UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

July 8, 1980

MEMORANDUM FOR: R. M. Gamble, Section Leader
Component Integrity Section
Materials Engineering Branch

FROM: J. Halapatz
Materials Engineering Branch

SUBJECT: DIFFERING PROFESSIONAL OPINION CONCERNING REPAIR OF SEQUOYAH UNIT ONE PRESSURIZER RELIEF PIPE - INTERIM COMMENTS

This memorandum submits written comments for the record concerning views expressed by his immediate supervisor and others in the chain-of-command in the course of resolution of a differing professional opinion. The differing professional opinion was expressed in the memorandum, Halapatz to Pawlicki, dated June 16, 1980, subject, "Expression of Differing Professional Opinion in the Matter of the Adequacy of Sequoyah Unit One Weld Drawbead Repair of Pressurizer Relief Pipe." These comments are submitted in accordance with SECY-80-164, Enclosure 1, Section B.8.b., dated March 25, 1980.

The minority submits comments to the following memoranda:

1. "Differing Professional Opinion - Sequoyah Pressurizer Pipe Weld," Gamble to Pawlicki, dated June 27, 1980.

2. "Comments on the Differing Professional Opinion on Sequoyah Weld Repair of Pressurizer Relief Pipe," Pawlicki to Noonan, dated June 27, 1980,

3. "Comments on the Differing Professional Opinion on Sequoyah Weld Repair of Pressurizer Relief Pipe (J. Halapatz)," Noonan to Vollmer, dated June 30, 1980.

<u>The Gamble to Pawlicki Memorandum</u>

1. The purpose of the radiography was to determine if the pressure boundary was penetrated. This determination is necessary to qualify exemption from post weld repair system hydrostatic testing required by the ASME Code.

   The Code (IWA-4130(b)) commits the owner to evaluate the suitability of the welding procedure to be used to make the repairs. The Code (IWA-4130(c)) states that repair programs shall be subject to review by the enforcement and regulatory authorities having jurisdiction at the plant site. Presumably the mockup of interest was made to satisfy these requirements. If this was not the case, then no basis as required by the code existed before the fact to implement the weld repair.

   The radiography, which was performed, was intended to determine, after the fact, whether or not the pressure boundary was penetrated. Radiography is

essentially meaningless for the characterization of degree of sensitization.

The determination of whether or not the pressure boundary was penetrated is an important technicality, which must be addressed within the context of 10 CFR 50 Appendix B. Third party inspection, which was mandated by the NRC for Sequoyah by the Rubenstein to Parris memorandum, dated September 21, 1979, subject, "Qualification of Inspectors, Inspection Specialists, and Inspection Agencies for Sequoyah," provides for an expedient resolution of this issue.

An authorized inspector affiliated with TVA's authorized inspection agency should view the radiographs of interest and make the official interpretation.

2. A significant disparity is noted between the safety significance involved as regarded by the licensee and by the majority opinion as expressed in the Gamble to Pawlicki memorandum. The licensee's safety implication, as stated in nonconformance report NCR SWP-79-S-8, states that failure of the piping of interest could lead to an uncontrolled blowdown of the reactor coolant system. This view is in contrast to the majority view that if cracking occurs, operating experience indicates that IGSCC has been detected by either inspection or leak detection before excessive leakage results. Perhaps the difference lies in what is likely to be leaking, steam or water.

3. With regard to the statement that no cracking incidents have been experienced in PWR pressurizer lines, the minority recalls that not too long ago Ginna had some sort of a problem in this area. Perhaps the majority could provide some enlightenment in the matter.

