DOCKET NUMBER PR 50
PROPOSED RULE (56FR03796)

(14)

DOCKETED USNRC

WOLF CREEK
NUCLEAR OPERATING CORPORATION

'91 APR 18 P3:28

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Forrest T. Rhodes
Vice President
Engineering & Technical Services

April 12, 1991

ET 91-0071

U.S. Nuclear Regulatory Commission
Attention: Mr. Samuel J. Chilk,
Secretary
Mail Station P1-137
Washington, D.C. 20555

SUBJECT: Proposed Amendment to 10 CFR 50.55a,
Endorsing ASME Codes

Dear Mr. Chilk:

Please find enclosed the comments from myself and several key members of the ASME Operation and Maintenance (O&M) Main Committee on the proposed amendment to 10 CFR 50.55a as published in the Federal Register, Vol. 56, No. 21, dated January 31, 1991. I am the present Chairman of the ASME O&M Main Committee.

We are concerned that the NRC is taking exception to Part 10, "Inservice esting of Valves in Light-Water Reactor Power Plants" of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, "Operation and Maintenance of Nuclear Power Plants". OM Part 10 was developed as an industry standard using the consensus process with full NRC involvement.

We also recommend that the new 10 CFR 50.55a(f) "Inservice Testing Requirements" be expended to address testing of snubbers. As currently written, 10 CFR 55a(g) "Inservice Inspection Requirements" covers inservice inspection of component supports, but it is not clear that paragraph (f) covers inservice testing of snubbers.

We appreciate the opportunity to be able to comment on this proposed regulation amendment. Additional specific comments are attached. If we can be of further assistance, please contact me.

Sincerely,

Forrest T. Rhodes

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Vice President
Engineering & Technical Services

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COMMENTS ON 10 CFR 50.55a

The NRC is proposing to separate inservice inspection (ISI) and inservice testing (IST), which are currently contained in 10 CFR 50.55a(g), into separate sections, 10 CFR 50.55a(f) and (g). This proposed change will be better for the industry by clarifying which requirements are ISI and which are IST.

There does appear to be a oversight which should be corrected and included as part of the proposed change. Section XI, Subsection IWF makes reference to OM Part 4 Examination and Performance Testing of Nuclear Power Plant Dynamic Restraints (Snubbers). Part 4 contains inservice testing requirements for snubbers and should be referenced in the new 10 CFR 50.55a(f).

The commission has asked for comments relative to the need for revising, or possibly eliminating the proposed modification to Part 10. This proposed modification is not needed, or prudent for the following reasons:

- The ASME Operation & Maintenance Working Group on Pumps and Valves determined that having two sets of testing requirements for containment isolation valves was an element of confusion in the current testing programs. Therefore, when Part 10 (OM-10) was developed from its predecessor IWV, the requirements for testing Category A containment isolation valves, (valves in which leakage is important) were changed to require testing as required by 10 CFR 50, Appendix J. The working group felt that the Appendix J Program, as currently implemented by the industry, is very credible program, which is specific to containment isolation valves. Thus Appendix J Program treats containment leakage as a system, and acceptance criteria is based on a system approach. By imposing the proposed modification, each valve would have a specific leak rate associated with it and corrective action mandated when the leakage is exceeded. This approach removes flexibility from operations of a unit and could force unnecessary shutdowns.

- The leakage limits contained in OM-10 are based on expected leakage of a valve in good operating condition. These limits have not basis when compared to the function of the containment. Appendix J does contain an overall limit, which was specifically developed for leakage for a containment in order to maintain off-site releases below that established by 10 CFR 100. Therefore the Appendix J limit is more appropriate.

- It appears that the reason the NRC is taking exception to Part 10 is a perceived inadequacy in the containment isolation valves testing programs as contained in 10 CFR 50, Appendix J. The Operation & Maintenance Committee feels that the testing program referenced by Part 10, that is, 10 CFR Appendix J, is adequate and no augmented requirements need to be imposed by the NRC. If the Appendix J limit is not appropriate, then this limit should be changed instead of imposing additional Code limits.

- The NRC contends that approximately 30% of the time containment leakage would exceed plant technical specifications. It should be noted that these results are based on the very conservative approach that the lowest leakage rate valve of two valves in series has failed to isolate. This analysis technique is referred to as "Maximum Pathway Leakage", which goes beyond any plant design basis where the plant is designed to protect the health and safety of the public from the consequences of a design basis accident, assuming a single failure. A true measure of the containment structure to limit the release of radioactivity within the limits of 10 CFR 100 would be an analysis of the <u>minimum</u> pathway leakage of each containment penetration, including the most limiting <u>single</u> failure.

- 10 CFR 50 Appendix J, as implemented by most nuclear utilities, and as enforced by the NRC, requires that the summation of all containment penetration leakages, calculated using <u>Maximum</u> Pathway analysis, be less than $0.6L_a$ (where L_a = Allowable Leakage Limit for Containment) <u>prior</u> to declaring primary containment operable and subsequently restarting after each shutdown for refueling (where L_a = Allowable Leakage Limit for Containment). This plant technical specification requirement precludes allowing a containment isolation valve to be declared operable with gross leakage. There exists no basis for applying a specific leakage limit on a valve by valve basis, nor will such a limit appreciably affect the total Appendix J leakage value calculated via maximum pathway leakage techniques. The current Appendix J/Technical Specification limit is based on the 10 CFR 100 analysis for the function of a valve as it relates to containment as a whole. The ASME limit (Part 10 paragraph 4.2.2.3(e)) has no basis for application to total containment leakage.

- The NRC, by this proposed regulatory action, implied that containment isolation valves need to be viewed from both an overall containment leakage and component specific leakage standpoint. It is true that component specific leakage rate values are important. Every utility has to make a determination of when to repair a valve. This determination will consider many factors, including target values, current leakage rate, leakage rate history, combined leakage rate of other valves tested, difficulty of repair, availability of spare parts, outage schedule, system availability for testing, etc. This approach to repair adds flexibility, but still ensures that the Appendix J limit is met.

  Many utilities establish administrative limits, or target values for each valve. One method used is to divide the allowable leakage (.6 L_a) by the total valves times valve size. This gives a limit, based on valve size, which will ensure the Appendix J limit is not exceeded. This approach is a good guide, but depending upon other considerations as described above, exceeding a target value for any particular valve may or may not lead to repair. However, when the target value is exceeded, a penalty must be paid by reducing limits on other valve(s). A running total of the leakage of all valves is maintained to ensure the Appendix J limit is not exceeded.

It does not appear that additional regulations, i.e. in addition to Appendix J, is an appropriate way to regulate a testing program. If changes to Appendix J are in order, the NRC can certainly make them. If a new ANSI standard is appropriate, that can and will be developed by the consensus standard developed programs of which the NRC is a vital and appropriate participant.