AmerGen

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10 CFR 50.90

2130-03-20308
December 23, 2003

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

Oyster Creek Generating Station
Facility Operating License No. DPR-16
NRC Docket No. 50-219

Subject: Technical Specification Change Request No. 317 - Main Steam Isolation Valve (MSIV) Local Leak Rate Test (LLRT) Pressure

Pursuant to 10 CFR 50.90 AmerGen Energy Company, LLC (AmerGen), hereby requests the following amendment to the Technical Specifications (TS), Appendix A of Operating License No. DPR-16 for Oyster Creek Generating Station (OCGS).

This proposed change will revise OCGS Technical Specifications 4.5.D, and associated Bases, to allow the Main Steam Isolation Valves (MSIVs) to be leak rate tested at a pressure less than P_a. Using the standards in 10 CFR 50.92, AmerGen has concluded that these proposed changes do not constitute a significant hazards consideration, as described in the enclosed analysis performed in accordance with 10 CFR 50.91(a)(1).

AmerGen requests NRC approval of this change by October 15, 2004, based on the scheduled start date of the 1R20 refueling outage.

No new regulatory commitments are established by this submittal.

These proposed changes have been reviewed by the OCGS Plant Operations Review Committee and approved by the Nuclear Safety Review Board.

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Pursuant to 10 CFR 50.91(b)(1), a copy of this Technical Specification Change Request is being provided to the designated official of the State of New Jersey, Bureau of Nuclear Engineering, as well as the Chief Executive of the township in which the facility is located.

If any additional information is needed, please contact Mr. Dave Helker at 610-765-5525.

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

Respectfully,

12-23-03
Executed on

Michael P. Gallagher
Michael P. Gallagher
Director, Licensing and Regulatory Affairs
AmerGen Energy Company, LLC

Attachments: 1-Description of Proposed Changes
2-Markup of Proposed Technical Specification Pages
3-Markup of Proposed Technical Specification Bases Pages
4-Retyped Technical Specification Pages
5-Retyped Technical Specification Bases Pages

cc: H.J. Miller, Administrator, USNRC Region 1
P. S. Tam, USNRC Senior Project Manager, Oyster Creek
R. Summers, USNRC Senior Resident Inspector, Oyster Creek
File No. 01036

ATTACHMENT 1 CONTENTS

SUBJECT: Revision to Technical Specifications 4.5.D and associated Bases, to allow the Main Steam Isolation Valves (MSIVs) to be leak rate tested at a pressure less than P_a.

1.0 DESCRIPTION

This letter is a request to amend Operating License DPR-16 for the Oyster Creek Generating Station (OCGS).

The proposed change would revise the Operating License to allow the Main Steam Isolation Valves (MSIVs) to be leak rate tested at a pressure less than P_a. A corresponding Exemption to 10 CFR 50, Appendix J, Option B, has been developed and is being submitted via separate letter in accordance with 10 CFR 50.12. AmerGen Energy Company, LLC (AmerGen) requests approval of this Technical Specifications change request by October 15, 2004, based on the scheduled start date of the 1R20 refueling outage.

2.0 PROPOSED CHANGE

Oyster Creek Technical Specification 4.5.D.2 currently specifies that the leakage rate acceptance criteria for an MSIV shall be $0.05(0.75)L_a$ at P_a (L_a is the maximum allowable Primary Containment leakage rate). The proposed change revises 4.5.D.2 to state "verify leakage rate through each MSIV is ≤ 11.9 scfh when tested at ≥ 20 psig."

For consistency, an editorial change is proposed to Technical Specification 4.5.D.1 ("except as stated in Specification 4.5.D.2") to clarify that MSIVs will not be tested at P_a as currently required in 4.5.D.1.

Minor changes to the Technical Specification Bases are proposed to add reference to the Primary Containment Leakage Rate Testing Program "as modified by approved exemptions."

