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January 9, 2002
JAFP-01-0246 Revision 1

T.A. Sullivan
Vice President, Operations-JAF

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station O-P1-17
Washington, DC 20555-0001

SUBJECT: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Proposed Change to the Technical Specifications Regarding Allowable MSIV Leakage (JPTS-01-004)

Dear Sir:

This application for an amendment to the James A. FitzPatrick Technical Specifications (TS) proposes a change to the Main Steam Isolation Valve (MSIV) Leakage Surveillance Requirement, 4.7.A.2.b, the associated Bases, and Technical Specification 6.20, Primary Containment Leakage Rate Testing Program. Specifically, this change establishes a total leakage rate limit for the sum of the four main steam lines that is equal to four times the current individual MSIV leakage rate limit.

Analyses that assume MSIV leakage do not consider individual valve leakage. These analyses consider the total primary containment leakage rate, which includes an allowance for total MSIV leakage rate. The current technical specification limit is less than or equal to 11.5 scfh for each MSIV when tested at greater than or equal to 25 psig. This value was derived by taking an analytically assumed MSIV leakage rate at the accident pressure (P_a) of 45 psig and interpolating the equivalent leakage rate at reduced pressure, nominally 25 psig. That leakage rate, 46 scfh, was then divided evenly between the four main steam lines and established as an individual MSIV leakage rate limit of 11.5 scfh for each MSIV.

The NRC has previously approved an amendment to the Technical Specifications for the Pilgrim Nuclear Power Station, Amendment 160 to License No. DPR-35, that allows use of a combined main steam line leakage rate limit.

The signed original of the Application for Amendment to the Operating License is enclosed for filing. Attachment I contains the proposed new TS pages and Attachment II is the Safety Evaluation for the proposed changes. A markup of the affected TS pages is included as Attachment III.

A copy of this application and the associated attachments are being provided to the designated New York State official in accordance with 10 CFR 50.91.

A001

BEFORE THE UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the Matter of)
Entergy Nuclear Operations, Inc.) Docket No. 50-333
James A. FitzPatrick Nuclear Power Plant)

<u>APPLICATION FOR AMENDMENT TO OPERATING LICENSE</u>

Entergy Nuclear Operations, Inc. requests an amendment to the Technical Specifications (TS) contained in Appendix A to Facility Operating License DPR-59 for the James A. FitzPatrick Nuclear Power Plant. This application is filed in accordance with Section 10 CFR 50.90 of the Nuclear Regulatory Commission's regulations.

This application for an amendment to the James A. FitzPatrick TS proposes a change to Surveillance Requirement 4.7.A.2.b, the associated Bases, and Technical Specification 6.20, Primary Containment Leakage Rate Testing Program. Specifically, this change establishes a combined leakage rate limit for the sum of the four main steam line leakage rates that is equal to four times the current individual MSIV leakage rate limit.

The signed original of the Application for Amendment to the Operating License is enclosed for filing. Attachment I contains the proposed new TS pages and Attachment II is the Safety Evaluation for the proposed changes. A markup of the affected TS pages is included as Attachment III.

Entergy Nuclear Operations, Inc.

T. A. Sullivan
Vice President, Operations-JAF

STATE OF NEW YORK
COUNTY OF OSWEGO
Subscribed and sworn to before me
this 22 day of [illegible] 2002.

Notary Public

DEANA ROSHAU
Notary Public, State of New York
No. 01RO6062748
Qualified in Oswego County
My Commission expires 08/13/05

This revision is being made to correct typographic errors in the original transmittal. The date should have been November 2, 2001 and in the Attachment II the mathematical symbols for greater than or equal to and less than or equal to were misprinted as different symbols. There are no changes to the technical content of this transmittal. There are no new commitments made in this letter. If you have any questions, please contact Mr. Andrew Halliday at 315-349-6055.

Very truly yours,

T. A. Sullivan

Attachments as stated

cc: Regional Administrator
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Attachment I to JAFP-01-0246, Revision 1

REVISED TECHNICAL SPECIFICATION PAGES
Proposed Change to the Technical Specifications Regarding Allowable MSIV Leakage

Entergy Nuclear Operations, Inc.
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

3.7 (cont'd)

(2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10°F above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.

(4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120°F.

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F, and fuel is in the reactor vessel, except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MWt.

