July 28, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Dr. Joram Hopenfeld
Engineering Research Applications Branch
Division of Engineering Technology
Office of Nuclear Regulatory Research

SUBJECT: DIFFERING PROFESSIONAL OPINION ON STEAM GENERATOR TUBE INTEGRITY ISSUES

Your July 20, 2000, memorandum informed me that you have requested the Advisory Committee on Reactor Safeguards (ACRS) to function as the DPO ad- hoc panel. The following are my concerns about your request:

1. Since ACRS members can not devote more than 50% of their time to NRC related business they may not be able to pack into their already busy schedule the amount of time that the DPO review requires. Over the past 10 years, both the industry and the NRC have generated numerous documents to promote the practice of leaving defective steam generators in service. These documents, which deal with highly complex technical subjects, contain inaccurate, inconsistent and misleading information, as well as important assumptions which are poorly stated. Additionally, certain subjects have implied assumptions which are not stated. I estimated that it would take about two months for each panel member to properly and adequately review the necessary documents to make a valid technical determination of the facts.

2. The ACRS has participated in the agency positions which are at issue. In 1994 (Attachment 1), the ACRS agreed to the implementation of GL-95-05 “Voltage Based Repair Criteria For Westinghouse Steam Generator Tubes” and disagreed with the DPO claim that the GL represents a serious safety risk to the public. In 1997, in a disagreement with the DPO position, the ACRS concluded that GL-98-XX “Steam Generator Tube Integrity” may be released for public comments (Attachment 2).

The DPO position has always been that the Voltage Based Criteria as prescribed in GL-95-05 should not be accepted by the NRC as a substitute for the 40% plugging rule. The ACRS has previously rejected the position of the DPO. There were adequate technical information and uncertainties at the time of the ACRS’s decision concerning voltage based repair criteria to have rejected that, but the ACRS did not do so.

It should be noted, however, that ACRS approval of GL 95-05 was conditioned on that GL-95-05 was an interim measure and the understanding that the tube support plate provided structural constraints. Both of these stipulations have been shown to be incorrect. The ACRS also expressed reservations regarding how radiological releases were calculated, the lack of adequate data base for leakage calculations, and the lack of adequate techniques for detection and characterization of degradation. There has been no significant progress in these fields since GL-95-05 was released.

cc: Chairman Meserve
John Larkins

Accession Number: ML003735901

ATTACHMENT 1

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D940912

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission Washington, D.C.

Dear Chairman Selin:

SUBJECT: PROPOSED GENERIC LETTER 94-XX, 'VOLTAGE-BASED REPAIR CRITERIA

During the 412th meeting of the Advisory Committee on Reactor Safeguards, August 4-5, 1994, we reviewed the subject generic letter (GL), an associated differing professional opinion (DPO),and a draft of an Advance Notice of Proposed Rulemaking on Steam Generator Tube Integrity. During the 413th meeting, September 8-10, 1994, we discussed the NRC staff's revised calculations for radiological consequences of a main steamline break associated with a degraded steam generator. During our review, we had the benefit of discussions with representatives of the NRC staff and theNuclear Energy Institute (NEI), as well as the author of the DPO.We also had the benefit of the documents referenced. In part, this report is in response to a request made by the Executive Director for Operations in a July 15, 1994, memorandum to the Executive Director of the Advisory

Although existing mechanics-based design criteria and evaluation methods have served to ensure adequate steam generator tube integrity, they appear to be overly conservative for some types of degradation, and result in unnecessary tube plugging or repair. The proposed GL provides an alternate approach applicable solely to axially oriented outside diameter stress corrosion cracking (ODSCC) of tubes at the tube-support-plate intersections in Westinghouse steam generators with drilled-hole support plates.

We support the issuance of the proposed GL for public comment. We have reviewed the DPO and do not believe that it identifies any fundamental shortcomings in the approach proposed in the GL.

The.DPO cites a high core damage frequency (CDF) of 3.4 x 10-4/RY. This value was based on a preliminary scoping analysis performed by the office of Nuclear Regulatory Research (RES). Subsequent analyses performed by RES in support of the application of the interim plugging criteria for the Trojan Nuclear Plant and for NUREG-1477 give CDFs of less than 2 x 10-6/RY. These values are based on conservative estimates of leakage from degraded tubes. Except perhaps for steamline breaks, the structural restraint provided by the tube-support plate provides a high degree of assurance against tube bursts.