3.0 BACKGROUND

The Oyster Creek Nuclear Steam Supply System (NSSS) supplies steam from the reactor through two main steam lines to drive the Turbine Generator. Main steam line isolation is accomplished by means of the Main Steam Isolation Valves (MSIVs). The MSIVs are containment isolation valves designed to minimize coolant loss from the vessel and thus offsite doses in the event of a main steam line break accident. Two isolation valves are installed in each of the two 24 inch main steam lines in parallel horizontal runs that penetrate the drywell through 36 inch diameter openings. Valves V-1-7 and V-1-8 are located inside the drywell; Valves V-1-9 and V-1-10 are located in the Secondary Containment (trunnion room), beyond the drywell wall. Both sets of valves are located as close as possible to the drywell penetrations. The basic design of the four valves is identical. The valves are 24 inch angled globe valves of "Y" configuration. The cup shaped poppet moves on a centerline that is 45° upward from the horizontal centerline of the piping run. The valves in the inboard and outboard sets are rotated inward toward each other (at 22°30' from vertical) so that the air cylinders clear downcoming steam lines and other neighboring lines. Each MSIV is oriented to provide effective sealing in the direction of post-accident containment atmosphere leakage, i. e., the forward direction, as compared to the between-the-valve test which tends to unseat the inboard valve. The design of the steam lines is such that the preferred method of testing is through the use of a between-the-valve test tap. The Main Steam Isolation Valves are described in UFSAR Section 5.4.5.

4.0 TECHNICAL ANALYSIS

Technical Specifications sections 4.5.A, 4.5.B, 4.5.C and 4.5.D provide containment and local leakage test requirements for OCGS as required by 10 CFR 50, Appendix J, Option B. The underlying purpose is to demonstrate by periodic testing that the primary reactor containment will be able to perform its function of providing a leak tight barrier against the uncontrolled release of radioactivity to the environment. The proposed MSIV leak rate test will meet the underlying purpose of the regulation.

Each MSIV is oriented to provide effective sealing in the direction of post-accident containment atmosphere leakage. The MSIVs are periodically leak rate tested between the valves to verify that the leakage assumed in the radiological analysis is not exceeded per Technical Specifications 4.5.C and 4.5.D.

This TS change is being requested in order to reduce the probability of lifting the inboard MSIVs during testing. As discussed later, lifting of the inboard MSIV due to the test method creates undue hardship during outages by potentially requiring the installation of plugs at the inside steam line nozzle penetrations.

Testing of the two valves simultaneously at P_a, by pressurizing between the valves, tends to lift the disc of the inboard valve. This results in a test which may not accurately reflect the isolation capabilities of the valves. A review of the test database back to refueling outage 12 in 1988 has identified 3 instances of inconsistent test results, indicating the inboard valve was unseating due to the reverse test pressure, with the most recent occurrence in the last outage (1R19 in 2002). These results required retests using more challenging test methods (e.g., via steam line plugs) to determine that the valves were in satisfactory condition. After retest, the valves were determined to be acceptable.

The proposed test calls for pressurizing between the MSIVs at greater than one-half (20 psig) of P_a to avoid lifting the disc of the inboard valve. The measured leakage rate for any one main steam line through the isolation valves shall be limited to a pathway leakage value of 11.9 scfh, in accordance with Oyster Creek Technical Specifications. This is the value used in the radiological analysis for control room habitability as discussed in Section 6.4 of the OCGS FSAR (Reference 4). A summary of the control room habitability analysis was supplied to the NRC in the Reference 5 letter, and was approved in an NRC safety evaluation in the Reference 6 letter. As discussed in the Reference 5 letter, although the MSIVs are designed to provide a leak-tight barrier, some leakage through the valves will occur. The leakage limit discussed in the Reference 5 letter (11.9 scfh) is the value proposed for this TS change. Offsite (exclusion area boundary and low population zone) doses are well below the 10 CFR Part 100 limits and are unaffected by this change. Additionally, as shown in Figure 6.2-3 of the OCGS FSAR, the primary containment pressure following a LOCA reaches its peak within 2 to 3 seconds, and rapidly drops below 20 psig. Therefore, establishing the Technical Specification test pressure at ≥ 20 psig is acceptable.

