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10 CFR 50.12

2130-03-20309
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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: Request for Exemption from 10 CFR 50, Appendix J

In accordance with 10 CFR 50.12, "Specific exemptions," paragraphs (a)(1) and (a)(2), AmerGen Energy Company, LLC (AmerGen), is requesting NRC approval of an exemption for the Primary Containment Leakage Testing requirements in 10 CFR 50, Appendix J, Option B. This exemption would allow the main steam isolation valves (MSIVs) to be Type C tested at a pressure less than P_a. The details of the 10 CFR 50.12 request are attached.

AmerGen requests approval of the subject request by October 15, 2004, based on the scheduled start date for the 1R20 refueling outage.

If you have any questions or require additional information, please contact Mr. Dave Helker at 610-765-5525.

Respectfully,

Michael P. Gallagher

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

Attachment: Request for Exemption

cc: H.J. Miller, Administrator - USNRC Region I
P.S. Tam, USNRC Senior Project Manager, Oyster Creek
R. Summers, USNRC Senior Resident Inspector, Oyster Creek
File No. 02069

A017

Request for Exemption from the Requirements of Paragraph III.B of 10 CFR 50, Appendix J, Option B

I. Specific Exemption Request:

In accordance with 10 CFR 50.12, "Specific exemptions", AmerGen Energy Company requests 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 Main Steam Isolation Valves (MSIVs) at the peak calculated containment pressure related to the design basis accident. An exemption is requested to allow leak testing of the MSIVs (identified below) at reduced pressure.

II. Basis for Exemption Request:

The criteria for granting specific exemptions from 10 CFR 50 regulations are stated in 10 CFR 50.12. In accordance with 10 CFR 50.12(a)(1), the NRC is authorized to grant an exemption upon determining that the exemption is: 1) authorized by law; 2) will not present an undue risk to the public health and safety, and; 3) is consistent with the common defense and security. Furthermore, as stated in 10 CFR 50.12(a)(2), special circumstances must exist for the NRC to consider granting an exemption.

1. The Requested Exemptions and the Activities Which Would Be Allowed Thereunder Are Authorized by Law

   If the criteria established in 10 CFR 50.12(a) are satisfied, and if no other prohibition of law exists to preclude the activities which would be authorized by the requested exemption, the Commission is authorized by law to grant the exemption request. Since, as demonstrated herein, the requested exemption meets the applicable criteria and there is no legal prohibition to its grant, the Commission is authorized by law to grant the exemption.

2. The Requested Exemption will not Present an Undue Risk to the Public Health and Safety

   For the reasons stated in Section 4, Special Circumstances, the proposed local leak rate testing of the MSIVs, in lieu of the specified Appendix J, Option B requirements, does 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.

3. The Requested Exemption is Consistent with the Common Defense and Security

   The common defense and security are not endangered by this exemption request. Further, the potential impact on public health and safety has been determined to be inconsequential.

4. Special Circumstances

Two special circumstances of the type described in 10 CFR 50.12(a)(2) are present in the request under consideration in that: (a) the application of the regulation is not necessary to achieve the underlying purpose of the rule, and; (b) compliance would result in undue hardship.

The purpose of 10 CFR 50, Appendix J is to provide appropriate containment leakage test requirements for nuclear power reactors. 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 alternative measures proposed in the discussion below, in lieu of the applicable Appendix J, Type C test, will meet the underlying purpose of the regulation.

10 CFR 50, Appendix J, Option B, Paragraph III.B requires leak rate testing of the MSIVs at the peak calculated containment pressure related to the design basis accident. An exemption is requested to allow leak testing of the MSIVs at reduced pressure. The MSIVs (globe type) for Oyster Creek are as follows:

Steam Line	Inboard Valve #	Outboard Valve #
North	V-1-7	V-1-9
South	V-1-8	V-1-10

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. A 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.

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, 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).

III. <u>Environmental Assessment:</u>

In accordance with 10 CFR 51.30 and 51.32, the following information is provided in support of an environmental assessment and finding of no significant impact for the proposed change.

An exemption is requested to allow leak testing of the MSIVs at reduced pressure (i.e., ≥ 20 psig). The alternative measures proposed in the discussion above, will provide assurance that the primary reactor containment is an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment. 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. The 11.9 scfh 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).

Based on the above discussion, there is no increase in the probability of higher post accident offsite or onsite doses related to the exemption and therefore no increase in environmental impact beyond that experienced with no exemption.

IV. Conclusion:

As demonstrated above, we consider that this exemption request is in accordance with the criteria of 10 CFR 50.12. Specifically, the requested exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. Also, special circumstances are present as previously described.

V. 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.