Task Force (TSTF) 445-A, Revision 1, (TAC No. MD6808)," dated February 4, 2008 (NRC-08-0017)

6.5 Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 3, 1988

6.6 Letter from NRC (S. D. Bloom) to OPPD (T. L. Patterson), "Fort Calhoun Station, Unit 1 – Amendment No. 157 to Facility Operating License No. DPR-40 (TAC No. M86928," dated November 22, 1993 (NRC-93-0397)

6.7 Letter from OPPD (W. G. Gates) to NRC (Document Control Desk), "Application for Amendment of Operating License," dated June 17, 1993 (LIC-93-0159)

6.8 Letter from OPPD (T. L. Patterson) to NRC (Document Control Desk), "Application for Amendment of Operating License (TAC No. 86928)," dated October 8, 1993 (LIC-93-0256)

6.9 Letter from NRC (A. B. Wang) to OPPD (R. T. Ridenoure), "Fort Calhoun Station, Unit No. 1 – Issuance of Amendment [220] (TAC No. MB6473)," dated August 15, 2003 (NRC-03-0156)

6.0. REFERENCES (Continued)

6.10 Letter from OPPD (D. J. Bannister) to NRC (Document Control Desk), "Fort Calhoun Station, Unit No. 1, License Amendment Request, Risk-Informed One-Time Increase in Integrated Leak Rate Test Surveillance Interval," dated October 8, 2002 (LIC-02-0198)

6.11 Letter from OPPD (R. T. Ridenoure) to NRC (Document Control Desk), "Response to Request for Additional Information, Integrated Leak Rate Testing Surveillance Interval Amendment Request (TAC No. MB6473)," dated April 11, 2003 (LIC-03-0055)

6.12 Letter from OPPD (R. T. Ridenoure) to NRC (Document Control Desk), "Supplemental Response to Request for Additional Information, Integrated Leak Rate Testing Surveillance Interval Amendment Request (TAC No. MB6473)," dated May 21, 2003 (LIC-03-0079)

6.13 Letter from NRC (L. R. Wharton) to OPPD (S. K. Gambhir), "Fort Calhoun Station, Unit No. 1 - Amendment No. 185 to Facility Operating License No. DPR-40 (TAC No. M99546)," dated March 23, 1998 (NRC-98-0051)

6.14 Letter from OPPD (W. G. Gates) to NRC (Document Control Desk), "Application for Amendment of Operating License," dated July 25, 1997 (LIC-97-0124)

Technical Specification Pages
(Mark-up of Proposed Changes)

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System

2.1.1 Operable Components

Applicability

Applies to the operable status of the reactor coolant system components.

Objective

To specify certain conditions of the reactor coolant system components.

Specifications

Limiting conditions for operation are as follows:

(1) Reactor Critical

All four (4) reactor coolant pumps shall be in operation.

Exceptions

The limitations of this specification may be suspended during the performance of physics tests provided the power level is $\leq 10^{-1}$% of rated power and the flow requirements of Table ~~1.12-11~~ No. 2 are met.

(2) Hot Shutdown or 350°F $\leq T_{cold} \leq$ 515°F

(a) The reactor coolant loops listed below shall be operable:

(i) Reactor coolant loop 1 and at least one associated reactor coolant pump.

(ii) Reactor coolant loop 2 and at least one associated reactor coolant pump.

(b) At least one of the above reactor coolant loops shall be in operation.

Exceptions

All reactor coolant pumps may be de-energized for up to one hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.1 Operable Components (Continued)

(12) Reactor Coolant System Pressure Isolation Valves

(a) The integrity of all pressure isolation valves listed in Table ~~2.9~~ 2-9 shall be demonstrated, except as specified in (b). Valve leakage shall not exceed the amounts indicated.

(b) In the event that the integrity of any pressure isolation valve specified in Table 2-9 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a nonfunctional valve are in and remain in the mode corresponding to the isolated condition. Manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supply deenergized.

(c) If Specifications (a) and (b) above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

Basis

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation and maintain DNBR above the minimum DNBR limit during all normal operations and anticipated transients.