The 11.9 scfh acceptance criteria will be effective and reliable in determining the status of the MSIVs, and in verifying that substantial degradation of these valves has not occurred since the last Integrated Leakage Rate Test (ILRT). The 11.9 scfh leakage limit is more conservative than the value calculated by adjusting the existing 35 psig leakage limit to 20 psig, per ASME Code. Additionally, the leakage path through the MSIVs is included during an ILRT, and therefore the effect of this leakage on containment integrity is taken into account.

Testing of the inboard MSIVs in the forward direction when the reactor vessel head is removed requires the installation of plugs at the inside steam line nozzle penetrations. Experience has shown that these plugs will hold pressure when subjected to the 35 psig Local Leak Rate Test (LLRT) pressure, however, the large test volume created includes the Main Steam Safety Valves, Electromatic Relief Valves, and Main Steam Line Drain piping. These increase the number of potential leak paths, thereby complicating the methodology and implementation of the test. As a result, this method of testing is difficult to implement, and provides conservative leakage results which may not accurately reflect the leak tightness of the MSIVs.

A second alternative to the proposed test would be to install a 24 inch block valve and a one inch test tap in each of the 2 main steam lines in order to Type C test the inboard MSIV in the forward direction. This modification would require significant monetary expenditures to implement without a commensurate increase in safety levels. This design would also require the local leak rate test to be conducted inside the drywell as opposed to the outboard MSIV room (trunnion room), thus, subjecting those performing the test to higher radiation doses. Therefore, these modifications would result in undue hardship due to the costs incurred without a commensurate increase in safety levels and the hardships incurred as a result of the increased radiation exposure to plant personnel. Additionally, application of these modifications would not be necessary to serve the underlying purpose of the rule, which is to ensure that the primary containment serves as an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment.

In addition to the above discussion, in an NRC Memorandum (from C.Y. Shiraki (Project Directorate III-2) to NRR Project Directors, "10 CFR Part 50, Appendix J, Option B; Technical Specifications Amendments", dated March 13, 1996) guidance was provided regarding the processing of Technical Specification Amendments that request implementation of 10 CFR 50 Appendix J, Option B. This Memorandum states as follows:

> "However, there is one exemption to Option A, commonplace in Boiling Water Reactors (BWR), that will be applicable to Option B. Most BWRs have exemptions to Option A that allow the main steam isolation valves (MSIV) to be Type C tested at a pressure less than Pa (usually one-half of Pa), and often with a separate leakage rate limit or acceptance criterion for the MSIVs. Option B, Section II.B., "Type B and C Tests," states: "The tests must demonstrate that the sum of the leakage rates at accident pressure of Type B tests, and ... Type C tests, is less than the performance criterion (La) with margin, as specified in the Technical Specification.".... *Further, in almost all BWRs, the MSIVs are tested at a pressure less than Pa, the "accident pressure."* Thus, for most BWRs, the MSIV exemption to Option A will still be needed and will be applicable to Option B."

This Memorandum demonstrates that the requested Technical Specifications changes are supported by an NRC exemption that is typical for BWRs, and an accepted NRC exemption that would apply to the implementation of Option A or Option B of 10 CFR 50, Appendix J. Although this Memorandum infers that the majority of BWR's have this exemption and associated technical specification changes, OCGS had not requested this exemption.

Further, specific exemption requests supporting this type of testing have been granted at Limerick Generating Station (Safety Evaluation Report (SER) Section 6.2.6.1 (NUREG-0991, Safety Evaluation Report (SER), Section 6.2.6.1, dated December 1983), Peach Bottom Atomic Power Station ("Appendix J Exemption for Peach Bottom Atomic Power Station, Units 2 and 3 (TAC NOS. 76195 and 76196)", dated November 21, 1990), Dresden Nuclear Power Stations, Units 2 and 3 (NRC SER dated June 25, 1982), and Quad Cities Nuclear Power Station, Units 1 and 2 (NRC SER dated June 12, 1984).

