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10 CFR 50.90

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

Prairie Island Nuclear Generating Plant, Unit 2
Docket 50-306
Renewed License No. DPR-60

License Amendment Request for Exception to Technical Specification 5.5.14 Testing Requirements Associated With Steam Generator Replacement

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota Corporation (NSPM), doing business as Xcel Energy, hereby requests an amendment to the renewed operating license for Prairie Island Nuclear Generating Plant (PINGP) Unit 2. Specifically, NSPM proposes to revise Technical Specification (TS) 5.5.14, "Containment Leakage Rate Testing Program."

The proposed change would revise Appendix A of the Operating License to except PINGP Unit 2 from the requirements of Regulatory Guide (RG) 1.163 as specified in TS 5.5.14 for post-modification containment leakage rate testing associated with steam generator replacement. This exception is being requested to avoid performing an unnecessary integrated leakage rate test (ILRT). This exception is being requested so that the American Society of Mechanical Engineers (ASME) Code Section III/XI pressure test requirements for the replacement steam generators may be used to satisfy the intent of the RG 1.163 requirements rather than performing a Type A test, i.e., a containment ILRT. To accomplish this change, NSPM is requesting that this license amendment request (LAR) be approved prior to the steam generator replacement currently scheduled for the fall of 2013.

Enclosure 1 to this letter provides the evaluation of the proposed TS change and its supporting justification, including a no significant hazards determination. Enclosure 2 provides the existing TS page marked-up to show the proposed change. Enclosure 3 provides the revised TS page incorporating the proposed change.

NSPM has determined that the information for the proposed amendment does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment meets the categorical exclusion requirements of 10 CFR 51.22(c)(9) and an environmental impact assessment need not be prepared.

A copy of this submittal, including the Determination of No Significant Hazards Consideration is being forwarded to the designated State of Minnesota official pursuant to 10 CFR 50.91(b)(1).

NSPM requests approval of this proposed amendment by August 1, 2013 to support the steam generator replacement currently scheduled for the fall of 2013. Once approved, the amendment shall be implemented within 60 days.

If there are any questions or if additional information is needed, please contact Glenn Adams at 612-330-6777.

Summary of Commitments

This letter contains no new commitments or revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on JUL 25 2012

James E. Molden
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosures (3)

cc: Regional Administrator, Region III, USNRC
Project Manager, Prairie Island Nuclear Generating Plant, USNRC
Resident Inspector, Prairie Island Nuclear Generating Plant, USNRC
State of Minnesota

ENCLOSURE 1

Evaluation of the Proposed Change

License Amendment Request for Exception to Technical Specification 5.5.14 Testing Requirements Associated With Steam Generator Replacement

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota Corporation (NSPM), doing business as Xcel Energy, hereby requests an amendment to the renewed operating license for Prairie Island Nuclear Generating Plant (PINGP) Unit 2. Specifically, NSPM proposes to revise Technical Specification (TS) 5.5.14, "Containment Leakage Rate Testing Program."

The proposed change would revise Appendix A of the Operating License to except PINGP Unit 2 from the requirements of Regulatory Guide (RG) 1.163 as specified in TS 5.5.14 for post-modification containment leakage rate testing associated with steam generator replacement. This exception is being requested so that the American Society of Mechanical Engineers (ASME) Code Section III/XI pressure test requirements for the replacement steam generators may be used to satisfy the intent of the RG 1.163 requirements rather than performing a Type A test, i.e., a containment integrated leak rate test (ILRT). To accomplish this change, NSPM is requesting that this license amendment request (LAR) be approved prior to the steam generator replacement currently scheduled for the fall of 2013.

2.0 DETAILED DESCRIPTION

A brief description of the proposed change is provided below along with a discussion of the justification for the change. The specific wording changes to the Technical Specification (TS) are provided in Enclosures 2 and 3.

PINGP Technical Specification (TS) 5.5.14.a states:

> 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 exception:
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> 1. Unit 1 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

RG 1.163 (Reference 6.1) endorses Nuclear Energy Institute (NEI) 94-01, Revision 0 for methods acceptable to comply with the requirements of Option B. In its Safety Evaluation Report dated June 25, 2008, NRC approved the use of NEI 94-01, Revision 2-A (Reference 6.2). Prior to returning the primary containment system to operation, NEI 94-01 requires leakage rate testing following repairs and modifications that affect the containment leakage integrity.

