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June 24, 1981
EF2 - 53,873

Mr. L. L. Kintner
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

RECEIVED
JUN 25 1981
U.S. NUCLEAR REGULATORY COMMISSION

Dear Mr. Kintner:

Reference: Enrico Fermi Atomic Power Plant, Unit 2
NRC Docket No. 50-341

Subject: Position ASB-1 (Generic BWR Scram Pipe Breaks)
Position CEB-2 (Post-Accident Sampling)
Position RSB-11 (II.K.3.18-ADS Logic Mods)
Fracture Toughness (GDC-51)
Position MTEB-3 (ISI-Head Spray/SLCS)

Detroit Edison's responses to these NRC Staff positions are enclosed. This information will be included in a forthcoming FSAR amendment as appropriate.

Sincerely,

William F. Colbert
William F. Colbert
Technical Director
Enrico Fermi 2

WFC:jl

cc: Mr. B. Little

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NRC QUESTION - CRD SYSTEM - ASB - 1

By letter dated May 5, 1981 we requested information regarding our Office of Analyses and Evaluation of Operational Data (AEOD) report entitled, "Safety Concerns Associated with a Pipe Break in the BWR Scram System." The report describes a potential sequence of events which could result from a break in the BWR scram discharge piping during a scram condition concurrent with an inability to reclose the scram outlet valves. Provide generic information requested.

RESPONSE

Detroit Edisons response to the May 5, 1981 letter regarding a break in the Scram Discharge Volume is being directed to Robert Tedesco, Assistant Director for Licensing. The May 5, 1981 letter requests a generic evaluation of the Office of Analysis and Evaluation of Operational Data (AEOD) report entitled, "Safety Concerns Associated with a Pipe Break in the BWR Scram System." Detroit Edison has reviewed the responses in the General Electric NEDO-24342, "General Electric Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Breaks," and has confirmed that the generic responses are applicable for Fermi 2.

Detroit Edison is aware of the coordinated efforts being made by the NRC and General Electric on behalf of the BWR licnesees to resolve this issue. A generic evaluation of NEDO-24342 is being made by the NRR and is scheduled for late July, 1981. The applicant will review the Fermi 2 Scram Discharge Volume design for compliance with the criteria of the generic safety evaluation report in submitting the plant specific information.

J. R. Green
/dk
6-24-81

DECO RESPONSES TO NRC

1-II.B.3 - Requirement for Monitoring Chloride in Reactor Coolant

Detroit Edison's exception to the monitoring of chloride as stated in draft Amendment 36 is based on typical BWR operation in which chlorides are closely monitored and controlled. With this philosophy in mind, one would not expect chlorides in an accident situation unless, for example, heat exchanger tube leaks or massive demineralizer performance reductions occurred simultaneously. In either case, process instrumentation would provide an indication immediately of possible chloride intrusion and corrective action would be implemented.

Chloride monitoring of reactor coolant will be performed if process instrumentation indicates a possible chloride intrusion. This situation would initiate the chloride monitoring program within 96 hours. The program will involve off-site analysis of a grab sample and/or continuous on-line chloride analysis by specific ion electrode.

We do take exception to the st ff response recommending the control of chlorides during an accident. Chloride control is not possible during an accident situation.

4-II.B.3 - Requirement for Monitoring Boron in Reactor Coolant

Detroit Edison takes exception to the requirement for the monitoring of boron because boron is not a normal constituent in reactor coolant. Boron monitoring will be performed subsequent to boron injection.

Capability for boron analysis will be provided by off-site analysis of a grab sample and/or continuous on-line boron analyzer utilizing specific ion electrode.

2-II.B.3(4) - Recommendation for Monitoring Dissolved Oxygen in Reactor Coolant

Detroit Edison's exception to monitoring dissolved oxygen is because during an accident situation, dissolved oxygen levels cannot be controlled. Verification that oxygen is $>$ 0.1 ppm will not initiate any corrective action. However, monitoring of the reactor coolant oxygen for chloride stress corrosion cracking susceptibility of associated safety systems will provide input for assessment of safety boundary integrity.

Detroit Edison will monitor dissolved oxygen by providing grab sample capability for off-site analysis and/or an on-line polarographic oxygen analyzer.

CEB-2
2

3.1.97 (Revision 2) Requirement for Monitoring pH in Reactor Coolant

Detroit Edison takes exception to the recommendation for monitoring pH in the reactor coolant to verify that pH is $\geq$ 7.0 to assure protection against chloride stress corrosion cracking. BWR operating philosophy dictates operation in the neutral pH region with no provisions for chemical additions for pH control. In order for pH to be $\geq$ 7.0 in an accident condition, some chemical addition would be required which would be contrary to operating philosophy.