In conclusion, the proposed MSIV local leak rate test pressure (20 psig), is technically acceptable in lieu of P_a (35 psig) due to the unique design and orientation of the MSIVs. The reduced pressure test will equally determine the condition of the MSIVs, and their ability to maintain containment isolation integrity during an accident.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

AmerGen has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

Response: No.

The proposed revisions to Technical Specifications 4.5.D will revise the test pressure at which the MSIVs are tested as part of the Appendix J, Option B containment testing program.

The design of the MSIVs is such that testing in the reverse direction tends to unseat the valve. Testing of the two valves simultaneously at peak containment pressure, by pressurizing between the valves, tends to lift the disc of the inboard valve. This results in a test which may not accurately reflect the isolation capabilities of the valves, based on the design of the valve which has better seating capability in the forward direction. This change does not increase the Primary Containment Leakage Rate allowable limits defined in Technical Specification 4.5.D. The proposed alternative leak rate testing method will equally determine the condition of the MSIVs and their ability to maintain containment isolation integrity during an accident.

The measured leakage rate for any one main steam line through the isolation valves shall be limited to a pathway leakage value of 11.9 scfh, in accordance with Oyster Creek Technical Specifications. This is the value used in the radiological analysis for control room habitability as discussed in Section 6.4 of the OCGS FSAR. Offsite (exclusion area boundary and low population zone) doses are well below the 10 CFR Part 100 limits and are unaffected by this change. The 11.9 scfh acceptance criteria will be an effective and reliable in determining the status of the MSIVs, and in verifying that substantial degradation of these valves has not occurred since the last Integrated Leakage Rate Test (ILRT). Additionally, the leakage path through the MSIVs is included during an ILRT, and therefore the effect of this leakage on containment integrity is taken into account. Therefore, this change has no effect on the consequences of an accident previously evaluated.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed revision to Technical Specifications 4.5.D to revise the MSIV test pressure and leakage rate does not create the possibility of a new or different type of accident.

The proposed acceptance criteria is effective and reliable in determining the status of the MSIVs, and in verifying that substantial degradation of these valves has not occurred since the last Integrated Leakage Rate Test (ILRT). This change has no affect on the design and operation of plant structures, systems, and components. This change does not introduce any new accident precursors and does not involve any alterations to plant configurations, which could initiate a new or different kind of accident.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

   Technical Specifications section 4.5.D provides containment and local leakage test requirements for OCGS as required by 10 CFR 50, Appendix J, Option B. The underlying purpose is to demonstrate by periodic testing that the primary reactor containment will be able to perform its function of providing a leak tight barrier against the uncontrolled release of radioactivity to the environment. The proposed MSIV leak rate test will meet the underlying purpose of the regulation. This change does not increase the Primary Containment Leakage Rate allowable limits defined in Technical Specification 4.5.D. The proposed alternative leak rate testing method will equally determine the condition of the MSIVs and their ability to maintain containment isolation integrity during an accident.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

For the reasons stated above, the proposed Technical Specifications changes do not present undue risk to the public health and safety because the proposed alternative testing will equally determine the condition of the MSIVs and their ability to maintain containment isolation integrity during an accident.