The proposed amendment would except PINGP Unit 2 from the requirements of RG 1.163 as specified in the TS for post-modification leakage rate testing

associated with replacement of the steam generators. This would be accomplished by extending the exception statement (TS 5.5.14.a.1) to not only apply to the now-completed Unit 1 steam generator replacement, but also the pending Unit 2 steam generator replacement. The proposed revision to the PINGP TS 5.5.14 is shown on the marked-up page in Enclosure 2.

In summary, this LAR will provide an exception to containment leakage testing requirements of RG 1.163 and TS 5.5.14 such that those testing requirements will not be invoked for post-modification testing after the replacement of Unit 2 steam generators.

3.0 TECHNICAL EVALUATION

3.1 Background:

PINGP is a dual unit site. Each unit is a two-loop Westinghouse design, licensed to an uprated power of 1677 megawatts-thermal (MWt). The units were originally provided with Westinghouse Model 51 steam generators. In 2004, the licensee replaced the original steam generators in Unit 1 with steam generators fabricated by Framatome ANP. Presently, NSPM is preparing to replace the Unit 2 Westinghouse steam generators with AREVA (formerly Framatome ANP) models designed and fabricated by the same manufacturer to the same ASME Code year and addendum as the Unit 1 replacement steam generators. NSPM plans to install these replacement steam generators in fall 2013.

Each ASME Section III replacement steam generator (RSG) consists of a new lower part and new upper part, and is essentially identical in design to the Unit 1 RSGs with the exception of the cylindrical head ring. The cylindrical head ring was a separate item for Unit 1 whereas for Unit 2 it is included as part of the tubesheet forging.

The RSGs will occupy the same physical envelope as the original steam generators (OSGs). There are no changes to interfaces with the reactor coolant, main steam, feedwater, or auxiliary feedwater systems. The pipes attaching these systems to the OSGs will be cut and welded back to the RSGs during installation.

The Unit 2 reactor containment vessel is a cylindrical steel pressure vessel with hemispherical dome and ellipsoidal bottom which houses the reactor pressure vessel, the steam generators, reactor coolant pumps, the reactor coolant loops, the accumulators of the safety injection system, the primary coolant pressurizer, the pressurizer relief tank, and other branch connections of the reactor coolant system. The reactor containment vessel is, in turn, housed completely within the shield building. Since the rigging and handling necessary to perform the Unit 2 steam generator replacement are designed to use the equipment hatch that services the reactor containment vessel, no alteration or modification of the reactor containment vessel will be required. For the same reason, no modifications to the

structure of the Unit 2 shield building will be required to achieve access to the equipment hatch for rigging and handling of the steam generators. Thus, there are no structural effects to the reactor containment vessel and the shield building resulting from the steam generator replacement activities.

Although the steam generators are not part of the reactor containment vessel, portions of them are relied upon to act as a barrier against the uncontrolled release of radioactivity to the environment during a design basis loss of coolant accident (LOCA). Thus, the outer shell of the steam generators, the inside-containment portions of the main steam line, the main and auxiliary feedwater lines, the steam generator blowdown lines, sampling lines, the recirculation lines, and instrument lines may be considered part of the primary containment system boundary. During the steam generator replacement, portions of these system pressure boundaries are being replaced with new components connected with new welds and connection mechanisms. Thus, replacing the steam generators will constitute a modification to the primary containment system boundary.

3.2 Current Licensing Basis

PINGP TS 5.5.14.a requires that a program be established to implement the leakage testing of the containment as required by 10 CFR 50 Appendix J, as modified by approved exemptions. This program is in accordance with the guidelines contained in RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. RG 1.163 (Reference 6.1) endorses NEI 94-01, Revision 0 (Reference 6.2) for methods acceptable to comply with the requirements of Option B. Section 9.2.4 of NEI 94-01 requires that a Type A or local leakage rate testing be conducted prior to returning the primary containment system to operation following a modification that affects the containment leakage integrity. As stated above, replacing the steam generators will constitute a modification to the primary containment system boundary and thus affects containment leakage integrity. As discussed in Section 3.3 below, performing local leakage rate testing for this modification is not practical. Therefore, to satisfy TS 5.5.14.a, a Type A test (i.e., an ILRT) would have to be performed. Because the next ILRT for Unit 2 is not scheduled to occur until approximately 2022, an additional ILRT would have to be performed unless an exception to the requirement is obtained.

