Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

May 7, 2004
LIC-04-0062

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

References:
1. Docket No. 50-285
2. Letter from NRC (A. B. Wang) to OPPD (R. T. Ridenoure) dated January 16, 2004, "Fort Calhoun Station Unit No. 1 – Issuance of Amendment" (TAC No. MC0029) (NRC-04-005)
3. Letter from OPPD (S. K. Gambhir) to NRC (Document Control Desk) dated February 6, 2004, Fort Calhoun Station Unit No. 1 License Amendment Request, "Extension of Implementation Period for License Amendment 224" (LIC-04-0017)
4. Letter from NRC (A. B. Wang) to OPPD (R. T. Ridenoure) dated February 13, 2004, "Fort Calhoun Station Unit No. 1 – Issuance of Amendment" (TAC No. MC1949) (NRC-04-017)

SUBJECT: Fort Calhoun Station Unit No. 1 Exigent License Amendment Request, "Restoration of Previous Licensed Rated Power Limit"

Pursuant to 10 CFR 50.90, Omaha Public Power District (OPPD) hereby transmits an application for exigent amendment to the Fort Calhoun Station (FCS) Unit 1 Operating License. OPPD proposes to restore the licensed rated power from 1524 megawatts thermal (MWt) approved (but not implemented) in Amendment 224 (Reference 2) back to the previous limit of 1500 MWt.

Amendment 224 approved the increase in rated power to allow for measurement uncertainty recapture (MUR) based upon the successful installation and testing of the CROSSFLOW ultrasonic flow measurement system. As indicated in Reference 3, problems associated with installation and testing of the CROSSFLOW system required OPPD to seek an extension of the original implementation period. The NRC approved the request by issuing Amendment 225 (Reference 4).

OPPD and the vendor have worked diligently but have been unable to resolve technical issues associated with the CROSSFLOW system. OPPD does not anticipate that these issues can be resolved by the implementation date of May 15, 2004. Therefore, OPPD believes it is prudent to restore the previous licensed rated power limit (1500 MWt) as measured by existing feedwater flow instrumentation.

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The proposed 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 does not involve a no significant hazards consideration. The bases for these determinations, information supporting the change, a no significant hazards consideration, and an environmental consideration are included. No new regulatory commitments are included in this application.

Attachment 1 provides the No Significant Hazards Evaluation and the technical bases for this requested change. The exigency and why it could not have been avoided are addressed in Attachment 2 pursuant to 10 CFR 50.91(a)(6)(vi). Attachment 3 contains the existing TS pages marked-up to show the proposed change. Attachment 4 provides revised, clean TS pages.

OPPD requests approval and issuance of the proposed license amendment on an exigent basis no later than May 15, 2004.

I declare under penalty of perjury that the forgoing is true and correct. (Executed on May 7, 2004.)

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

Sincerely,

R. T. Ridenoure
Vice President

RTR/MLE/mle

Attachments
1. OPPD Evaluation for Amendment of Operating License
2. Explanation of the Exigency and Why the Situation Could Not Have Been Avoided
3. Markup of Technical Specification Pages
4. 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

OPPD Evaluation
For
Restoration of Previous Licensed Rated Power Limit

1.0 INTRODUCTION

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

Omaha Public Power District (OPPD) proposes to restore the licensed rated power level of 1500 megawatts thermal (MWt), which existed prior to Amendment 224. Because of unforeseen equipment problems, OPPD is unable to complete all modifications associated with the measurement uncertainty recapture (MUR) power uprate and comply with the following commitment:

"Modifications associated with the MUR power uprate will be completed prior to implementation. This includes implementation of control room alarm functions".

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed changes are as follows:

Restore Condition 3.A. of the Renewed Operating License to its previous licensed rated power level of 1500 MWt.

Restore the Definition of Rated Power to its previous value of 1500 MWt.

Restore the reference to RATED POWER in the Basis of Technical Specifications 2.1.6 and 3.5 to 1500 MWt.