5.2 Applicable Regulatory Requirements/Criteria

Technical Specifications sections 4.5.A, 4.5.B, 4.5.C and 4.5.D provide containment and local leakage test requirements for OCGS as required by 10 CFR 50, Appendix J, Option B. The underlying purpose is to demonstrate by periodic testing that the primary reactor containment will be able to perform its function of providing a leak tight barrier against the uncontrolled release of radioactivity to the environment. The proposed MSIV leak rate test will meet the underlying purpose of the regulation. Concurrent with this Technical Specification Change Request, AmerGen is requesting (via separate letter) an exemption from the requirements of 10 CFR 50, Appendix J, Option B, Paragraph III.B. Paragraph III.B requires leak rate testing of the MSIVs at P_a. Exemption from paragraph III.B of Appendix J, Option B, is consistent with the proposed Oyster Creek Technical Specification change submittal regarding this issue. This change is also consistent with the "Standard Technical Specifications, General Electric Plants, BWR/4", NUREG-1433, Revision 1, April 1995, in that Surveillance Requirement 3.6.1.3.13, concerning verification of leakage rate through each MSIV, permits exemptions to 10 CFR 50 Appendix J. Further, the exemption is consistent with regulatory practice and policy, as evidenced by the granting of a similar exemptions for Limerick Generating Station (Safety Evaluation Report (SER) Section 6.2.6.1 (NUREG-0991, Safety Evaluation Report (SER), Section 6.2.6.1, dated December 1983), Peach Bottom Atomic Power Station ("Appendix J Exemption for Peach Bottom Atomic Power Station, Units 2 and 3 (TAC NOS. 76195 and 76196)", dated November 21, 1990), Dresden Nuclear Power Stations, Units 2 and 3 (NRC SER dated June 25, 1982), and Quad Cities Nuclear Power Station, Units 1 and 2 (NRC SER dated June 12, 1984).

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.

6.0 ENVIRONMENTAL CONSIDERATION

The proposed test calls for pressurizing between the MSIVs at greater than one-half (20 psig) of P_a to avoid lifting the disc of the inboard valve. The measured leakage rate for any one main steam line through the isolation valves shall be limited to a pathway leakage value of 11.9 scfh, in accordance with Oyster Creek Technical Specifications. This is the value used in the radiological analysis for control room habitability as discussed in Section 6.4 of the OCGS FSAR (Reference 4). A summary of the control room habitability analysis was supplied to the NRC in the Reference 5 letter, and was approved in an NRC safety evaluation in the Reference 6 letter. A review has determined that the proposed Technical Specifications change would change a requirement with respect to a surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Safety Evaluation Report (SER) for Limerick Generating Station, Units 1 and 2, NUREG-0991, Section 6.2.6.1, dated December 1983.

2. Letter from G. Y. Suh (U. S. Nuclear Regulatory Commission) to G.A. Hunger, Jr. (Philadelphia Electric Company), "Appendix J Exemption for Peach Bottom Atomic Power Station, Units 2 and 3 (TAC NOS. 76195 and 76196)", dated November 21, 1990.

3. NRC Memorandum from C.Y. Shiraki (Project Directorate III-2) to NRR Project Directors, "10 CFR Part 50, Appendix J, Option B; Technical Specifications Amendments", dated March 13, 1996.

4. Oyster Creek Updated Final Safety Analysis Report, Section 6.4, "Habitability Systems."

5. Letter from R. F. Wilson (GPU Nuclear Corporation) to J. A. Zwolinski (NRC), "Control Room Habitability (NUREG-0737 Item III.D.3.4) Results of Whole Body and Beta Skin Dose Analysis", dated June 17, 1985.

6. Letter from NRC to P. B. Fiedler (Oyster Creek Generating Station), "Control Room Habitability (TAC 46466, 57905)", dated July 15, 1986.

7. Letter from D. G. Eisenhut (U. S. Nuclear Regulatory Commission) to L. DelGeorge (Commonwealth Edison Company), NRC Safety Evaluation Report, Dresden Nuclear Power Station, Units 2 and 3, dated June 25, 1982.

8. Letter from D. B. Vassallo (U. S. Nuclear Regulatory Commission) to D. L. Farrar (Commonwealth Edison Company), NRC Safety Evaluation Report, Quad Cities Nuclear Power Station, Units 1 and 2, dated June 12, 1984.

