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

JUL 29 1983

MEMORANDUM FOR: Victor Stello, Jr., Deputy Executive Director for Regional Operations and Generic Requirements

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: CRGR REVIEW OF PROPOSED EXEMPTIONS TO THE REGULATIONS IN THE IMPLEMENTATION OF MULTIPLANT ACTION D-10 (USI A-2) FOR PWRS EVALUATED BY WESTINGHOUSE TOPICAL REPORTS WCAP 9558, REV. 2 AND WCAP 9787

The staff has concluded its technical review of the Westinghouse topical reports, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack," WCAP 9558, Rev. 2 and, "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation," WCAP 9787. Attached as Enclosure 2 is the staff's topical report evaluation which addresses the application of mechanistic fracture mechanics to resolve the issue of asymmetric blowdown loads on PWR primary systems. The generic issue is described in detail in NUREG-0609 (resolution of generic task action plan A-2). The staff's topical report evaluation concludes that large margins against unstable crack extension exist for certain stainless steel PWR primary coolant piping postulated to have large flaws and subjected to the safe shutdown earthquake in combination with the loads associated with normal plant conditions. However, the topical reports, and therefore the staff's evaluation, are limited to operating units of eleven licensees, belonging to the A-2 Owners Group, that engaged Westinghouse to develop and submit to the staff WCAP-9558, Rev. 2, and WCAP-9787 on their behalf. The operating units of these eleven licensees are identified in Table 1 of Enclosure 2.

Sections 1.0 and 2.0 of Enclosure 2 provide an overview of the issues that led to the establishment of USI A-2, a brief chronology of its generic resolution and the scope and a summary of the staff's review of the Westinghouse topical reports. The ACRS Subcommittee on Metal Components comments and questions regarding these reports have been resolved and they are in general agreement with the staff's proposed technical resolution of this issue. The evaluation contained in Enclosure 2 constitutes justification for the identified licensees to be exempted from General Design Criterion 4 in the context of the dynamic loads associated with the definition of a LOCA as including a break equivalent in size to the double-ended rupture of the largest

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pipe in the reactor coolant system (Appendix A to 10 CFR Part 50). Explicitly, the scope of the exemption is only applicable to the measures required for protection of reactor coolant system core, components and supports against the dynamic effects (i.e., the loads resulting from asymmetric blowdown) of postulated main coolant loop double-ended pipe ruptures; it does not pertain to the definition of a LOCA nor its relationship to the regulations addressing design requirements for ECCS (10 CFR 50.46), containment (GDC 16, 50), other engineered safety features and the conditions for environmental qualification of equipment (10 CFR 50.49). The enclosed program package is being submitted to the CRGR for consideration of this proposed exemption to the regulations relative to the implementation of generic issue A-2 for the eleven licensees identified.

The information contained in this package for the consideration of the CRGR responds to the information requirements identified in NRR Office Letter No. 39 Revision 1, dated December 15, 1982. Part of this information is included in Enclosure 3 to this memorandum which provides background information in a question and answer format to support the CRGR review.

The value-impact analysis supporting the proposed exemption is presented in Enclosure 4 to this memorandum and follows the recommended regulatory analysis guidelines in NUREG/BR-0058. This analysis indicates insignificant effect on public risk with substantial savings in cost and occupational radiation exposure associated with the proposed exemption.

The staff is proposing an implementation plan and schedule consisting of the following elements. Letters to Westinghouse and each of the eleven licensees will inform them of the results of the staff's topical report evaluation and transmit a copy of that evaluation. As specified in these letters and in the topical report evaluation, justification for an exemption to the pertinent regulations is contingent upon satisfying the staff's leak detection criteria and, in addition, for two licensees under the Systematic Evaluation Program, the results of their seismic reanalyses. Exemption requests will be treated as routine licensing actions. This is procedurally the most practical and expeditious manner of implementation. Enclosure 1 is a draft of the letters discussed above.

The proposed exemption to the regulations is classified as a CRGR Category 2 action, i.e., one that does not meet the criteria for designation as Emergency or Category 1. We request that the CRGR complete its review and schedule a meeting on August 24 or 31, 1983. For further information on this review package, contact B. D. Liaw, Chief, Materials Engineering Branch (X-27258) or K. R. Wichman, Materials Engineering Branch (X-24679).