4.7 (cont'd)

2. a. Perform required visual examination and leakage rate testing of the Primary Containment in accordance with the Primary Containment Leakage Rate Testing Program.

b. Demonstrate combined Main Steam Line leakage rate of $\leq$ 46 scfh when tested at $\geq$ 25 psig. The testing frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

c. Once per 24 months, demonstrate the leakage rate of 10AOV-68A,B for the Low Pressure Coolant Injection system and 14AOV-13A,B for the Core Spray system to be less than 11 scfm per valve when pneumatically tested at $\geq$ 45 psig at ambient temperature, or less than 10 gpm per valve if hydrostatically tested at $\geq$ 1,035 psig at ambient temperature.

4.7 BASES (cont'd)

assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide additional margin between expected offsite doses and 10CFR100 guidelines.

The leakage rate testing program was originally based on NRC guidelines for development of leak rate testing and surveillance schedules for reactor containment vessels. Containment structural integrity is currently verified with visual inspections and containment leak tightness is verified by the leakage rate surveillance testing described in the JAFNPP Primary Containment Leakage Rate Testing Program.

The following are the exemptions to 10 CFR 50 Appendix J, Option A, that have been approved by the NRC, and remain applicable to Option B of 10 CFR 50, Appendix J:

1. The Type C exceptions listed on Table 4.7-2, "Exception to Type C Test", as of the date of issuance of Amendment 194 (July 29, 1993).

2. Valves which are sealed with fluid from a seal system, such as the liquid in the suppression chamber are not required to be Type C tested. This exemption was approved by the NRC in the original Technical Specifications (SR 4.7.A.2.c(3)).

3. When MSIVs are tested at a pressure less than P_a and $\geq$ 25 psig, the limit for the combined main steam leakage rate is $\leq$ 46 scfh. The exemption for reduced pressure testing was approved by the NRC in the original Technical Specifications (Table 4.7-2).

The Program as implemented meets the requirements of Option B of 10 CFR 50 Appendix J (16) and Regulatory Guide 1.163 (13), with the exception stated in Specification 6.20. This exception applies to valves currently installed in this configuration, and does not apply to new installations. This exception is consistent with TS Table 4.7-2, previously contained in the TS, which allows reverse direction testing of valves as an exception to the requirements of the draft Appendix J, on the basis that pressurization direction was not a requirement at the time of plant design.

B. <u>Standby Gas Treatment System and</u>
C. <u>Secondary Containment</u>

Initiating reactor building isolation and operation of the Standby Gas Treatment System to maintain at least a 1/4 in. of water vacuum within the secondary containment provides an adequate test of the operation of the reactor

6.19 POSTACCIDENT SAMPLING PROGRAM

A program shall be established, implemented, and maintained which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

A) Training of personnel,

B) Procedures for sampling and analysis,

C) Provisions for maintenance of sampling and analysis

6.20 PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the Primary 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 exception that Type C testing of valves not isolable from the containment free air space may be accomplished by pressurization in the reverse direction provided that testing in this manner provides equivalent or more conservative results than testing in the accident direction. If potential atmospheric leakage paths (e.g., valve stem packing) are not subjected to test pressure, the portions of the valve not exposed to test pressure shall be subjected to leakage rate measurement during regularly scheduled Type A testing. A list of these valves, the leakage rate measurement method, and the acceptance criteria, shall be contained in the Program.

A. The peak Primary Containment internal pressure for the design basis loss of coolant accident (P_a), is 45 psig.

B. The maximum allowable Primary Containment leakage rate (L_a), at P_a, shall be 1.5% of primary containment air weight per day.

C. The leakage rate acceptance criteria are:

1. Primary containment leakage rate acceptance criteria 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 for the Type B and Type C tests and $\leq$ 0.75 L_a for the Type A tests;

2. Airlock testing acceptance criteria are:

   a. Overall airlock leakage rate is $\leq$ 0.05 L_a when tested at $\geq$ P_a,

   b. For each door seal, leakage rate is $\leq$ 120 scfd when tested at $\geq$ P_a.

3. The combined Main Steam Line leakage rate limit is $\leq$ 46 scfh.

D. The provisions of Specification 4.0.B do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

E. The provisions of Specification 4.0.C are applicable to the Primary Containment Leakage Rate Testing Program.

Attachment II to JAFP-01-0246, Revision 1

SAFETY EVALUATION

Proposed Change to the Technical Specifications Regarding Allowable MSIV Leakage

Entergy Nuclear Operations, Inc.
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

I. DESCRIPTION

This proposed change to the James A. FitzPatrick Nuclear Power Plant (JAF) Technical Specifications (TS) establishes a combined leakage rate limit for the sum of the four main steam line leakage rates that is equal to four times the current individual MSIV leakage rate limit.