ATTACHMENT 2

Oyster Creek Technical Specification Change Request No. 317

Markup of Proposed Technical Specification Pages

Revised TS Page

4.5.2

b. If the airlock is opened during a period when Primary Containment is not required, it need not be tested while Primary Containment is not required, but must be tested at P_a prior to returning the reactor to an operating mode requiring PRIMARY CONTAINMENT INTEGRITY.

D. Primary Containment Leakage Rates shall be limited to:

, except as stated in Specification 4.5.D.2.

1. The maximum allowable Primary Containment leakage rate is 1.0 L_a. The maximum allowable Primary Containment leakage rate to allow for plant startup following a type A test is 0.75 L_a. The leakage rate acceptance criteria for the Primary Containment Leakage Rate Testing Program for Type B and Type C tests is $\leq$0.60 L_a at P_a.

Verify leakage rate through each MSIV is $\leq$ 11.9 scfh when tested at $\geq$ 20 psig.

2. ~~The leakage rate acceptance criteria for an MSIV shall be 0.05(0.75) L_a at P_a.~~

3. The leakage rate acceptance criteria for the drywell airlock shall be $\leq$0.05 L_a when measured or adjusted to P_a.

E. Continuous Leak Rate Monitor

1. When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements.

2. This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical.

F. Functional Test of Valves

1. All automatic primary containment isolation valves shall be tested for automatic closure by an isolation signal during each REFUELING OUTAGE and the isolation time determined to be within its limit. The following valves are required to close in the time specified below:

Main steam line isolation valves: $\geq$ 3 seconds and $\leq$ 10 seconds

2. Each automatic primary containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on

ATTACHMENT 3

Oyster Creek Technical Specification Change Request No. 317

Markup of Proposed Technical Specification Bases Pages

Revised Bases Page

4.5.11

as modified by approved exemptions.

A Primary Containment Leakage Rate Testing Program has been established to implement the requirements of 10 CFR 50, Appendix J, Option B. Guidance for implementation of Option B is contained in NRC Regulatory Guide 1.163, "Performance Based Containment Leak Test Program", Revision 0, dated September 1995. Additional guidance for NRC Regulatory Guide 1.163 is contained in Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance Based Option of 10 CFR 50, Appendix J, "Revision 0, dated July 26, 1995, and ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements". The Primary Containment Leakage Rate Testing Program conforms with this guidance.

The maximum allowable leakage rate for the primary containment (L_a) is 1.0 percent by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a). As discussed below, P_a for the purpose of containment leak rate testing is 35 psig.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a double gasketed penetration (primary containment head equipment hatches and the absorption chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 35 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time be kept to a practical minimum.

Automatic primary containment isolation valves are provided to maintain PRIMARY CONTAINMENT INTEGRITY following the design basis loss-of-coolant accident. Closure times for the automatic primary containment isolation valves are not critical because it is on the order of minutes before significant fission product release to the containment atmosphere for the design basis loss of coolant accident. These valves are highly reliable, see infrequent service and most of them are normally in the closed position. Therefore, a test during each REFUELING OUTAGE is sufficient.

Large lines connecting to the reactor coolant system, whose failure could result in uncovering the reactor core, are supplied with automatic isolation valves (except containment cooling). Closure times restrict coolant loss from the circumferential rupture of any of these lines outside primary containment to less than that for a main steam line break (the design basis accident for outside containment line breaks). The minimum time for main steam isolation valve (MSIV) closure of 3 seconds is based on the transient analysis that shows the pressure peak 76 psig below the lowest safety valve setting. The maximum time for MSIV closure of 10 seconds is based on the value assumed for the main steam line break dose calculations and restricts coolant loss to prevent uncovering the reactor core. Per ASME Boiler and Pressure Vessel Code, Section XI, the full closure test of the MSIVs during COLD SHUTDOWNs will ensure OPERABILITY and provide assurance that the valves maintain the required closing time. The provision for a minimum of 92 days between the tests ensures that full closure testing is not too frequent. The MSIVs are partially stroked quarterly as part of reactor protection system instrument surveillance testing.