4. In equating the potential service experience of the repair welds of interest to that of the population of full penetration welds in LWRs, the majority's attention is called to cracks which have occurred to the astonishment and consernation of experts. The minority cites, in reference, "Operating Experience Memorandum No. 17, Pipe Cracks in Borated Water Systems of TMI-1," Eisenhut to Schroeder, October 5, 1979. What is particularly disturbing in this case is that the environmental conditions experienced by the material apparently have been accepted generally as non-aggressive. A major difference in philosophies is obvious in this issue. The minority's view put simply, is that the public would be better served by addressing potential worst case materials problems during construction rather than determining the causes of material failures occurring in service. The minority finds little solace in the association by the majority of small leakage rates with inservice IGSCC and small leak detectability with the maintenance of system safety margins. The minority view seeks to prevent cracks; the majority view seeks to provide accommodation for cracks when they occur. Undoubtedly the public interest would be best served by a meld of the two views.

The minority finds the in situ metallography performed inconclusive. The minority sees, even in the Xerox copies of photomicrographs available to him, evidence of severely smeared metal and has serious doubts that the true

microstructure was developed. The minority is at a loss to explain how NRR could meaningfully interpret the in situ metallography as acceptable.

Within this context, the minority finds that laboratory examination and corrosion testing of actual piping material is much more convincing that the most inspired pontifications as to why a material shouldn't crack. In this light, if not boat sample material, then mocked-up archive material, whose identity should be reassured by 10 CFR 50 Appendix B requirements, should be examined and corrosion tested.

The Pawlicki to Noonan Memorandum

1. The majority statement that the primary purpose of the mockup used by TVA was to experiment with the effectiveness of the straightening process itself, and very little can be deduced about the condition of the pressurizer relief pipe from examination of the mockup is contradictory. The majority's acceptance of the repair via the memorandum, Pawlicki to Rubenstein, dated December 4, 1979, subject, "Tennessee Valley Authority, Sequoyah Nuclear Plant, Unit No. 1" was premature in that it preceded the majority consultant Gustafson's January 16, 1980 meeting with TVA, which was addressed in the memorandum, Gustafson to Pawlicki, dated January 25, 1980, subject, "Trip Report of Visit to Tennessee Valley Authority Sequoyah Nuclear Plant, Unit-1." In his statement of the problem, Gustafson made reference to the matter that TVA's request to NRC for approval did not provide background details on the repair procedure employed, details concerning weld procedure qualification, non-destructive examination, information on the susceptibility of repaired area to intergranular stress corrosion cracking or any consideration of residual welding stresses. Gustafson further stated that approval was held up until additional information could be obtained and evaluated.

   The minority notes notes that Gustafson did make significant deductions based on the mockup. The minority calls attention to the memorandum, Pawlicki to Rubenstein, dated February 28, 1980, Subject, "Tennessee Valley Authority, Sequoyah Nuclear Plant, Unit No. 1 Realignment of Pressurizer Relief Pipe," which recommended that the minority meet with TVA and examine the metallographic samples in question. It is to be noted that only available metallographic samples in question at that time were those parted from the mockup. The memorandum, Pawlicki to Rubenstein, dated March 3, 1980, subject, "Metallurgical Examination of TVA Sequoyah Nuclear Plant, Unit No. 1 - Weld Drawbead Realignment of Pressurizer Relief Pipe Mockup," specifically identified that the purpose of the minority visit with TVA was to perform a metallurgical examination of the mockup used in qualification of the weld drawbead realignment of the Sequoyah pressurizer relief pipe. At this point, there is little doubt of the majority's acceptance of the mockup to qualify the repair welding.

   The minority's metallographic examination of the mockup, however, disclosed, in part, that the weld was fully penetrated, thereby disqualifying TVA for exemption from system hydrostatic testing.

In the ensuing controversy, which developed as a result of the contradictory findings, the compromise was reached that in situ examination and reradiography would be performed to (1) determine whether or not the pressure boundary had been penetrated and (2) determine the degree of sensitization in the production repair. It is the opinion of the minority that the determinations that have been made are inconclusive, as addressed in the memorandum, Vollmer to Denton, dated July 2, 1980, subject, "Differing Professional Opinion Concerning Repair of Pressurizer Relief Pipe at Sequoyah Unit One."