This exception is requested to avoid performing an unnecessary ILRT. As discussed below, the ILRT is unnecessary because the ASME Section III/XI pressure test requirements for the replacement steam generators will satisfy the intent of Regulatory Guide 1.163 and NEI 94-01 requirements. This exception is similar to that granted to Unit 1 in Reference 6.3.

3.3 Justification for the Proposed Changes

The PINGP Unit 2 plant design incorporates a closed system for transferring steam from the steam generators inside of the primary containment to the main turbine generator in the turbine building. The inside-containment portion of this closed system consists of the outer shell of the steam generators, the main steam line, the main and auxiliary feedwater lines, the steam generator blowdown lines, the recirculation lines, sampling lines, and instrument lines. During a design basis LOCA, these elements inside containment form a barrier against the uncontrolled release of radioactivity to the environment and thus are considered part of the primary containment system boundary.

The planned replacement of the PINGP Unit 2 steam generators includes the following activities:

- Cutting and removing the main steam lines, main and auxiliary feedwater lines, steam generator blowdown lines, and associated instrument and sampling lines.
- Cutting and removing the upper part of the steam generators.
- Cutting the reactor coolant piping and removing the steam generator lower assemblies.
- Installing the new steam generator lower subassemblies and re-welding the reactor coolant piping.
- Installing the new steam generator upper subassemblies on the new lower assemblies.
- Re-installing and re-welding the main steam lines, main and auxiliary feedwater lines, steam generator blowdown lines, recirculation lines, and associated instrument lines.

The planned replacement of the Unit 2 steam generators affects only these closed piping systems inside the reactor containment vessel. The steam generator replacement activities do not affect the reactor containment vessel structure or the structure of the shield building.

NEI 94-01 requires integrated leakage testing (Type A) or local leakage rate testing (Type B or Type C) prior to returning the primary containment system to operation following repairs and modification that affect the containment leakage integrity. The Type C testing requirements apply to leakage testing of containment isolation valves. The planned replacement does not affect any containment isolation valves, so Type C testing requirements are not applicable. The Type B testing requirements apply to leakage testing of gasketed or sealed containment penetrations (e.g., electrical penetrations), air lock door seals, and other doors with resilient seals or gaskets. Although the secondary side of the steam generators has access manways and handhole ports with gaskets, it is impractical to perform a Type B test for these items. Hence, since Type B or Type C testing cannot test all the affected areas, Appendix J would require that a Type A test be performed

prior to startup following the planned steam generator replacement. A Type A test measures the primary containment system overall integrated leakage rate under conditions representing design basis accident containment pressure and system alignment.

For preservice and inservice inspection requirements, the affected area of the primary containment system boundary is subject to ASME Section XI. The pressure boundary of the RSGs is constructed in accordance with ASME Section III Class 1. As such, the replacement of the steam generators is subject to the requirements of ASME Sections III and XI. The acceptance criterion for ASME Section III/XI system pressure testing for the base metal and welds is no leakage. Furthermore, the acceptance criterion for the specified field hydrostatic tests is no leakage (no exceptions for manways and handholes). Because the base metal, welds, manways, and handholes are not allowed to leak, the ASME Section III/XI pressure test requirements are more stringent than the Type A testing requirements. In addition, the hydrostatic test pressure (1357 psig) for the SG vessel and the inservice leak test pressure (at least 948 psig) for the main steam line weld will be at least 20 times that of a Type A test (46 psig per TS 5.5.14), noting that the hydrostatic test pressure will be applied in a direction opposite that of a Type A test (as discussed below).