3.0 BACKGROUND

Amendment 224 of FCS Operating License DPR-40 was approved and issued by the Nuclear Regulatory Commission (NRC) on January 16, 2004. Amendment 224 approved a MUR power uprate by revising the renewed operating license and the Technical Specifications to increase the licensed rated power by 1.6 percent from 1500 megawatts thermal (MWt) to 1524 MWt.

The MUR power uprate at FCS was based on decreased instrument uncertainty provided in part by installation of a CROSSFLOW ultrasonic flow measurement system. Included in the Regulatory Commitments contained in the application was the following statement: "Modifications associated with the MUR power uprate will be completed prior to implementation."

Difficulties encountered with the installation and testing of the CROSSFLOW system resulted in OPPD submitting an exigent license amendment on February 6, 2004 to extend the implementation period to 120 days, which ends on May 15, 2004. The NRC subsequently granted the requested extension in Amendment 225 dated February 13, 2004.

OPPD and the vendor have worked diligently throughout the implementation period to resolve problems with installation and testing of the CROSSFLOW system. Problems encountered during the testing include a discrepancy in main feedwater flow readings from the CROSSFLOW system. OPPD has concluded that the technical issues associated with the CROSSFLOW system cannot be resolved within the foreseeable future.

Although OPPD and the vendor may continue testing of the CROSSFLOW system, previously existing instrumentation for feedwater flow measurement will be used to determine reactor power. Therefore, OPPD proposes to restore the previously licensed rated power level of 1500 MWt.

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

Verbal guidance from the Chief, Section 2, Project Directorate IV for this situation recommended: (1) restoration of the pre-Amendment 224 licensed rated power level of 1500 MWt through submittal of a license amendment request, (2) request approval prior to the end of the implementation period (May 15, 2004) in order to avoid a violation of Item 3 in Amendment 225, and (3) use of an exigent amendment request.

5.0 TECHNICAL ANALYSIS

Amendment 224 has not been implemented. All accident analyses performed under the previous licensed rated power level of 1500 MWt remain valid. The proposed change restores the previously NRC approved licensed rated power level of 1500 MWt.

6.0 REGULATORY ANALYSIS

NRC approval to raise licensed rated power to 1524 MWt was contingent upon operability of the CROSSFLOW system, as documented in the OPPD application and the NRC Safety Evaluation Report that supports License Amendment 224. Amendment 224, approved by the NRC, has not been implemented by OPPD. Amendment 225 extended the implementation period specified in Amendment 224 from 30 days to 120 days. However, despite diligent efforts by OPPD and the vendor, problems with installation and testing of CROSSFLOW system have not been resolved and are unlikely to be resolved in the foreseeable future.

Therefore, OPPD is unable to meet the regulatory commitment that modifications associated with the MUR power uprate will be completed prior to implementation. As a result, OPPD proposes to restore the licensed rated power level that existed prior to Amendment 224 of 1500 MWt. Since the pre-existing feedwater flow measurement instrumentation is still in place and operable, it is OPPD's position that the proposed amendment does not affect any technical or safety aspects of plant operation. The exigency and why it could not have been avoided are addressed in Attachment 2 pursuant to 10 CFR 50.91(a)(6)(vi).

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

OPPD has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) 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 exigent amendment restores the previously approved licensed rated power level of 1500 MWt that existed prior to Amendment 224. Amendment 224 was never implemented. All accident analyses performed under the previous licensed rated power level of 1500 MWt remain valid.

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 exigent amendment restores the previously approved licensed rated power level of 1500 MWt that existed prior to Amendment 224. Amendment 224 was never implemented. All accident analyses performed under the previous licensed rated power level of 1500 MWt remain valid.

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

   Response: No.

   The proposed exigent amendment restores the previously approved licensed rated power level of 1500 MWt that existed prior to Amendment 224. Amendment 224 was never implemented. All accident analyses performed under the previous licensed rated power level of 1500 MWt remain valid.