When Specification 2.1.1(2) is applicable, the reactor coolant pumps (RCPs) are used to provide forced circulation heat removal during heatup and cooldown. Under these conditions, decay heat removal requirements are low enough that a single reactor coolant system (RCS) loop with one RCP is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to provide redundant paths for decay heat removal. Only one RCP needs to be OPERABLE to declare the associated RCS loop OPERABLE. Reactor coolant natural circulation is not normally used but is sufficient for core cooling. However, natural circulation does not provide turbulent flow conditions. Therefore, boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be assured.

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate

Applicability

Applies to the temperature change rates and pressure of the Reactor Coolant System (RCS).

Objective

To specify limiting conditions of the reactor coolant system heatup and cooldown rates.

Specification

The combination of RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR and as designated below:

a. Allowable combinations of pressure and temperature (T_c) for a specific heatup rate shall be below and to the right of the applicable limit lines as shown on the pressure and temperature (P-T) limit Figure(s) in the PTLR.

b. Allowable combinations of pressure and temperature (T_c) for a specific cooldown rate shall be below and to the right of the applicable limit lines as shown on the P-T limit Figure(s) in the PTLR.

c. The heatup rate of the pressurizer shall not exceed ~~1~~100°F in any one hour period.

d. The cooldown rate of the pressurizer shall not exceed 200°F in any one hour period.

Required Actions

(1) When any of the above limits are exceeded, the following corrective actions shall be taken:

1. Immediately initiate action to restore the temperature or pressure to within the limit.

2. Perform an analysis to determine the effects of the out of limit condition on the fracture toughness properties of the reactor coolant system.

3. Determine that the reactor coolant system remains acceptable for continued operation or be in cold shutdown within 36 hours.

(2) Before the radiation exposure of the reactor vessel exceeds the exposure for which they apply, the P-T limit Figure(s) shown in the PTLR shall be updated in accordance with the following criteria and procedures:

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

(c) When the linear heat rate is continuously monitored by the excore detectors, withdraw the full length CEA's beyond the long term insertion limits of Specification 2.10.2(7) and maintain the Axial Shape Index, Y_I within the limits of Limiting Condition for Operations for the Excore Monitoring of LHR Figure provided in the COLR. If the linear heat rate is exceeding its limits as determined by the Axial Shape Index, Y_I, being outside the limits of the Limiting Condition for Operation for Excore Monitoring of LHR Figure provided in the COLR:

(i) Restore the reactor power and Axial Shape Index, Y_I, to within the limits of the Limiting Condition for Operations for Excore Monitoring of LHR Figure provided in the COLR within 2 hours, or

(ii) Be in at least hot standby within the next 6 hours.

(d) When calibration of the ex-core detectors has not been accomplished within the previous 30 equivalent full power days, then:

(i) reduce the axial power distribution monitoring trip setpoints ~~(Figure 1-2)~~ as shown in the Axial Power Distribution for 4 Pump Operation LSSS Figure provided in the COLR by 0.03 ASI units; and

(ii) reduce the axial power distribution monitoring Limiting Condition for Operations (LCO for Excore Monitoring of LHR and LCO for DNB Monitoring Figures provided in the COLR) by 0.03 ASI units.

When calibration of the ex-core detectors has not been accomplished within the previous 200 equivalent full power days, the power shall be limited to less than that corresponding to 75% of the peak linear heat rate permitted by Specification ~~2.10.4.(1)~~2.10.4(1).

(2) Total Integrated Radial Peaking Factor

The calculated value of F_R^T defined by $F_R^T = F_R (1+T_q)$ shall be within the limit provided in the COLR. F_R is determined from a power distribution map with no non-trippable CEA's inserted and with all full length CEA's at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. The azimuthal tilt, T_q, is the measured value of T_q at the time F_R is determined.

2.0 LIMITING CONDITIONS FOR OPERATION

2.13 Limiting Safety System Settings, Reactor Protective System (continued)

TABLE 2-11

RPS LIMITING SAFETY SYSTEM SETTINGS

No.	Reactor Trip	Allowable Value ~~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 LSSS for 4 Pump Operation Figure provided in the COLR)
9	Steam Generator Differential Pressure	≤ 135 psid

A Setpoint cannot be set greater than 10% above measured power whenever reactor power is greater than 10% of rated power.