The intent of performing a Type A test is to assure the leak-tight integrity of the area affected by the modification (i.e., the closed system inside the reactor containment vessel formed by the outer shell of the steam generators and the main steam, feedwater, steam generator blowdown, and feedwater piping) does not alter the overall leakage rate of the primary containment. Although the leak test is in a direction reverse that of a LOCA environment, the leak tightness of the components, piping, and welds is not dependent on the direction the pressure is applied. Thus, the ASME Section III/XI inspection and testing requirements fulfill the intent of the requirements of RG 1.163 and NEI 94-01. Likewise, the post-installation testing of the steam generator instrument lines will be in the direction opposite that of a LOCA environment and will show that the lines meet their specified leakage requirements. This test also fulfills the intent of the requirements of RG 1.163 and NEI 94-01 because the leak tightness of the fittings in the instrument lines is dependent on the mechanical makeup of the fitting and not the direction of the pressure being applied.

3.4 Conclusion

Therefore, NSPM proposes a revision to TS 5.5.14 to except Unit 2 from the requirements of RG 1.163 as specified in the Technical Specifications for post-modification integrated leakage rate testing associated with replacement of the steam generators. The effect of this amendment request would be to eliminate the post-modification containment leakage rate (Type A) testing required for the modifications to the primary containment system boundary specifically associated with replacement of the steam generators.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

PINGP Technical Specification 5.5.14 states:

> 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 exception:
>
> 1. Unit 1 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

RG 1.163 endorses NEI 94-01, Revision 0 for methods acceptable to comply with the requirements of Option B. In its Safety Evaluation Report dated June 25, 2008, NRC approved the use of NEI 94-01, Revision 2-A (Reference 6.2). Prior to returning the primary containment system to operation, NEI 94-01 requires leakage rate testing following repairs and modification that affect the containment leakage integrity.

The ASME Code Section III/XI inspection and testing requirements fulfill the intent of the requirements of RG 1.163 as specified in the TS. Because the leak-tight integrity of the primary containment boundary affected by the steam generator replacement will be assured, there is no change in the primary containment boundary's ability to fulfill its design function.

4.2 Precedent

In 2004, NRC approved Unit 1 License Amendment 165 (Reference 6.3), revising TS 5.5.14 to allow the licensee to perform post-modification testing of the containment pressure boundary following Unit 1 steam generator replacement in accordance with the ASME Code, Sections III and XI, instead of 10 CFR Part 50, Appendix J, Option B.

4.3 Significant Hazards Consideration

Northern States Power Company, a Minnesota Corporation (NSPM), doing business as Xcel Energy, proposes to amend the renewed facility operating license of Prairie Island Nuclear Generating Plants (PINGP) Unit 2. The purpose of this amendment is to modify the PINGP Technical Specifications (TS) to add a statement that provides an exception for Unit 2 from post-modification integrated leak rate test requirements associated with replacement of the steam generators.

NSPM has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92(c) 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 would provide the Prairie Island Nuclear Generating Plant an exception from performing a containment integrated leak rate test following the replacement of the steam generators in Unit 2.

   Integrated leak rate tests are performed to assure the leak-tightness of the primary containment boundary system, and as such they are not accident initiators. Therefore, not performing an integrated leak rate test will not affect the probability of an accident previously evaluated.

   The intent of post-modification integrated leak rate testing requirements is to assure the leak-tight integrity of the area affected by the modification. For the Unit 2 steam generator replacement modification, this intent will be satisfied by performing the inspections and tests required by the American Society of Mechanical Engineers (ASME) Code. Because the leak-tightness integrity of the primary containment boundary affected by the steam generator replacement will be assured, there is no change in the primary containment boundary's ability to confine radioactive materials during an accident.

   Therefore, adding a Technical Specification statement that provides an exception for Unit 2 from the steam generator replacement post-modification integrated leak rate testing requirements does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   The proposed change would provide the Prairie Island Nuclear Generating Plant an exception from performing a required containment integrated leak rate test following the replacement of the steam generators in Unit 2.

   Providing an exception from performing a test does not involve a physical change to the plant nor does it change the operation of the plant. Thus, it cannot introduce a new failure mode. Therefore, adding a Technical Specification statement that provides an exception for Unit 2 from the steam generator replacement post-modification integrated leak rate testing

requirements does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Do the proposed changes involve a significant reduction in a margin of safety?**

   Response: No

   The proposed change would provide the Prairie Island Nuclear Generating Plant an exception from performing a required containment integrated leak rate test following the replacement of the steam generators in Unit 2.