Based on the above, OPPD 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

The proposed exigent amendment restores the previously NRC approved licensed rated power level of 1500 MWt. All accident analyses performed under the previous licensed rated power level of 1500 MWt remain valid. Despite diligent efforts, OPPD and the CROSSFLOW vendor are unable to meet the regulatory commitment that modifications associated with the MUR power uprate will be completed prior to implementation. The proposed amendment does not affect any technical or safety aspects of plant operation. The changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

- As demonstrated in Section 7.0, the proposed amendment does not involve a significant hazards consideration.

- The proposed amendment does not result in a significant change in the types or increase in the amounts of any effluents that may be released off-site. Also, the change does not introduce any new effluents or significantly increase the quantities of existing effluents. As such, the change cannot significantly affect the types or amounts of any effluents that may be released off-site.

- The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure. The proposed change does not result in any physical plant changes. No new surveillance requirements are anticipated as a result of these changes that would require additional personnel entry into radiation controlled areas. Therefore, the amendment has no significant affect on either 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.

9.0 PRECEDENCE

None

10.0 REFERENCES

10.1 Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated July 18, 2003 (LIC-03-0067)

10.2 Letter from OPPD (S. K. Gambhir) to NRC (Document Control Desk) dated August 28, 2003 (LIC-03-0122)

10.3 Letter from OPPD (S. K. Gambhir) to NRC (Document Control Desk) dated October 31, 2003 (LIC-03-0148)

10.4 Letter from OPPD (S. K. Gambhir) to NRC (Document Control Desk) dated December 15, 2003 (LIC-03-0164)

10.5 Letter from NRC (A. B. Wang) to OPPD (R. T. Ridenoure), Amendment 224 to Fort Calhoun Station, Unit No. 1 Operating License DPR-40, dated January 16, 2004 (NRC-04-005) (ADAMS Accession #ML040200757).

10.6 Letter from OPPD (S. K. Gambhir) to NRC (Document Control Desk) dated February 6, 2004 Fort Calhoun Station Unit No. 1 License Amendment Request, "Extension of Implementation Period for License Amendment 224" (LIC-04-0017)

10.7 Letter from NRC (A. B. Wang) to OPPD (R. T. Ridenoure) dated February 13, 2004, "Fort Calhoun Station Unit No. 1 – Issuance of Amendment" (TAC No. MC1949) (NRC-04-017)

Attachment 2

Explanation of the Exigency and Why the Situation Could Not Have Been Avoided

OPPD has worked diligently with the vendor to resolve problems encountered during installation and testing of the CROSSFLOW system. OPPD and the vendor fully expected to be able to resolve the technical issues within the 120 day implementation period. However, only recently was it determined that this could not be accomplished. Therefore, OPPD must submit an exigent license amendment to restore the previously licensed rated power level of 1500 MWt.

The CROSSFLOW modification associated with the MUR power uprate cannot be completed prior to the end of the 120 day implementation period on May 15, 2004 as approved in Amendment 225. Therefore, OPPD is unable to meet a license condition necessary to implement Amendment 224. OPPD will continue to work with the vendor to resolve technical issues with the CROSSFLOW system. When these issues are resolved, OPPD plans to resubmit a revised license amendment request for MUR power uprate.

OPPD has continued to operate Fort Calhoun Station, Unit No. 1 using existing feedwater flow measurement instrumentation at 1500 MWt or less. FCS procedures and design basis documents continue to reflect the pre-Amendment 224 licensed rated power level of 1500 MWt since Amendment 224 has not been implemented.

Verbal guidance from the Chief, Section 2, Project Directorate IV for this situation recommended: (1) restoration of the pre-Amendment 224 licensed rated power level of 1500 MWt through submittal of a license amendment request, (2) request approval prior to the end of the implementation period (May 15, 2004) in order to avoid a violation of Item 3 in Amendment 225, and (3) use of an exigent amendment request.

ATTACHMENT 3

Markup of Technical Specification Pages

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or instrument calibration or when associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter 1: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

Omaha Public Power District is authorized to operate the Fort Calhoun Station, Unit 1, at steady state reactor core power levels not in excess of ~~1524~~ 1500 megawatts thermal (rated power).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~224~~, are hereby incorporated in the license. Omaha Public Power District shall operate the facility in accordance with the Technical Specifications.