The criticism in the DPO of the approach used in the proposed GL and in the Standard Review Plan to compute radiological releases during a main steamline break appears to warrant further consideration. The basis for the definition of the iodine spike during a rapid depressurization transient as 500 times the equilibrium release rate is not clear. However, an alternate calculation of the release based on the gap inventory of iodine in leaking fuel elements appears to give comparable releases. In both approaches there appears to be margin in meeting the 10 CFR Part 100 limits. The staff should review the spiking data or consider other approaches to estimate the iodine release to provide a more

satisfactory basis for the radiological dose estimates. In particular, we encourage the staff to quantify the level of conservatism in its analyses.

While the proposed GL appears to provide a useful interim approach for assessing steam generator tube integrity, the database for the present empirical correlations for burst pressure and leakage with the bobbin

The use of such empirical correlations as the basis for assuring the integrity of steam generator tubing would also seem to require an ongoing tube-pull program with associated burst and leak testing and metallurgical examinations as outlined in the proposed GL to ensure that the correlations remain valid as degradation continues. In the longer term, it would be worthwhile to reconsider a fracture-mechanics-based approach utilizing improved non- destructive examination techniques that provide more accurate detection and characterization of degradation. Ongoing efforts in RES and in industry to develop and implement such an

We agree with the staff position that rulemaking is the preferred regulatory approach to the problem of steam generator tube degradation, although we are skeptical that a new rule can be developed as expeditiously as the proposed schedule suggests. The overall objective and attributes of the new rule, as described by the staff, pay proper obeisance to performance-based regulation. We would like to be kept informed of the progress by the staff in the implementation of a performance-based approach.

Sincerely

T. S. Kress

References:

1. Memorandum dated July 8, 1994, from F. J. Miraglia, Deputy Director, Office of Nuclear Reactor Regulation, for E. L. Jordan, Chairman, Committee to Review Generic Requirements, Subject: CRGR Review of Generic Letter 94-XX, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes"
2. Memorandum dated July 15, 1994, from J. M. Taylor, NRC Executive Director for Operations, for J. T. Larkins, ACRS Executive Director, Subject: ACRS Review of Proposed Generic Letter 94-XX, Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes
3. U.S. Nuclear Regulatory Commission, 10 CFR Part 50, RIN 3150-; Steam Generator Tube Integrity (7590-01), Draft Advance Notice of Proposed Rulemaking, received July 20, 1994

Memorandum dated August 17, 1994, from J. A. Calvo, NRC Office of Nuclear Reactor Regulation, for J. T. Larkins, ACRS Executive Director, Subject: Revisions to Slides Used by Staff During August 3, 1994, Subcommittee Briefing on Steam Generator Alternate Repair Criteria

U. S. Nuclear Regulatory Commission, NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes," Draft Report for Comment, June 1993

~ Memorandum dated January 15, 1993, from E. S. Beckjord, Director, Office of Nuclear Regulatory Research, to T. E.* Murley, Director, Office of Nuclear Reactor Regulation, Subject: interim Plugging Criteria for Trojan Nuclear Plant

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ATTACHMENT_ -Z

UNITED STATES
NUCLEAR REGULATORY COMMISSION

November 4, 1997

MEMORANDUM Jocelyn A. Mitchell
Technical Assistant
Office of the Executive
Director for Operations

FROM: Joram Hopenfeld
Generic Safety Issues Branch
Division of Engineering Technology
Office of Nuclear Regulatory Research

SUBJECT: DPO REGARDING REPAIR CRITERIA FOR STEAM GENERATOR TUBES- ISSUE RESOLUTION DOCUMENT AND RESOLUTION

REFERENCE Memo dated Octcber 1, 1997, Mitchel subject to Hopenfeld, same

This is in reply to the referenced memorandum. I am preoarlIna comments on the staff efforts to resolve my DPO for the proposed Generic Letter Please provide me a date by when I must submit my comments for the announcement in the Federal Register.

The October ACRS letter to the EDO pointed out that the staff made a Serious, effort to resolve the DPO. The letter merely restated the respective positions of the DPO and the staff without discussing the technical merits of the opposing views. It should be noted that the ACRS letter, incorrectly cited the DPO position on leakage and also incorrectly represented the NUREG-1477 discussion regarding field experience with leakage.

The ACRS concluded that the GL may be released for public comments, thereby directly participated in the formulation of the agency position that is at issue. According to DPO guidelines, this would disqualify the ACRS from participation in the technical review of the DPO. For this reason I do not wish that the ACRS serve as a substitute to the ad hock panel.

Following are three names for the panel:

(1) Dr. George Marino (NRC)
(2) Dr. Rick Kurtz (PNL)

cc: John Larkins