B May be bypassed below 10^{-4}% power.

C May be bypassed below 600 psia.

D Bypass allowed for containment leak test.

E Inhibited below 15% power.

F For physics testing at power levels less than 10^{-1}% of rated power, the low reactor coolant flow and thermal margin/low pressure trips may be bypassed until their reset points are exceeded if automatic bypass removal of 10^{-1}% of rated power is operable.

TABLE 2-1

ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENT SETTING LIMITS

Functional Unit	Channel	Allowable Value~~Setting Limit~~
1. High Containment Pressure	a. Safety Injection b. Containment Spray (3) c. Containment Isolation d. Containment Air Cooler DBA Mode e. Steam Generator Isolation	≤ 5 psig
2. Pressurizer Low/Low Pressure	a. Safety Injection b. Containment Spray (3) c. Containment Isolation d. Containment Air Cooler DBA Mode	≥ 1600 psia (1)
3. Containment High Radiation	Containment Ventilation Isolation	In accordance with the applicable Chemistry Manual calibration procedure
4. Low Steam Generator Pressure	a. Steam Line Isolation b. Auxiliary Feedwater Actuation c. Containment Spray (3)	≥ 500 psia (2) ≥ 466.7 psia ≥ 500 psia (2)
5. SIRW Low Level Switches	Recirculation Actuation	16 inches +0, -2 in. above tank bottom

TABLE 2-1 (continued)

ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENT SETTING LIMITS

Functional Unit	Channel	Allowable Value~~Setting Limit~~	
6. 4.16 KV Emergency Bus Low Voltage	a. Loss of Voltage	(2995.2 + 104, -20.8) volts ≤ 5.9 (4)~~-20.8~~-seconds(4)	} Trip }
	b. Degraded Voltage		
	i) Bus 1A3 Side	≥ 3988.8 volts (4.8 ± .5) seconds	} Trip }
	ii) Bus 1A4 Side	≥ 3990.6 volts (4.8 ± .5) seconds	} Trip }
7. Low Steam Generator Water Level	Auxiliary Feedwater Actuation	≥ 28.2% of wide range tap span	
8. High Steam Generator Delta Pressure	Auxiliary Feedwater Actuation	≤ 119.7 psid	

(1) May be bypassed below 1700 psia and is automatically reinstated prior to exceeding 1700 psia.
(2) May be bypassed below 600 psia and is automatically reinstated prior to exceeding 600 psia.
(3) Simultaneous containment high pressure, pressurizer low/low pressure, and steam generator low pressure.
(4) Applicable for bus voltage ≤ 2995.2 - 20.8 volts only. (For voltage ≥ (2995.2 - 20.8) volts, time delay shall be > 5.9 seconds).

5.0 ADMINISTRATIVE CONTROLS

5.7 Not used.

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~~24.

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.

5.0 ADMINISTRATIVE CONTROLS

5.18 Process Control Program (PCP) (Continued)

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program. This documentation shall contain:

1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2. A determination that the change will maintain the overall conformance of the solidified waste program to existing requirements of federal, state, or other applicable regulations.

b. Shall become effective after the review and acceptance by the Plant Review Committee and the approval of the plant manager.

c. Temporary changes to the PCP may be made in accordance with Technical Specification 5.8.2.

d. Shall be submitted to the Nuclear Regulatory Commission in the form of a complete, legible copy of the entire PCP as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the PCP was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed and shall indicate the date (e.g., month/year) the change was implemented.

5.19 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995," as modified by the following exceptions:

(1) If the Personnel Air Lock (PAL) is opened during periods when containment integrity is not required, the PAL door seals shall be tested at the end of such periods and the entire PAL shall be tested within 14 days after RCS temperature T_{cold} > 210°F.

(2) Type A tests may be deferred for penetrations of the steel pressure retaining boundary where the nominal diameter does not exceed one inch.

(3) Elapsed time between consecutive Type A tests used to determine performance shall be at least 24 months or refueling interval.