C. Security and Safeguards Contingency Plans

The Omaha Public Power District shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Fort Calhoun Station Physical Security Plan," with revisions submitted through September 30, 1988; "Fort Calhoun Station Guard Training and Qualification Plan," with revisions submitted through August 17, 1979; and "Fort Calhoun Station Safeguards Contingency Plan," with revisions submitted through March 20, 1979. If certain security modifications are delayed beyond expectations of the schedule, approved compensatory measures must be implemented during the transition period.

TECHNICAL SPECIFICATIONS

DEFINITIONS

The following terms are defined for uniform interpretation of these Specifications.

REACTOR OPERATING CONDITIONS

Rated Power

A steady state reactor core output of ~~1524~~ 1500 MWt.

Reactor Critical

The reactor is considered critical for purposes of administrative control when the neutron flux logarithmic range channel instrumentation indicates greater than 10^{-4}% of rated power.

Power Operation Condition (Operating Mode 1)

The reactor is in the power operation condition when it is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

Hot Standby Condition (Operating Mode 2)

The reactor is considered to be in a hot standby condition if the average temperature of the reactor coolant (Tavg) is greater than 515°F, the reactor is critical, and the neutron flux power range instrumentation indicates less than 2% of rated power.

Hot Shutdown Condition (Operating Mode 3)

The reactor is in a hot shutdown condition if the average temperature of the reactor coolant (Tavg) is greater than 515°F and the reactor is subcritical by at least the amount defined in Paragraph 2.10.2.

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (continued)
2.1.6 Pressurizer and Main Steam Safety Valves (continued)

Action statements (5)b. and c. include the removal of power from a closed block valve to preclude any inadvertent opening of the block valve at a time the PORV may not be closed due to maintenance. However, the applicability requirements of the LCO to operate with the block valve(s) closed with power maintained to the block valve(s) are only intended to permit operation of the plant for a limited period of time not to exceed the next refueling shutdown (Mode 5), so that maintenance can be performed on the PORV(s) to eliminate the seat leakage condition.

To determine the maximum steam flow, the only other pressure relieving system assumed operational is the main steam safety valves. Conservative values for all systems parameters, delay times and core moderator coefficients are assumed. Overpressure protection is provided to portions of the reactor coolant system which are at the highest pressure considering pump head, flow pressure drops and elevation heads.

If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve lift pressure would be less than half of the capacity of one safety valve. This specification, therefore, provides adequate defense against overpressurization when the reactor is subcritical.

Performance of certain calibration and maintenance procedures on safety valves requires removal from the pressurizer. Should a safety valve be removed, either operability of the other safety valve or maintenance of at least one nozzle open to atmosphere will assure that sufficient relief capacity is available. Use of plastic or other similar material to prevent the entry of foreign material into the open nozzle will not be construed to violate the "open to atmosphere" provision, since the presence of this material would not significantly restrict the discharge of reactor coolant.

The total relief capacity of the ten main steam safety valves is 6.606 x 10^6 lb/hr. If, following testing, the as found setpoints are outside +/-1% of nominal nameplate values, the valves are set to within the +/-1% tolerance. The main steam safety valves were analyzed for a total loss of main feedwater flow while operating at ~~RATED POWER~~ 1500 MWt[3] to ensure that the peak secondary pressure was less than 1100 psia, the ASME Section III upset pressure limit of 10% greater than the design pressure. At the power of ~~RATED POWER~~ 1500 MWt, sufficient relief valve capacity is available to prevent overpressurization of the steam
system on loss-of-load conditions.[4] These analyses are based on a minimum of four-of-five operable main steam safety valves on each main steam header.

The power-operated relief valve low setpoint will be adjusted to provide sufficient margin, when used in conjunction with Technical Specification Sections 2.1.1 and 2.3, to prevent the design basis pressure transients from causing an overpressurization incident. Limitation of this requirement to scheduled cooldown ensures that, should emergency conditions dictate rapid cooldown of the reactor coolant system, inoperability of the low temperature overpressure protection system would not prove to be an inhibiting factor. The effective full flow area of an open PORV is 0.94 in^2.

