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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR RELIEF FROM THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME)

CODE REPAIR REQUIREMENTS

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-317

1.0 BACKGROUND

Temporary Non-Code Repairs

10 CFR 50.55a(g) requires nuclear power facility piping and components to meet the applicable requirements of Section XI of the ASME Boiler and Pressure Vessel Code (hereafter called the Code). Section XI of the Code specifies Code-acceptable repair methods for flaws that exceed Code acceptance limits in piping that is in service. A Code repair is required to restore the structural integrity of flawed Code piping, independent of the operational mode of the plant when the flaw is detected. Those repairs not in compliance with Section XI of the Code are non-Code repairs. However, the required Code repair may be impractical for a flaw detected during plant operation unless the facility is shut down. Pursuant to 10 CFR 50.55a(g)(6)(i), the Commission will evaluate determinations of impracticability, and may grant relief and may impose alternative requirements. Generic Letter (GL) 90-05, entitled "Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping," dated June 15, 1990, provides guidance for the staff in evaluating relief requests submitted by licensees for temporary non-Code repairs of Code Class 3 considering the guidance in GL 90-05.

Licensee's Relief Requests

By letter dated November 27, 1991, Baltimore Gas and Electric Company, the licensee, requested relief from making ASME Code repairs to a leak in the No. 11 Saltwater Header on the service water supply line of Calvert Cliffs, Unit 1. This is a moderate energy Code Class 3 system. The leak is approximately 3/8-inch diameter and located on the inside radius of a 30-inch diameter, concrete lined, carbon steel, 90 degree elbow. The elbow is attached to the discharge end of a butterfly valve (1-CV-5150) and upstream of the inlet to the No. 11 Service Water Heat Exchanger. The licensee evaluation of the leak concludes that the probable cause is accelerated corrosion as a result of failure to the concrete liner. The liner failure is attributed to stress fluctuations caused by flow instabilities immediately downstream from the butterfly valve (1-CV-5150). The licensee has determined that conformance with Code repair requirements is impractical. Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee submitted a relief request to the Commission. The temporary leak mitigation measures consists of a rubber patch with a steel plate backing fastened to the pipe with a U-bolt. Replacement of the degraded elbow is planned for the next refueling outage scheduled for March 6, 1992.

2.0 EVALUATION OF RELIEF REQUESTS

Code Requirement

Article IWA-4000 of Section XI of the Code specifies the Code repair procedures.

Code Relief Requests

Relief is requested from performing Code repairs of the flaw detected during plant operation in Code Class 3 piping.

Basis for Relief

Code repair requirements are impractical unless the unit is shut down.

Proposed Alternative

The licensee proposed to utilize the guidance in GL 90-05 to perform a temporary non-Code repair. Replacement of the degraded piping is planned for the next refueling outage scheduled for March 6, 1992.

3.0 STAFF EVALUATION AND CONCLUSION

The staff has determined that the Code repair requirement for a flaw in the 30-inch diameter elbow in the Unit 1, No. 11 Saltwater Header is impractical, as defined in GL 90-05. The flaw detected is in Class 3 piping and cannot be isolated to complete a Code repair within the time permitted by the limiting condition for operation. Compliance with the Code would require plant shutdown unless the system is redesigned. The staff finds, based on the licensee's flaw analysis, that the flawed piping has adequate structural integrity.

Furthermore, the licensee has committed to the guidance provided in GL 90-05 which will provide reasonable assurance that structural integrity will be maintained, and thus, the public health and safety will continue to be protected. Accordingly, the staff concludes that granting relief where code requirements are impractical and imposing alternative requirements are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest, given due consideration to the burden upon the licensee and facility that could result if the Code requirements were imposed on the facility. Pursuant to 10 CFR 50.55a(g)(6)(i) and consistent with the guidance in GL 90-05, relief is granted until the next scheduled outage exceeding 30 days, but no later than the next scheduled refueling outage. The temporary non-Code repair must then be replaced with Code repair or the flawed elbow can be replaced.

Principal Contributor:
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Date: February 19, 1992

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BG&E utilized the guidance provided in GL 90-05 in its relief request and the NRC staff has determined that a code-acceptable repair is impractical, as defined in GL 90-05, the flawed pipe has adequate structural integrity, and there is reasonable assurance that the structural integrity will be maintained until the flaw can be repaired during a plant shutdown. Additional details are included in the enclosed Safety Evaluation. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), the NRC staff concludes that code-acceptable repairs are impractical, the temporary non-code repairs are consistent with the guidance of GL 90-05, and relief is granted until the next scheduled outage exceeding 30 days, but no later than the next refueling outage during which the temporary non-code repair must be replaced in accordance with the ASME Code, Article IWA-4000, or the flawed piping replaced. Such relief is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. This relief has been granted giving due consideration to the burden upon the licensee that could result if the requirement were imposed upon the facility.

This completes our action related to the above referenced TAC number.

Sincerely,

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Safety Evaluation

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Robert A. Capra, Director
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