2. The majority votes against taking boat samples. The minority notes that 18 pieces of tubing of the same size and heat (C7492) as the production repair were procured, as shown in TVA certified material test documentation. An obvious alternative then, would be to mockup some of this material using the production repair design and weld procedure. Laboratory metallographic examination and corrosion testing (in .2 ppm oxygen-bearing steam) of ID and transwall samples parted from the mockup should provide definitive test data on which the NRC evaluation of the weld repair should be based. Put simply, it is the minority view that engineering judgments, when possible, should be based on test data rather than opinion.

3. The majority votes that third party inspection be performed since it may help remove remaining minority doubts. Minority doubts notwithstanding, attention is again called to the memorandum, Rubenstein to Parris, dated September 21, 1979, subject, "Qualification of Inspectors, Inspection Specialists, and Inspection Agencies for Sequoyah," which directs that TVA institute third party inspection for Sequoyah.

The Noonan to Vollmer Memorandum

1. Given TVA's statement of the safety implication involved, viz., that failure of the repaired piping could lead to an uncontrolled blowdown of the reactor coolant system, augmented ISI would appear not to provide the necessary assurance of safe shutdown. Augmented ISI would address failure in terms of slow leaking cracks, a circumstance which TVA apparently did not have in mind when nonconformance report NCR AWP-79-S-8 was written.

2. With respect to the majority opinion that the weld drawbead repair of the type used at Sequoyah is a common and accepted method of repair, the minority offers the purely hearsay information that TVA considered the procedure unique to the extent that a technical paper describing it was under consideration.

Joseph Halapatz
Materials Engineering Branch
Division of Engineering

For cc's, see attached sheet

cc: H. R. Denton
D. G. Eisenhut
✔R. H. "ollmer
V. S. Noonan
R. L. Tedesco
S. S. Pawlicki
A. Schwencer
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 15, 1980

The Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: REPORT ON THE SEQUOYAH NUCLEAR POWER PLANT, UNITS 1 & 2

Dear Dr. Ahearne:

During its 243rd meeting, July 10-12, 1980, the Advisory Committee on Reactor Safeguards completed its review of the application of the Tennessee Valley Authority (hereinafter referred to as the Applicant) for authorization to operate the Sequoyah Nuclear Plant, Units 1 & 2 at full power. The Committee had considered aspects of the application during its 242nd meeting, June 5-7, 1980; 236th meeting, December 6-8, 1979; 229th meeting, May 10-12, 1979; and 228th meeting, April 5-7, 1979. A tour of the facility was made by members of the Subcommittee on January 24, 1976 and the application was considered at Subcommittee meetings on July 9, 1980; June 2, 1980; November 5, 1979; and March 12, 1979. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicant, the Westinghouse Electric Corporation, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed. The Committee reported on interim low power operation of Unit 1 on December 11, 1979 and on a construction permit for this plant on February 11, 1970.

In its letter of December 11, 1979 the Committee addressed the proposed special low power test program, to be carried out on Unit 1, the seismic reevaluation of the Sequoyah plant, actions on recommendations resulting from the review of the accident at the Three Mile Island Station, Unit 2, and actions on various generic problems. These generic problems were further discussed in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 7," dated March 21, 1979. The Committee's recommendations in its December 11, 1979 letter are also applicable to Unit 2 except that the special low power test program will not be repeated on Unit 2.

The special low power test program has been reviewed by Westinghouse Electric Corporation and by the NRC Staff. The Applicant began these tests on July 11, 1980 and the Applicant, Westinghouse, and the NRC Staff will review the results of these tests. It is expected that the additional operator training and operator experience will prove to be beneficial.