ATTACHMENT 4

Oyster Creek Technical Specification Change Request No. 317

Retyped Technical Specification Pages

Revised TS Page

4.5.2

b. If the airlock is opened during a period when Primary Containment is not required, it need not be tested while Primary Containment is not required, but must be tested at P_a prior to returning the reactor to an operating mode requiring PRIMARY CONTAINMENT INTEGRITY.

D. Primary Containment Leakage Rates shall be limited to:

1. The maximum allowable Primary Containment leakage rate is 1.0 L_a. The maximum allowable Primary Containment leakage rate to allow for plant startup following a type A test is 0.75 L_a. The leakage rate acceptance criteria for the Primary Containment Leakage Rate Testing Program for Type B and Type C tests is $\leq$0.60 L_a at P_a, except as stated in Specification 4.5.D.2.

2. Verify leakage rate through each MSIV is ≤ 11.9 scfh when tested at ≥ 20 psig.

3. The leakage rate acceptance criteria for the drywell airlock shall be ≤ 0.05 L_a when measured or adjusted to P_a.

E. Continuous Leak Rate Monitor

1. When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements.

2. This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical.

F. Functional Test of Valves

1. All automatic primary containment isolation valves shall be tested for automatic closure by an isolation signal during each REFUELING OUTAGE and the isolation time determined to be within its limit. The following valves are required to close in the time specified below:

   Main steam line isolation valves: ≥ 3 seconds and ≤ 10 seconds

2. Each automatic primary containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on

ATTACHMENT 5

Oyster Creek Technical Specification Change Request No. 317

Retyped Technical Specification Bases Changes

<u>Revised TS Page</u>

4.5.11

A Primary Containment Leakage Rate Testing Program has been established to implement the requirements of 10 CFR 50, Appendix J, Option B, as modified by approval exemptions. Guidance for implementation of Option B is contained in NRC Regulatory Guide 1.163, "Performance Based Containment Leak Test Program", Revision 0, dated September 1995. Additional guidance for NRC Regulatory Guide 1.163 is contained in Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance Based Option of 10 CFR 50, Appendix J," Revision 0, dated July 26, 1995, and ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements." The Primary Containment Leakage Rate Testing Program conforms with this guidance as modified by approval exemptions.

The maximum allowable leakage rate for the primary containment (L_a) is 1.0 percent by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a). As discussed below, P_a for the purpose of containment leak rate testing is 35 psig.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a double gasketed penetration (primary containment head equipment hatches and the absorption chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 35 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time be kept to a practical minimum.

Automatic primary containment isolation valves are provided to maintain PRIMARY CONTAINMENT INTEGRITY following the design basis loss-of-coolant accident. Closure times for the automatic primary containment isolation valves are not critical because it is on the order of minutes before significant fission product release to the containment atmosphere for the design basis loss of coolant accident. These valves are highly reliable, see infrequent service and most of them are normally in the closed position. Therefore, a test during each REFUELING OUTAGE is sufficient.

Large lines connecting to the reactor coolant system, whose failure could result in uncovering the reactor core, are supplied with automatic isolation valves (except containment cooling). Closure times restrict coolant loss from the circumferential rupture of any of these lines outside primary containment to less than that for a main steam line break (the design basis accident for outside containment line breaks). The minimum time for main steam isolation valve (MSIV) closure of 3 seconds is based on the transient analysis that shows the pressure peak 76 psig below the lowest safety valve setting. The maximum time for MSIV closure of 10 seconds is based on the value assumed for the main steam line break dose calculations and restricts coolant loss to prevent uncovering the reactor core. Per ASME Boiler and Pressure Vessel Code, Section XI, the full closure test of the MSIVs during COLD SHUTDOWNs will ensure OPERABILITY and provide assurance that the valves maintain the required closing time. The provision for a minimum of 92 days between the tests ensures that full closure testing is not too frequent. The MSIVs are partially stroked quarterly as part of reactor protection system instrument surveillance testing.