Removal of the reactor vessel head provides sufficient expansion volume to limit any of the design basis pressure transients. Thus, no additional relief capacity is required.

References
(1) Article 9 of the 1968 ASME Boiler and Pressure Vessel Code, Section III
(2) USAR, Section 14.9
(3) USAR, Section 14.10
(4) USAR, Sections 4.3.4, 4.3.9.5

3.0 SURVEILLANCE REQUIREMENTS

3.5 Containment Tests (Continued)

Basis

The containment is designed for an accident pressure of 60 psig.[2] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a maximum temperature of about 120°F. With these initial conditions the temperature of the steam-air mixture at the peak accident pressure of 60 psig is 288°F.

Prior to initial operation, the containment was strength-tested at 69 psig and then was leak tested. The design objective of the pre-operational leakage rate test has been established as 0.1% by weight for 24 hours at 60 psig. This leakage rate is consistent with the construction of the containment, which is equipped with independent leak-testable penetrations and contains channels over all inaccessible containment liner welds, which were independently leak-tested during construction.

Safety analyses have been performed on the basis of a leakage rate of 0.1% of the free volume per day of the first 24 hours following the maximum hypothetical accident. With this leakage rate, a reactor power level of ~~RATED POWER~~ 1500 MWt, and with minimum containment engineered safety systems for iodine removal in operation (one air cooling and filtering unit), the public exposure would be well below 10 CFR Part 100 values in the event of the maximum hypothetical accident.[3] The performance of an integrated leakage rate test and performance of local leak rate testing of individual penetrations at periodic intervals during plant life provides a current assessment of potential leakage from the containment.

The reduced pressure (5 psig) test on the PAL is a conservative method of testing and provides adequate indication of any potential containment leakage path. The test is conducted by pressurizing between two resilient seals on each door. The test pressure tends to unseat the resilient seals which is opposite to the accident pressure that tends to seat the resilient seals. A periodic test ensures the overall PAL integrity at 60 psig.

The integrated leakage rate test (Type A test) can only be performed during refueling shutdowns.

ATTACHMENT 4

Proposed Technical Specification Pages (clean)

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or instrument calibration or when associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter 1: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

Omaha Public Power District is authorized to operate the Fort Calhoun Station, Unit 1, at steady state reactor core power levels not in excess of 1500 megawatts thermal (rated power).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. , are hereby incorporated in the license. Omaha Public Power District shall operate the facility in accordance with the Technical Specifications.

C. Security and Safeguards Contingency Plans

The Omaha Public Power District shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Fort Calhoun Station Physical Security Plan," with revisions submitted through September 30, 1988; "Fort Calhoun Station Guard Training and Qualification Plan," with revisions submitted through August 17, 1979; and "Fort Calhoun Station Safeguards Contingency Plan," with revisions submitted through March 20, 1979. If certain security modifications are delayed beyond expectations of the schedule, approved compensatory measures must be implemented during the transition period.

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Hot Standby Condition (Operating Mode 2)

The reactor is considered to be in a hot standby condition if the average temperature of the reactor coolant (Tavg) is greater than 515°F, the reactor is critical, and the neutron flux power range instrumentation indicates less than 2% of rated power.

Hot Shutdown Condition (Operating Mode 3)

The reactor is in a hot shutdown condition if the average temperature of the reactor coolant (Tavg) is greater than 515°F and the reactor is subcritical by at least the amount defined in Paragraph 2.10.2.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (continued)
2.1.6 Pressurizer and Main Steam Safety Valves (continued)

Action statements (5)b. and c. include the removal of power from a closed block valve to preclude any inadvertent opening of the block valve at a time the PORV may not be closed due to maintenance. However, the applicability requirements of the LCO to operate with the block valve(s) closed with power maintained to the block valve(s) are only intended to permit operation of the plant for a limited period of time not to exceed the next refueling shutdown (Mode 5), so that maintenance can be performed on the PORV(s) to eliminate the seat leakage condition.