~~(4) The first Type A test performed after the November 1993 Type A test shall be no later than November 2008.~~

b. The containment design accident pressure (P_a) is 60 psig.

5.0 ADMINISTRATIVE CONTROLS

5.19 Containment Leakage Rate Testing Program (Continued)

c. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

d. Leakage Rate acceptance criteria are:

~~a~~1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ Maximum Pathway Leakage Rate (MXPLR) for Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

~~b~~2. Personnel Air Lock testing acceptance criteria are:

(~~1~~i) Overall Personnel Air Lock leakage is $\leq 0.1\ L_a$ when tested at $\geq P_a$.

(~~2~~ii) For each PAL door, seal leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq$ 5.0 psig.

~~c~~e. Containment Purge Valve (PCV-742A/B/C/D) testing acceptance criterion is:

For each Containment Purge Valve, leakage rate is < ~~18.000~~18,000 SCCM when tested at $\geq P_a$.

~~d~~f. If at any time when containment integrity is required and the total Type B and C measured leakage rate exceeds 0.60 L_a Minimum Pathway Leakage Rate (MNLPR), repairs shall be initiated immediately. If repairs and retesting fail to demonstrate conformance to this acceptance criteria within 48 hours, then containment shall be declared inoperable.

g. The provisions of Specification 3.0.1 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

h. The provisions of Specifications 3.0.4 and 3.0.5 are applicable to the Containment Leakage Rate Testing Program.

5.20 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license or

2. A change to the USAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

Proposed Technical Specification Pages
(Clean)

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System

2.1.1 Operable Components

Applicability

Applies to the operable status of the reactor coolant system components.

Objective

To specify certain conditions of the reactor coolant system components.

Specifications

Limiting conditions for operation are as follows:

(1) Reactor Critical

All four (4) reactor coolant pumps shall be in operation.

Exceptions

The limitations of this specification may be suspended during the performance of physics tests provided the power level is $\leq 10^{-1}$% of rated power and the flow requirements of Table 2-11 No. 2 are met.

(2) Hot Shutdown or 350°F $\leq T_{cold} \leq$ 515°F

(a) The reactor coolant loops listed below shall be operable:

(i) Reactor coolant loop 1 and at least one associated reactor coolant pump.

(ii) Reactor coolant loop 2 and at least one associated reactor coolant pump.

(b) At least one of the above reactor coolant loops shall be in operation.

Exceptions

All reactor coolant pumps may be de-energized for up to one hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.1 Operable Components (Continued)

(12) Reactor Coolant System Pressure Isolation Valves

(a) The integrity of all pressure isolation valves listed in Table 2-9 shall be demonstrated, except as specified in (b). Valve leakage shall not exceed the amounts indicated.

(b) In the event that the integrity of any pressure isolation valve specified in Table 2-9 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a nonfunctional valve are in and remain in the mode corresponding to the isolated condition. Manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supply deenergized.

(c) If Specifications (a) and (b) above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

Basis

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation and maintain DNBR above the minimum DNBR limit during all normal operations and anticipated transients.

When Specification 2.1.1(2) is applicable, the reactor coolant pumps (RCPs) are used to provide forced circulation heat removal during heatup and cooldown. Under these conditions, decay heat removal requirements are low enough that a single reactor coolant system (RCS) loop with one RCP is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to provide redundant paths for decay heat removal. Only one RCP needs to be OPERABLE to declare the associated RCS loop OPERABLE. Reactor coolant natural circulation is not normally used but is sufficient for core cooling. However, natural circulation does not provide turbulent flow conditions. Therefore, boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be assured.

2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate

Applicability

Applies to the temperature change rates and pressure of the Reactor Coolant System (RCS).

Objective

To specify limiting conditions of the reactor coolant system heatup and cooldown rates.

Specification

The combination of RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR and as designated below:

a. Allowable combinations of pressure and temperature (T_c) for a specific heatup rate shall be below and to the right of the applicable limit lines as shown on the pressure and temperature (P-T) limit Figure(s) in the PTLR.

b. Allowable combinations of pressure and temperature (T_c) for a specific cooldown rate shall be below and to the right of the applicable limit lines as shown on the P-T limit Figure(s) in the PTLR.

c. The heatup rate of the pressurizer shall not exceed 100°F in any one hour period.

d. The cooldown rate of the pressurizer shall not exceed 200°F in any one hour period.