The Specific changes are as follows:

Technical Specification Surveillance Requirement 4.7.A.2.b

Modify the surveillance requirement to reflect a combined total main steam line leakage rate limit:

> " Demonstrate combined main steam line leakage rate of ≤ 46 scfh when tested at a reduced pressure of ≥25 psig. The testing frequency is in accordance with the Primary Containment Leakage Rate Testing Program."

Bases for Technical Specification Surveillance Requirement 4.7.A.2.b

Revise Item 3 in the discussion of exemptions to read as follows:

> "When MSIVs are tested at a pressure less than P_a and ≥ 25 psig the limit for the combined main steam line leakage rate is ≤46 scfh. The exemption for reduced pressure testing was approved by the NRC in the original Technical Specifications (Table 4.7-2)"

Technical Specification 6.20 (Ref. 1) Section C.3

Modify the technical specification to reflect a combined total main steam line leakage rate limit:

> "3. The combined main steam line leakage rate limit is ≤ 46 scfh when tested at ≥ 25 psig."

II. PURPOSE OF THE PROPOSED CHANGE

The purpose of the proposed amendment is to allow alignment of the MSIV leakage rate limit and testing methodology with the plant design and licensing analyses. Currently leakage rate testing is accomplished using a combined leakage rate test. This test is performed by pressurizing the space between the MSIVs and measuring the combined leakage for the inboard and outboard MSIVs. The test is performed on each main steam line at reduced pressure (≥ 25 psig) and the results are compared to the individual MSIV leakage Rate Limit. Those results are then mathematically converted to the equivalent values that would be obtained by testing at P_a. The converted results are added together with other results from Type C Testing to establish a total primary containment leakage rate which is evaluated against the limit for total primary containment leakage rate. The proposed changes would address the combined total main steam line leakage rate as a single element of primary containment leakage rate consistent with the design and licensing basis analyses.

III. SAFETY IMPLICATIONS OF THE PROPOSED CHANGE

This proposed amendment does not change the limit for MSIV leakage rate used in any of the design or licensing basis analyses for JAF.

Each of JAF's four main steam lines contains two (inboard and outboard), quick-closing MSIVs. UFSAR Section 4.6.1 describes the safety functions of the MSIVs. The MSIVs prevent damage to the fuel barrier by limiting the loss of reactor coolant and limit release of radioactive materials by maintaining the primary containment barrier.

In the case of a LOCA, as evaluated in UFSAR Section 14.6.1.3, the MSIVs isolate the reactor from the environment and prevent the direct release of fission products from the containment. In the case of a steam line break, as evaluated in FSAR Section 14.6.1.5, closure of the MSIVs terminates the blowdown of reactor steam in sufficient time to prevent an uncontrolled release of radioactivity from the reactor vessel to the environment.

UFSAR Section 14.8.2.1.1 makes the following statement regarding MSIV Leakage rate:

> "(d) Leakage from the drywell is at the rate of 1.5% per day, consisting of 1.27 % per day due to containment leakage and 0.23 % per day due to MSIV leakage, and is constant for the duration of the accident (31 days)."

FSAR Tables 14.8-8 through 14.8-11 summarize the postulated radiation dose for the design basis accident scenarios, as calculated in JAF-CALC-RAD-00042 (Ref. 2) and JAF-CALC-RAD-00048 (Ref.3). Each of these calculations assumed leakage from the drywell at a rate of 1.5% per day, and that the drywell leakage included the MSIV leakage.

In summary, the design analyses quantify a maximum volume of primary containment atmosphere that can bypass the secondary containment and leak directly to the environment following a design basis LOCA. This volume is then factored in to the radiation dose analyses that are evaluated against the radiation dose guidelines contained in 10 CFR Part 100 for offsite and 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 19 for control room occupancy. The assumed primary containment leakage rate in these analyses includes the total main steam line leakage rate rather than individual MSIV leakage rates. 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" establishes requirements for testing to ensure that the containment is operated within the design analyses. The Technical Specification leakage rate limits provide empirical evidence of operation within the design base. The current Technical Specification limit for MSIV leakage rate is ≤ 11.5 scfh for each MSIV when tested ≥ 25 psig. This value was derived by taking an analytically assumed MISV leakage rate at the assumed accident pressure of 45 psig (P_a) and interpolating the equivalent leakage rate at reduced pressure, nominally 25 psig. That leakage rate, 46 scfh, was then conservatively divided evenly between the four main steam lines and established as an individual MSIV leakage rate limit. Therefore, the proposed changes actually make the consideration of MSIV leakage rate more in line with the design and licensing bases.