To determine the maximum steam flow, the only other pressure relieving system assumed operational is the main steam safety valves. Conservative values for all systems parameters, delay times and core moderator coefficients are assumed. Overpressure protection is provided to portions of the reactor coolant system which are at the highest pressure considering pump head, flow pressure drops and elevation heads.

If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve lift pressure would be less than half of the capacity of one safety valve. This specification, therefore, provides adequate defense against overpressurization when the reactor is subcritical.

Performance of certain calibration and maintenance procedures on safety valves requires removal from the pressurizer. Should a safety valve be removed, either operability of the other safety valve or maintenance of at least one nozzle open to atmosphere will assure that sufficient relief capacity is available. Use of plastic or other similar material to prevent the entry of foreign material into the open nozzle will not be construed to violate the "open to atmosphere" provision, since the presence of this material would not significantly restrict the discharge of reactor coolant.

The total relief capacity of the ten main steam safety valves is 6.606×10^6 lb/hr. If, following testing, the as found setpoints are outside +/-1% of nominal nameplate values, the valves are set to within the +/-1% tolerance. The main steam safety valves were analyzed for a total loss of main feedwater flow while operating at 1500 MWt[3] to ensure that the peak secondary pressure was less than 1100 psia, the ASME Section III upset pressure limit of 10% greater than the design pressure. At the power of 1500 MWt, sufficient relief valve capacity is available to prevent overpressurization of the steam system on loss-of-load conditions.[4] These analyses are based on a minimum of four-of-five operable main steam safety valves on each main steam header.

The power-operated relief valve low setpoint will be adjusted to provide sufficient margin, when used in conjunction with Technical Specification Sections 2.1.1 and 2.3, to prevent the design basis pressure transients from causing an overpressurization incident. Limitation of this requirement to scheduled cooldown ensures that, should emergency conditions dictate rapid cooldown of the reactor coolant system, inoperability of the low temperature overpressure protection system would not prove to be an inhibiting factor. The effective full flow area of an open PORV is 0.94 in^2.

Removal of the reactor vessel head provides sufficient expansion volume to limit any of the design basis pressure transients. Thus, no additional relief capacity is required.

References

(1) Article 9 of the 1968 ASME Boiler and Pressure Vessel Code, Section III
(2) USAR, Section 14.9
(3) USAR, Section 14.10
(4) USAR, Sections 4.3.4, 4.3.9.5

3.0 SURVEILLANCE REQUIREMENTS

3.5 Containment Tests (Continued)

Basis

The containment is designed for an accident pressure of 60 psig.[2] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a maximum temperature of about 120°F. With these initial conditions the temperature of the steam-air mixture at the peak accident pressure of 60 psig is 288°F.

Prior to initial operation, the containment was strength-tested at 69 psig and then was leak tested. The design objective of the pre-operational leakage rate test has been established as 0.1% by weight for 24 hours at 60 psig. This leakage rate is consistent with the construction of the containment, which is equipped with independent leak-testable penetrations and contains channels over all inaccessible containment liner welds, which were independently leak-tested during construction.

Safety analyses have been performed on the basis of a leakage rate of 0.1% of the free volume per day of the first 24 hours following the maximum hypothetical accident. With this leakage rate, a reactor power level of 1500 MWt, and with minimum containment engineered safety systems for iodine removal in operation (one air cooling and filtering unit), the public exposure would be well below 10 CFR Part 100 values in the event of the maximum hypothetical accident.[3] The performance of an integrated leakage rate test and performance of local leak rate testing of individual penetrations at periodic intervals during plant life provides a current assessment of potential leakage from the containment.

The reduced pressure (5 psig) test on the PAL is a conservative method of testing and provides adequate indication of any potential containment leakage path. The test is conducted by pressurizing between two resilient seals on each door. The test pressure tends to unseat the resilient seals which is opposite to the accident pressure that tends to seat the resilient seals. A periodic test ensures the overall PAL integrity at 60 psig.

The integrated leakage rate test (Type A test) can only be performed during refueling shutdowns.