   The intent of post-modification integrated leak rate testing requirements is to assure the leak-tight integrity of the area affected by the modification. This intent will be satisfied by performing inspections and tests required by the ASME Code. The acceptance criterion for ASME Code system pressure testing for the base metal and welds is no leakage. In addition, the test pressure for the hydrostatic tests and the inservice system pressure test will be several times that required during an integrated leak rate test. Because the leak-tight integrity of the primary containment boundary affected by the steam generator replacement will be assured, there is no change in the primary containment boundary's ability to confine radioactive materials during an accident. Therefore, adding a Technical Specification statement that provides an exception for Unit 2 from the steam generator replacement post-modification integrated leak rate testing requirements does not involve a significant reduction in a margin of safety.

Therefore, based on the above, NSPM has concluded 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.

4.4 Conclusions

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.

5.0 ENVIRONMENTAL CONSIDERATIONS

10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment of an operating license for a

facility requires no environmental assessment if the operation of the facility in accordance with the proposed amendment does not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and (3) result in a significant increase in individual or cumulative occupational radiation exposure. NSPM has reviewed this LAR and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). 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. The basis for this determination follows.

1. As demonstrated in the 10 CFR 50.92 evaluation, the proposed amendment does not involve a significant hazards consideration.
2. The proposed amendment does not result in a significant change in the types or increase in the potential amounts of any effluents that may be released offsite. Implementation of the proposed project only involves a one-time exception to the containment integrated leakage rate test (ILRT), a routinely performed test with known radiological effects. As such, the amendment does not constitute a new plant operation or increased frequency of an existing operation. Rather, implementation of the amendment will mean that one less ILRT will be performed. Thus, the amendment would not introduce a new type of effluent and would only have the positive effect of actually precluding the potential release of any effluents caused by performing the ILRT which is eliminated by the amendment.
3. The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure. As discussed above, implementation of the proposed amendment will mean that one less ILRT will be performed. Thus, the amendment would only have the positive effect of actually precluding the occupational radiation exposure associated with performing the ILRT that is eliminated by the amendment.

6.0 REFERENCES

6.1 NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. (ADAMS Accession No. ML003740058)

6.2 Nuclear Energy Institute 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 2-A, dated October 2008 (including NRC Safety Evaluation Report dated June 25, 2008, ADAMS Accession No. ML081140105)

6.3 NRC letter to Nuclear Management Company and Safety Evaluation issuing amendment to facility operating license no. DPR-42 for Unit 1, dated August 20, 2004 (ADAMS Accession No. ML042090525)

Enclosure 2

Marked-Up Technical Specification Pages

1 page follows

5.5 Programs and Manuals (continued)

5.5.14 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. Unit 1 and Unit 2 are~~is~~ excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

2. Exception to NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J", Section 9.2.3, to allow the following:

(i). The first Unit 1 Type A test performed after December 1, 1997 shall be performed by December 1, 2012.

(ii). The first Unit 2 Type A test performed after March 7, 1997 shall be performed by March 7, 2012.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident is less than the containment internal design pressure, P_a, of 46 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.25% of primary containment air weight per day. For pipes connected to systems that are in the auxiliary building special ventilation zone, the total leakage shall be less than 0.1% of primary containment air weight per day at pressure P_a. For pipes connected to systems that are exterior to both the shield building and the auxiliary building special ventilation zone, the total leakage past isolation valves shall be less than 0.01% of primary containment air weight per day at pressure P_a.

Enclosure 3

Retyped Technical Specification Pages

1 page follows

5.5 Programs and Manuals (continued)

5.5.14 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. Unit 1 and Unit 2 are excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

2. Exception to NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J", Section 9.2.3, to allow the following:

(i). The first Unit 1 Type A test performed after December 1, 1997 shall be performed by December 1, 2012.

(ii). The first Unit 2 Type A test performed after March 7, 1997 shall be performed by March 7, 2012.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident is less than the containment internal design pressure, P_a, of 46 psig.

. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.25% of primary containment air weight per day. For pipes connected to systems that are in the auxiliary building special ventilation zone, the total leakage shall be less than 0.1% of primary containment air weight per day at pressure P_a. For pipes connected to systems that are exterior to both the shield building and the auxiliary building special ventilation zone, the total leakage past isolation valves shall be less than 0.01% of primary containment air weight per day at pressure P_a.