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January 3, 1991

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U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Reference: Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73

Steam Generator Tube Rupture Analysis (TAC 62929)

Gentlemen:

SER supplement 5 briefly describes the Staff's desire to review a plant-specific submittal for BVPS-2 which addresses the results of a generic Staff review of WCAP-10698. Duquesne Light Company has reviewed the Staff's evaluation (dated March 30, 1987) of WCAP-10698. The following information is provided to satisfy the concerns specified in your generic SER:

Staff SER

Each utility in the SGTR subgroup must confirm that they have in place simulators and training programs which provide the required assurance that the necessary actions and times can be taken consistent with those assumed for the WCAP-10698 design basis analysis. Demonstration runs should be performed to show that the accident can be mitigated within a period of time compatible with overfill prevention, using design basis assumptions regarding available equipment, and to demonstrate that the operator action times assumed in the analysis are realistic.

DLC Response

As input to a plant-specific analysis which used the methodology described in WCAP-10698, DLC conducted a series of simulator exercises to measure operator action times. The accident scenario was designed to mock the events to be postulated in the plant specific analysis (i.e. rupture of a single tube at 100% power with coincident loss of offsite power and worst single failure). All operating crews were confronted with this scenario. None of the crews were made aware of their participation in the study or that a scenario of this type was likely to occur. Due to a simulator malfunction, only one crew's data could not be obtained.

The operator response times for each crew were compared and the longest time for each operator action was chosen as a conservative input to the analysis. These action times are presented in Table II.1 of WCAP-12737 (Attachment 1).

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STAFF SER

Perform a site specific SGTR radiation offsite consequence analysis which assumes the most severe failure identified in WCAP-10698, Supplement 1. The analysis should be performed using the methodology in SRP Section 15.6.3, as supplemented by the guidance in Reference (1).

DLC Response

An analysis of offsite radiation consequences which assumes the limiting single failure of WCAP-10698, Supplement 1, has been performed. Results and methodology are provided in WCAP-12737 (Attachment 1).

Staff SER

Perform an evaluation of the structural adequacy of the main steam lines and associated supports under water-filled conditions as a result of SGTR overfill.

DLC Response

Consistent with the recommendations of the Westinghouse Owners Group (WCAP 11002), an evaluation for BVPS-2 has been made to determine the effect of water in the main steam lines at 560° F. The analysis of the piping was performed to determine the combined effect of the deadweight of water in the lines and the thermal expansion due to the temperature of the fluid. The main steam line conditions requiring analysis of deadweight with water and thermal expansion are normally exclusive of each other, one for hydrotest and the other for steam temperature at normal operation. In the case of hydrotest with water-filled lines many of the spring hangers are pinned to avoid "bottoming-out", a condition not normal under high temperature conditions.

The analysis confirmed that the stress in the piping was not excessive and the associated supports were not overloaded.

Staff SER

Provide a list of systems, components and instrumentation which are credited for accident mitigation in the plant specific SGTR EOP(s). Specify whether each system and component specified is safety grade. For primary and secondary PORVs and control valves specify the valve motive power and state whether the motive power and valve controls are safety grade. For non-safety grade systems and components state whether safety grade backups are available which can be expected to function or provide the desired information within a time period compatible with prevention of SGTR overfill or justify that non-safety grade components can be utilized for the design basis event. Provide a

Staff SER (continued)

List of all radiation monitors that could be utilized for identification of the accident and the ruptured steam generator and specify the quality and reliability of this instrumentation if possible. If the EOPs specify steam generator sampling as a means of ruptured SG identification, provide the expected time period for obtaining the sample results and discuss the effect on the duration of the accident.

DLC Response

Attachment 2 provides a list of equipment found in the EOPs which is sufficient to accomplish the major recovery actions which are specific to mitigation of a tube rupture accident. These actions are:

1. Identification and isolation of the ruptured steam generator.
2. Cooldown to establish subcooling margin.
3. Depressurization of the RCS.
4. Termination of SI to stop primary to secondary leakage.

Equipment found in the EOPs which is used for actions which are not of specific interest in a tube rupture accident (such as for determining whether to stop quench spray, if activated) has not been included. This list is also limited to the worst case scenarios described in Attachment 1.

Similarly, Attachment 3 provides a list of instrumentation and radiation monitoring described in the EOPs. Section 7.5 of the BVPS-2 UFSAR describes the classification and method of classifying this equipment. This information was formerly reviewed and accepted by the NRC.

The EOPs do not rely on steam generator sampling as a means of identifying the faulted generator, but steamline surveys may be used as an alternative or supplement to installed monitoring equipment. Since the primary means of identification is the installed monitors, survey times have not been obtained.

Staff SER

Perform a survey of plant primary and "balance-of-plant" systems design to determine the compatibility with the bounding plant analysis in WCAP-10698. Major design differences should be noted. The worst single failure should be identified if different from the WCAP-10698 analysis and the effect of the difference on the margin of overfill should be provided.

DLC Response

DLC has not attempted to show that BVPS-2 is bounded by the plant configuration used in the generic WCAP-10698 analysis. Instead, BVPS-2 was specifically analyzed using the approved methodology for the "bounding plant". Design differences which impacted the analysis were limited to such considerations as flow values and valve capacities rather than major hardware differences such as the number of reactor coolant loops. WCAP-12737 (Attachment 1) discusses the inputs used for the plant-specific analysis.

Please contact my staff if further information is needed for preparation of your SER.

Sincerely,

J. D. Sieber,
Vice President
Nuclear Group

GLB/sao

Attachment

cc: Central File (2)
Mr. J. Beall, Sr. Resident Inspector
Mr. T. T. Martin, NRC Region I Administrator
Mr. A. W. DeAgazio, Project Manager
Mr. R. Saunders (VEPCO)

Attachment 1

LOFTR2 Analysis for a Steam Generator Tube Rupture for BVPS-2

Enclosed are:

1. 1 copy of WCAP-12737, "LOFTR2 Analysis for a Steam Generator Tube Rupture for Beaver Valley Power Station, Unit 2" (Proprietary).

2. 1 copy of WCAP-12738, "LOFTR2 Analysis for a Steam Generator Tube Rupture of Beaver Valley Power Station, Unit 2" (Non-Proprietary).

Also enclosed are a Westinghouse authorization letter, CAW-90-083, accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 contains information proprietary to Westinghouse Electric Corporation, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-90-083 and should be addressed to R. A. Wiesemann, Manager of Regulatory & Legislative Affairs, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.