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The Committee has reviewed and reported on NUREG-0660, "NRC Action Plans Developed as a Result of the TMI-2 Accident," Draft 3. The status of the Applicant's compliance with the NTOL licensing requirements as well as a number of non-TMI-related items were reviewed during its 243rd meeting. There are a number of both non-TMI and TMI-related requirements not fully resolved. Both the NRC Staff and the Applicant expect that the complete resolution of these outstanding items is essentially a procedural or documentary matter which will be completed within a very few weeks. These items should be resolved to the satisfaction of the NRC Staff. The Committee wishes to be kept informed. The Committee believes that the implementation of the Action Plan as it will be realized at Sequoyah is adequate to assure the safe operation of this plant.

The Committee, in its March 11, 1980 report on the NTOL items, recommended that the licensees develop reliability assessments for their plants and that design studies of possible hydrogen control and filtered vented containment systems be required. The Applicant has conducted studies of a number of means for hydrogen control, and as an interim measure, has proposed installation of a distributed array of ignition sources which it expects to have in place by the fall of 1980. The Applicant has concluded that by this means the containment would be able to cope with the pressure resulting from the combustion of hydrogen released by the reaction with water of up to about 70% of the zirconium in the core. This compares with the 25% which the containment could cope with without any additional control measures and the 30 to 50% estimated to have reacted in the accident at TMI. The NRC Staff plans to review the proposed system in detail to assure itself of its efficacy and that all safety aspects have been taken into account. The Committee wishes to be kept informed of the further conclusions reached by the Staff and the Applicant in their continuing consideration of these matters. The Applicant has conducted reliability assessments of some features of the plant and has considered some aspects of the effects of a possible filtered vented containment. Though the work accomplished to date is limited in scope, these studies are definitely responsive to the Committee's recommendations on these points. The Applicant proposes to continue studies of this nature and to extend the range of their application. While these efforts, as well as those concerned with hydrogen control, should be vigorously pursued, in view of the commitments made by the Applicant, it is the opinion of the Committee that their present incomplete status need not delay the issuance of a full power operating license.

Early this year a differing professional opinion was advanced by a member of the NRC Staff concerning the acceptability of a particular weld repair in the piping to a pressurizer relief valve of Sequoyah Unit No. 1. All other qualified and responsible members of the NRC Staff, as well as professional personnel on the staff of the Applicant, take the position that the weld should be regarded as acceptable since there is no evident reason why it should not be at least as capable as other (more standard) welds which would

be considered acceptable. The differing opinion is not that the weld is demonstrably less capable than it need be, but 1) that the evidence available is inconclusive on this point, and 2) that more specifically relevant information could be obtained without serious difficulty. This could be done by constructing a mock-up of the weld in question using material and procedures as similar as possible to those which apply in the actual case and subjecting the mock-up to a through-wall metallographic examination. The results of this examination could then (for example) be compared with those from a full penetration weld in the same material, which has been performed in the standard fashion and deemed acceptable based on satisfactory operational experience with which the majority opinion has compared the present weld. This has not been done. The Committee does not consider it to be particularly likely that this weld repair presents a serious hazard; but it does believe the evidence on this point could be improved. The Committee believes that, in the interest of resolving the question that has been raised to the maximum extent readily possible, steps of the nature outlined should be taken.

The Committee believes, that if due consideration is given to the items mentioned above, the Sequoyah Nuclear Plant, Units 1 and 2 can be operated at levels up to full power without undue risk to the health and safety of the public.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References:

1. Tennessee Valley Authority, "Final Safety Analysis Report, Sequoyah Nuclear Power Plant," Volumes 1-13, and Amendments 1-63.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Sequoyah Nuclear Plant Units 1 and 2," NUREG-0011, March 1979.
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Sequoyah Nuclear Plant Units 1 and 2," Supplement No. 1, NUREG-0011, February 1980.
4. U.S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," NUREG-0660, May 1980.
5. U.S. Nuclear Regulatory Commission, "TMI-Related Requirements for New Operating Licenses," NUREG-0694, June 1980.