Required Actions

(1) When any of the above limits are exceeded, the following corrective actions shall be taken:

4. Immediately initiate action to restore the temperature or pressure to within the limit.

5. Perform an analysis to determine the effects of the out of limit condition on the fracture toughness properties of the reactor coolant system.

6. Determine that the reactor coolant system remains acceptable for continued operation or be in cold shutdown within 36 hours.

(2) Before the radiation exposure of the reactor vessel exceeds the exposure for which they apply, the P-T limit Figure(s) shown in the PTLR shall be updated in accordance with the following criteria and procedures:

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

(c) When the linear heat rate is continuously monitored by the excore detectors, withdraw the full length CEA's beyond the long term insertion limits of Specification 2.10.2(7) and maintain the Axial Shape Index, Y_I within the limits of Limiting Condition for Operations for the Excore Monitoring of LHR Figure provided in the COLR. If the linear heat rate is exceeding its limits as determined by the Axial Shape Index, Y_I, being outside the limits of the Limiting Condition for Operation for Excore Monitoring of LHR Figure provided in the COLR:

(i) Restore the reactor power and Axial Shape Index, Y_I, to within the limits of the Limiting Condition for Operations for Excore Monitoring of LHR Figure provided in the COLR within 2 hours, or

(ii) Be in at least hot standby within the next 6 hours.

(d) When calibration of the ex-core detectors has not been accomplished within the previous 30 equivalent full power days, then:

(i) reduce the axial power distribution monitoring trip setpoints as shown in the Axial Power Distribution LSSS for 4 Pump Operation Figure provided in the COLR by 0.03 ASI units; and

(ii) reduce the axial power distribution monitoring Limiting Condition for Operations (LCO for Excore Monitoring of LHR and LCO for DNB Monitoring Figures provided in the COLR) by 0.03 ASI units.

When calibration of the ex-core detectors has not been accomplished within the previous 200 equivalent full power days, the power shall be limited to less than that corresponding to 75% of the peak linear heat rate permitted by Specification 2.10.4(1).

(2) Total Integrated Radial Peaking Factor

The calculated value of F_R^T defined by $F_R^T = F_R (1+T_q)$ shall be within the limit provided in the COLR. F_R is determined from a power distribution map with no non-trippable CEA's inserted and with all full length CEA's at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. The azimuthal tilt, T_q, is the measured value of T_q at the time F_R is determined.

2.0 LIMITING CONDITIONS FOR OPERATION

2.13 Limiting Safety System Settings, Reactor Protective System (continued)

TABLE 2-11

RPS LIMITING SAFETY SYSTEM SETTINGS

No.	Reactor Trip	Allowable Value
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 LSSS for 4 Pump Operation Figure provided in the COLR)
9	Steam Generator Differential Pressure	≤ 135 psid

A Setpoint cannot be set greater than 10% above measured power whenever reactor power is greater than 10% of rated power.

B May be bypassed below 10^{-4}% power.

C May be bypassed below 600 psia.

D Bypass allowed for containment leak test.

E Inhibited below 15% power.

F For physics testing at power levels less than 10^{-1}% of rated power, the low reactor coolant flow and thermal margin/low pressure trips may be bypassed until their reset points are exceeded if automatic bypass removal of 10^{-1}% of rated power is operable.