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Operation of the JAF plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92 since it would not:

Involve an increase in the probability or consequences of an accident previously evaluated.

> The proposed amendment does not involve a change to structures, components, or systems that would affect the probability of an accident previously evaluated in the FitzPatrick Updated Final Safety Analysis Report (UFSAR). The proposed amendment results in no change in radiological consequences of the design basis LOCA as currently analyzed for the FitzPatrick Plant. These analyses were calculated assuming a combined total MSIV leakage at accident pressure for determining acceptance to the regulatory limits for the offsite, control room, and Technical Support Center (TSC) radiation doses as contained in 10 CFR 100 and 10 CFR 50, Appendix A, GDC 19. The proposed change does not compromise existing radiological equipment qualification, since the combined total MSIV leakage rate has been factored into existing equipment qualification analyses for 10 CFR 50.49.

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

> The proposed change does not modify the MSIVs or any other plant system or structure associated with this amendment and therefore, will not affect their capability to perform their design function. The combined total main steam line leakage rate is included in the current radiological analyses for the assessment of radiation exposure following an accident. This proposal changes the allowable leakage rate from a per valve limit to a total combined leakage rate limit for all four main steam lines but does not change the cumulative limit. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously analyzed.

Involve a significant reduction in a margin of safety.

> The leakage rate limit specified for the MSIVs is used to quantify the maximum amount of bypass leakage assumed in the LOCA radiological analysis. Results of the analysis are evaluated against the dose guidelines contained in GDC 19 and 10 CFR 100. The margin of safety in this context is considered to be the difference between the calculated dose exposures and the guidelines provided by the GDC 19 and 10 CFR 100. Therefore, since the proposed combined total main steam line leakage rate limit is unchanged from the assumed maximum leakage rate for MSIVs, for the purpose of calculating potential radiation dose, the margin of safety is not affected because the postulated radiation doses remain the same.

V. IMPLEMENTATION OF THE PROPOSED CHANGE

This amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) as follows:

(i) The amendment involves no significant hazards consideration.

As described in Section IV of this evaluation, the proposed change involves no significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed change does not involve the installation of any new equipment, or the modification of any equipment that may affect the types or amounts of effluents that may be released offsite. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes do not involve plant physical changes, or introduce any new mode of plant operation. Therefore, there is no significant increase in individual or cumulative occupational radiation exposure.

Based on the above, Entergy Nuclear Operations, Inc. concludes that the proposed changes meet the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.21 relative to requiring a specific environmental assessment by the Commission.

VI. CONCLUSION

The Plant Operating Review Committee (PORC) and Safety Review Committee (SRC) have reviewed this proposed change to the TS and have concluded that it does not involve a significant hazards consideration and will not endanger the health and safety of the public.

VII. REFERENCES

1. Technical Specification 6.20 Primary Containment Leakage Rate Testing Program

2. JAF-CALC-RAD-00042 Rev. 3B "Control Room Radiological Habitability Under Power Uprate Conditions and CREVASS Reconfiguration"

3. JAF-CALC-RAD-00048 Rev. 2A "Power Uprate Project – Radiological Impact at Onsite and Offsite Outdoor Receptors Following Design-Basis Accidents"

4. Pilgrim Nuclear Power Station Technical Specification Amendment 160

Attachment III to JAFP-01-0246, Revision 1

MARKED-UP TECHNICAL SPECIFICATION PAGES

Proposed Change to the Technical Specifications Regarding Allowable MSIV Leakage

Entergy Nuclear Operations, Inc.
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

3.7 (cont'd)

(2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10°F above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.

(4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120°F.

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F, and fuel is in the reactor vessel, except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MWt.

