PDR PDR

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

NOV 2 2 1978

Docket Nos. 50-369
and 50-370

Mr. William O. Parker, Jr.
Vice President, Steam Production
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

SUBJECT: AUGMENTED INSERVICE INSPECTION FOR PIPE RUPTURE PROTECTION
(McGuire Nuclear Station, Units 1 & 2)

During our conference telephone call with Mr. L. C. Dail on November 15, 1978, we discussed our need for specific additional information regarding your proposed augmented inservice inspection program for pipe rupture protection. This information is described in the Enclosure.

In order to proceed in our review of this matter we have arranged a meeting with your staff for November 30, 1978, to discuss the matters described in the Enclosure. Your response would be subsequently reflected in an amendment to the McGuire Final Safety Analysis Report. Participants at the meeting should be prepared to discuss each item in detail with appropriate handout information. It should be emphasized that lacking satisfactory information, we will be unable to conclude our review of your application.

Sincerely,

Robert L. Baer

Robert L. Baer, Chief
Light Water Reactors
Branch No. 2
Division of Project Management

Enclosure:
As Stated

cc w/enclosure:
See page 2

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cc: Mr. W. L. Porter
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Mr. R. S. Howard
Power Systems Division
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

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EDS Nuclear Incorporated
220 Montgomery Street
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cc: Robert M. Lazo, Esq., Chairman
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
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Washington, D. C. 20555

Dr. Cadet H. Hand, Jr., Director
Bodega Marine Lab of California
P. O. Box 247
Bodega Bay, California 94923

ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION
MCGUIRE UNITS 1 AND 2

1. When discussing the consequences of the postulated pipe breaks in the following five questions, consideration should be given only to the effects which can be mechanistically demonstrated to occur. If detailed analyses show that various postulated pipe breaks result in lesser damage to other components than indicated in your Report No. SRG-78-01 (Revision 1), "Augmented Inservice Inspection For Pipe Rupture Protection," than your responses need only consider the damage predicted. Detailed analyses, including dimensions and material properties, should be submitted which indicate the extent of these effects due to the actual structural and thermodynamic conditions as outlined in Position C.4.c of Regulatory Guide 1.46 and Positions B.3.a.(3) thru (5) and B.3.b.(4) and (5) of Branch Technical Position MEB 3-1.

   a. You state that break point No. 4 of the upper head injection would cause a critical crack in the 20" Nuclear Service Water (NSW) line to the component cooling heat exchanger. The critical crack in the NSW line results in flooding the Aux building basement and temporary loss of equipment cooling. Provide information that indicates the consequences of temporary loss of component cooling water assuming up to 20 minutes for operator action to switch to redundant CCW heat exchanger and demonstrate that the plant can still be brought to safe shutdown condition.

   b. Provide the results of a long term analysis of the containment transient response to the simultaneous double-ended rupture of a 10-inch accumulatory injection line (and the resulting damage to the 16-inch steam generator feedwater line and 2-inch steam generator blowdown line. Itemize and justify all assumptions made in the analysis, including the mass and energy release rate data.

c. Provide the results of a long term containment transient response analysis for the double-ended rupture of an auxiliary feedwater line at the steam generator nozzle and the resulting damage to the enclosure, assuming direct release of the break flow to the containment upper compartment; i.e., assuming the break flow bypasses the ice condenser. Itemize and justify all assumptions made in the analysis, including operator actions performed to limit steam generator blowdown releases. Provide tabulated information of all input data required for the analysis, including the mass and energy release rate data.

d. Identify the piping systems and containment penetrations associated with each of the following containment isolation valves: NM 217B, NM207A, NM22A, NM54A, RN253A, RN276A, NI267A, NI266A, KC332B, and KC424B. Discuss any backup capability such as closed system piping, for providing containment integrity in the event the above valves fail to close because of damage to the cabling leading to the valve operators (as a result of an accumulator line failure) and coincident failure of the off-train diesel generator. If continued containment integrity cannot be shown, we will require a commitment to relocate the cabling or provide local protection for the cables.

e. With regard to the consequences of breaks (Appendix V to Vol. 2), the applicant is required to provide analysis to assure that the following initiating break and subsequent damage to targets will not lead to violation of GDC 35 or 10 CFR 50.46:

Initiating break node - 153, Sheet 3, Appendix V

Longitudinal targets - 6-inch hot leg injection line, RTD

1 1/2-inch boron injection line

2. After discussions at Duke Power on August 14-15, 1978, the staff was provided a consequence analysis (enclosed in meeting summary dated August 17, 1978); there are some differences noted between that consequence analysis and that in Appendix V, e.g., the meeting enclosure -- break 153, case 2, shows 3" charging line, 4" pressurizer spray, and RTD manifold targets; the RTD manifold target is missing in Appendix V. The applicant should explain these differences.

3. Provide a discussion of the acoustic leak detection system, describing the method and equipment design in detail. Information should include limits of detection capability and any data which demonstrate effectiveness of the proposed leak detection system.

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