TABLE 2-1

ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENT SETTING LIMITS

Functional Unit	Channel	Allowable Value
1. High Containment Pressure	a. Safety Injection b. Containment Spray (3) c. Containment Isolation d. Containment Air Cooler DBA Mode e. Steam Generator Isolation	≤ 5 psig
2. Pressurizer Low/Low Pressure	a. Safety Injection b. Containment Spray (3) c. Containment Isolation d. Containment Air Cooler DBA Mode	≥ 1600 psia (1)
3. Containment High Radiation	Containment Ventilation Isolation	In accordance with the applicable Chemistry Manual calibration procedure
4. Low Steam Generator Pressure	a. Steam Line Isolation b. Auxiliary Feedwater Actuation c. Containment Spray (3)	≥ 500 psia (2) ≥ 466.7 psia ≥ 500 psia (2)
5. SIRW Low Level Switches	Recirculation Actuation	16 inches +0, -2 in. above tank bottom

TABLE 2-1 (continued)

ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENT SETTING LIMITS

Functional Unit	Channel	Allowable Value	
6. 4.16 KV Emergency Bus Low Voltage	a. Loss of Voltage	(2995.2 + 104, -20.8) volts ≤ 5.9 seconds(4)	} Trip
	b. Degraded Voltage		
	i) Bus 1A3 Side	≥ 3988.8 volts (4.8 ± .5) seconds	} Trip
	ii) Bus 1A4 Side	≥ 3990.6 volts (4.8 ± .5) seconds	} Trip
7. Low Steam Generator Water Level	Auxiliary Feedwater Actuation	≥ 28.2% of wide range tap span	
8. High Steam Generator Delta Pressure	Auxiliary Feedwater Actuation	≤ 119.7 psid	

(1) May be bypassed below 1700 psia and is automatically reinstated prior to exceeding 1700 psia.
(2) May be bypassed below 600 psia and is automatically reinstated prior to exceeding 600 psia.
(3) Simultaneous containment high pressure, pressurizer low/low pressure, and steam generator low pressure.
(4) Applicable for bus voltage ≤ 2995.2 - 20.8 volts only. (For voltage ≥ (2995.2 - 20.8) volts, time delay shall be > 5.9 seconds).

5.0 ADMINISTRATIVE CONTROLS

5.7 Not used.

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 5.24.

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.

5.0 ADMINISTRATIVE CONTROLS

5.18 Process Control Program (PCP) (Continued)

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program. This documentation shall contain:

1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2. A determination that the change will maintain the overall conformance of the solidified waste program to existing requirements of federal, state, or other applicable regulations.

b. Shall become effective after the review and acceptance by the Plant Review Committee and the approval of the plant manager.

c. Temporary changes to the PCP may be made in accordance with Technical Specification 5.8.2.

d. Shall be submitted to the Nuclear Regulatory Commission in the form of a complete, legible copy of the entire PCP as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the PCP was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed and shall indicate the date (e.g., month/year) the change was implemented.

5.19 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995," as modified by the following exceptions:

(1) If the Personnel Air Lock (PAL) is opened during periods when containment integrity is not required, the PAL door seals shall be tested at the end of such periods and the entire PAL shall be tested within 14 days after RCS temperature T_{cold} > 210°F.

(2) Type A tests may be deferred for penetrations of the steel pressure retaining boundary where the nominal diameter does not exceed one inch.

(3) Elapsed time between consecutive Type A tests used to determine performance shall be at least 24 months or refueling interval.

b. The containment design accident pressure (P_a) is 60 psig.

5.0 ADMINISTRATIVE CONTROLS

5.19 Containment Leakage Rate Testing Program (Continued)

c. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

d. Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ Maximum Pathway Leakage Rate (MXPLR) for Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Personnel Air Lock testing acceptance criteria are:

(i) Overall Personnel Air Lock leakage is $\leq 0.1\ L_a$ when tested at $\geq P_a$.

(ii) For each PAL door, seal leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 5.0 psig.

e. Containment Purge Valve (PCV-742A/B/C/D) testing acceptance criterion is:

For each Containment Purge Valve, leakage rate is < 18,000 SCCM when tested at $\geq P_a$.

f. If at any time when containment integrity is required and the total Type B and C measured leakage rate exceeds $0.60\ L_a$ Minimum Pathway Leakage Rate (MNLPR), repairs shall be initiated immediately. If repairs and retesting fail to demonstrate conformance to this acceptance criteria within 48 hours, then containment shall be declared inoperable.

g. The provisions of Specification 3.0.1 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

h. The provisions of Specifications 3.0.4 and 3.0.5 are applicable to the Containment Leakage Rate Testing Program.

5.20 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license or

2. A change to the USAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.