4.7 (cont'd)

2. a. Perform required visual examination and leakage rate testing of the Primary Containment in accordance with the Primary Containment Leakage Rate Testing Program.

b. ~~Demonstrate leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig.~~ The testing frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

REPLACE

Demonstrate combined Main Steam Line Leakage Rate of ≤ 46 scfh when tested at a reduced pressure of ≥ 25 psig.

c. Once per 24 months, demonstrate the leakage rate of 10AOV-68A,B for the Low Pressure Coolant Injection system and 14AOV-13A,B for the Core Spray system to be less than 11 scfm per valve when pneumatically tested at ≥ 45 psig at ambient temperature, or less than 10 gpm per valve if hydrostatically tested at ≥ 1,035 psig at ambient temperature.

4.7 BASES (cont'd)

assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide additional margin between expected offsite doses and 10CFR100 guidelines.

The leakage rate testing program was originally based on NRC guidelines for development of leak rate testing and surveillance schedules for reactor containment vessels. Containment structural integrity is currently verified with visual inspections and containment leak tightness is verified by the leakage rate surveillance testing described in the JAFNPP Primary Containment Leakage Rate Testing Program.

The following are the exemptions to 10 CFR 50 Appendix J, Option A, that have been approved by the NRC, and remain applicable to Option B of 10 CFR 50, Appendix J:

1. The Type C exceptions listed on Table 4.7-2, "Exception to Type C Test", as of the date of issuance of Amendment 194 (July 29, 1993).

2. Valves which are sealed with fluid from a seal system, such as the liquid in the suppression chamber are not required to be Type C tested. This exemption was approved by the NRC in the original Technical Specifications (SR 4.7.A.2.c(3)).

When MSIVs are tested at a pressure less than Pa and ≥ 25 psig the limit for the combined main steam line leakage rate is ≤ 46 scfh. The exemption for reduced pressure testing was approved by the NRC in the original Technical Specifications (Table 4.7-2).

REPLACE

3. ~~The MSIVs are tested at a pressure less than P_a and ≥ 25 psig, with a leakage rate acceptance criteria of ≤ 11.5 scfh per valve. This exemption was approved by the NRC in the original Technical Specifications (Table 4.7-2).~~

The Program as implemented meets the requirements of Option B of 10 CFR 50 Appendix J (16) and Regulatory Guide 1.163 (13), with the exception stated in Specification 6.20. This exception applies to valves currently installed in this configuration, and does not apply to new installations. This exception is consistent with TS Table 4.7-2, previously contained in the TS, which allows reverse direction testing of valves as an exception to the requirements of the draft Appendix J, on the basis that pressurization direction was not a requirement at the time of plant design.

B. C. <u>Standby Gas Treatment System and Secondary Containment</u>

Initiating reactor building isolation and operation of the Standby Gas Treatment System to maintain at least a 1/4 in. of water vacuum within the secondary containment provides an adequate test of the operation of the reactor

6.19 POSTACCIDENT SAMPLING PROGRAM

A program shall be established, implemented, and maintained which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

A) Training of personnel,

B) Procedures for sampling and analysis,

C) Provisions for maintenance of sampling and analysis

6.20 PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the Primary 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 exception that Type C testing of valves not isolable from the containment free air space may be accomplished by pressurization in the reverse direction provided that testing in this manner provides equivalent or more conservative results than testing in the accident direction. If potential atmospheric leakage paths (e.g., valve stem packing) are not subjected to test pressure, the portions of the valve not exposed to test pressure shall be subjected to leakage rate measurement during regularly scheduled Type A testing. A list of these valves, the leakage rate measurement method, and the acceptance criteria, shall be contained in the Program.

A. The peak Primary Containment internal pressure for the design basis loss of coolant accident (P_a), is 45 psig.

B. The maximum allowable Primary Containment leakage rate (L_a), at P_a, shall be 1.5% of primary containment air weight per day.

C. The leakage rate acceptance criteria are:

1. Primary containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for the Type A tests;

2. Airlock testing acceptance criteria are:

   a. Overall airlock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   b. For each door seal, leakage rate is $\leq$ 120 scfd when tested at $\geq P_a$.

3. ~~MSIV leakage rate acceptance criteria is $\leq$ 11.5 scfh for each MSIV~~ when tested at $\geq$ 25 psig.

REPLACE

D. The provisions of Specification 4.0.B do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

E. The provisions of Specification 4.0.C are applicable to the Primary Containment Leakage Rate Testing Program.

The combined Main Steam Line Leakage rate limit is $\leq$ 46 scfh