DOCKET NUMBER
PROPOSED RULE PR 50
(56 FR 03796)

(2)

# PSEG

DOCKETED
USNRC

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Nuclear Department

'91 APR 15 P4:26

NLR-N91060

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Secretary, US Nuclear Regulatory Commission
Docketing and Service Branch
Washington, DC 20555

Gentlemen:

COMMENTS ON PROPOSED CHANGE TO 10CFR50.55A
INSERVICE TESTING OF CONTAINMENT ISOLATION VALVES
SALEM AND HOPE CREEK GENERATING STATIONS
DOCKET NOS. 50-272, 50-311, AND 50-354

Public Service Electric & Gas Company (PSE&G) hereby submits comments for the proposed rule change to 10CFR 50.55a.

Attachment 1 contains comments on paragraph b(2)vii - Inservice Testing of Containment Isolation Valves and paragraph (6)(ii)(A)(3) Augmented Examination of Reactor Vessel.

Should there be any questions with regard to this submittal, please do not hesitate to contact us.

Sincerely,

R.T. Brown for

B. A. Preston
Manager -
Licensing and Regulation

Attachment

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ATTACHMENT 1

NLR-N91060

# COMMENTS TO 10CFR50.55A PROPOSED RULE CHANGE

Paragraph b(2)(vii) - Inservice Testing of Containment Isolation Valves

The proposed rule, through the adoption of the 1989 Edition of ASME Section XI, adopts ASME/OM-10 for valve testing. This specific paragraph takes exception to certain requirements of OM-10. Specifically, containment isolation valves must be analyzed in accordance with paragraph 4.2.2.3(e) and corrective action taken in accordance with paragraph 4.2.2.3(f) of ASME/OM-10. However, the ASME/OM-10 standard refers all requirements for containment isolation valves to 10CFR50 Appendix J.

Containment isolation valve leak rate testing is required by 10CFR50 Appendix J. These requirements, implemented through technical specifications, require valve leakage to be evaluated against acceptance criteria. Unacceptable leakage rates require valves to be repaired, leakage paths to be isolated, or operating units to be shutdown. In addition, if unacceptable leakage rates are identified during outages, plants are prevented restart.

If the proposed rule is adopted as written, it would require utilities to consider Technical Specification actions, in addition to those identified in the preceding paragraph, which may may force plants to shut down. If they evaluated leakage in accordance with Appendix J, they would not have to.

There have been reports published on excessive leakage through containment isolation valves and the effect on containment leakage integrity. One example is a paper titled, "Primary Containment Leakage Integrity: Availability and Review Experience"; published in Nuclear Safety, Vol. 21-5, dated September-October 1980. This report identifies many of the valve leakage failures to be repetitive in nature with these problems associated with large diameter MSIV's, purge or ventilation valves. As a result of reports such as these, the generic problems with these valves are being addressed and individual valve leakage acceptance criteria for these problem valves have been identified in plant Technical Specifications. The remaining problems with valve leakage are more than likely plant specific and are identified during Appendix J, Type C Testing. The Appendix J program has proven to be effective in identifying containment isolation valve leak rate problems.

It is recommended that any revisions the NRC feels is required to containment isolation valve leak rate testing requirements be made in conjunction with Appendix J. By duplicating valve leak rate acceptance criteria in the plants IST Program, plant operability will be challenged more often without a corresponding increase in safety and it will add confusion in Technical Specification implementation.

Paragraph (6)(ii)(A)(3) - Augmented Inspection of Reactor Vessel

This paragraph allows plants with fewer than 40 months remaining in the Inservice Inspection interval to defer the augmented examination of the Reactor Vessel shell welds to the first period of the next interval. The term augmented examination refers to the examination of 100% of the shell welds, whereas, Section XI requires only a partial examination of these welds. This paragraph also prohibits the use of the deferred augmented examination as a substitute for the reactor vessel shell examination scheduled for implementation during the current inspection interval.

Prohibiting the use of the deferred augmented examinations as a substitute for the reactor shell examinations scheduled within the next 12 months would cause additional expenditure of radiation exposure, time, and money within a short period of time. It would be too late to adequately plan to perform the augmented examinations during the next scheduled outage, even though the reactor vessel core barrel would be removed to complete the examination requirements of the current interval. Since the core barrel must be removed to perform the augmented examinations required by this rule change, PSE&G would thereafter be forced to remove the core barrel again within the subsequent 40 month period.

It has always been recognized within the code that removing the core barrel is a massive undertaking and performed only once in 10 years. It requires removing all fuel from the vessel, lifting the core barrel partly out of the reactor cavity water resulting in high exposure levels on the refueling floor, adds days to the duration of the outage, and significantly increases the cost of the outage.

Based on the above information, we recommend the proposed rule be revised to allow plants within the last 12 months of their current interval be allowed to substitute the deferred augmented reactor vessel shell examinations for the remaining reactor vessel shell examinations scheduled in the current interval.