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Draft letters to Licensees and to Westinghouse
2. Topical Report Evaluation
3. Background Information for CRGR Review
4. Regulatory Analysis for Mechanistic Fracture Mechanics Evaluation of Reactor Coolant Piping

ENCLOSURE 1

TO OPERATING PWR LICENSEES ON THE ENCLOSED LIST

SUBJECT: SAFETY EVALUATION OF WESTINGHOUSE TOPICAL REPORTS DEALING WITH ASYMMETRIC BLOWDOWN LOADS ON PWR PRIMARY SYSTEMS

References 1. WCAP 9558, Revision 2 (May 1982) "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack"

2. WCAP 9787 (May 1982) "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation"

3. Letter Report NS-EPR-2519, E. P. Rahe to D. G. Eisenhut (November 10, 1981) Westinghouse Response to Questions and Comments Raised by Members of ACRS Subcommittee on Metal Components During the Westinghouse Presentation on September 25, 1981.

The NRC staff has completed its review of the above referenced Westinghouse topical reports and letter report related to Asymmetric Blowdown Loads on PWR Primary Systems. Enclosed is a letter informing Westinghouse of the results of our review with the staff's topical report evaluation attached. Our evaluation concludes that the dynamic loads resulting from main coolant loop double-ended pipe breaks need not be considered as a design basis for the A-2 Westinghouse Owner's Group plants to satisfy Generic Issue A-2 as specified in NUREG-0609, provided the following two conditions as specified in Section 5.0 of the topical report evaluation are met:

1. Reactor primary coolant main loop piping at Haddam Neck and Yankee Atomic Power Station are acceptable provided the results of seismic analyses confirm that the maximum bending moments do not exceed 42,000 in-kips for the highest stressed vessel nozzle/pipe junction.

2. Leakage detection systems at your facility should be sufficient to provide adequate margin to detect the leakage from the postulated circumferential throughwall flaw utilizing the guidance of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," with the exception that the seismic qualification of the airborne particulate radiation monitor is not necessary.

With these conditions satisfied, the staff considers that an acceptable basis has been provided for the named licensees to demonstrate conformance with A-2 and to justify an exemption to General Design Criterion 4 in terms of the dynamic loads associated with the definition of a LOCA as a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system (Appendix A to 10 CFR Part 50).

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
Letter to Mr. Pres Rahe,
Westinghouse Electric Corp.

ENCLOSURE 1

Mr. Pres Rahe
Manager Nuclear Safety
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Rahe:

We have completed our evaluation of the following reports:

- o WCAP 9558, Revision 2 (May 1982) "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack"

- o WCAP 9787 (May 1982) "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation"

- o Letter Report NS-EPR-2519 E. P. Rahe to D. G. Eisenhut (November 10, 1981) Westinghouse Response to Questions and Comments Raised by Members of ACRS Subcommittee on Metal Components During the Westinghouse Presentation on September 25, 1981.

Based on our evaluation of these reports, the staff has concluded that the dynamic loads associated with double-ended pipe breaks in the primary system main loops need not be considered as a design basis to satisfy

Generic Issue A-2, "Asymmetric Blowdown Loads on PWR Primary Systems." Currently, this conclusion is only applicable to those operating facilities identified in Table 1 of the enclosed staff Topical Evaluation Report. In using these topicals, the identified licensees need to satisfy the following two conditions as specified in Section 5.0 of the Enclosure:

1. Reactor primary coolant main loop piping at Haddam Neck and Yankee Atomic Power Station are acceptable provided the results of seismic analyses confirm that the maximum bending moments do not exceed 42,000 in-kips for the highest stressed vessel nozzle/pipe junction.

2. Leakage detection systems at the specified facilities should be sufficient to provide adequate margin to detect the leakage from the postulated circumferential throughwall flaw utilizing the guidance of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," with the exception that the seismic qualification of the airborne particulate radiation monitor is not necessary.

With these conditions met, we have concluded that the dynamic loads associated with double-ended breaks of main loop piping in the reactor coolant system need not be considered as a design basis for resolution of Generic Issue A-2.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
Topical Evaluation Report