The monitoring of chloride, oxygen, and temperature is adequate to assure protaction against chloride stress corrosion cracking in safety related systems subsequent to an accident.

B.1a Following a DBA event, the reactor water level is not maintained above 2/3 core height. In this scenario, the reactor vessel is not pressurized and as a result the intended sample source is the discharge of the RHR pumps which are acting as the driving force for the circulation of reactor coolant.

B.1b Suppression pool atmospheric samples are taken from taps on opposite sides of the pool proper. Each tap location is selected to maximize the distance to either a downcomer or safety/relief valve discharge sparger. Since both of these steam sources discharge under water, there is no significant affect on the atmospheric sample.

Liquid samples from the suppression pool are obtained from the RHR pump discharge. RHR pump suction strainers are located as far as practical from both the downcomer and S/R valve discharges in order to maximize the temperature effects on pump NPSH. As a result, the sample should be representative of the mixture in the pool.

B.2 The post accident sampling control panel and all electrically operated valves within the sample station will be supplied with restorable balance of plant power. NUREG-0737 does not stipulate that the post-accident sampling system is classified as a safety grade system. Restorable power provides the necessary reliability for this system.

B.3 The sampling station is located in the Auxiliary Building on the first floor (elevation 583'-0") between columns G12 and H12. From the sample station, the samples are either manually carried (small cask) or wheeled on a dolly (large cask) to the analytical laboratory or to an exit for off-site shipment. The laboratory and exit are at elevation 583'-6" and are within 150 ft. of the sample station. The route to be traversed is through a doorway into the Turbine Building with a turn north to the analytical laboratory or south to the exit. It is estimated that these routes can be conservatively walked in less than 2 minutes. The transit doses (MREM) for a 2 minute walk from the Main Control Room to the sample station has been calculated as 2.1 for 1 hour, 6.9 for 1 day, 4.0 for 1 week, and .5 for 1 month. The analytical laboratory is in the general area of the control room except that it is at the same elevation as and closer to the sampling station. It can be extrapolated that the transit doses would be conservatively equal to or less than the doses identified for the main control room.

B.6 The post-accident sample collection system will be operated on a semiannual basis to insure operability and provide the necessary operator training. Specific procedures will be generated to include procurement, transporation and analysis of samples. Diluted and undiluted liquid and gaseous samples will be obtained from the reactor coolant, drywell containment, and suppression pool. This will allow verification of the operation of all components on the sample station and permit the operator to familiarize himself with the operation of sample station. All analytical instrumentation will be calibrated using approved Radchem procedures and traceable to the National Bureau of Standards.

B.4 The approved Radchem procedures used for the monitoring of reactor coolant are traceable to the National Bureau of Standards (NBS) or the American Society for Testing and Materials (ASTM) and qualified for accuracy and precision in an accident environment. This instrumentation will meet or exceed the accuracy ranges specified in Regulatory Guide 1.97 (Revision 2).

B.5 When an acceptable technique for estimating fuel damage is prepared, Detroit Edison will review it for inclusion in Enrico Fermi 2 procedures

RSB - 11 (II.K.3.18 - MODIFICATIONS TO ADS LOGIC)

Detroit Edison commits to complete and test the modifications to ADS logic required by NUREG - 0737 Item II.K.3.18 prior to fuel loading. FSAR Page H.II.K.3.18-2 will be revised to so indicate this schedule committment in a forthcoming FSAR amendment.

6/24/81

GDC 51 - MSIV Ductility

In response to a concern by NRC staff, Detroit Edison submits pertinent data on the Quaker Alloy Casting Co. Main Steam Isolation Valves as submitted for the TVA X20 and Grand Gulf 1 plants. This data was taken from the SSES response to NRC questions 251.1 - 251.5. This data indicates that additional heat treatment increases the ductility of the MSIV.

We believe this data will resolve the NRC staff concern on this item.

Table 11 Sh. No 9

Project TVA X20

Valve MSIV

Component Body

Applicable Code ASME Sect. III, 1974 with summer 1975 addenda.

Valve Vendor Atwood & Morrill Co., Material Vendor: Quaker Alloy Casting Co.

Material Specification - ASME SA216 Grade WCB

Heat No. F3547

Chemical Composition (wt. %) - C/.23 Mn/.88 Si/.38 P/.016 S/.015 Al/NA

Grain Size (ASTM No.) - NA

Heat Treatment 1700/1725°F (6 hr. 20 min.) air cool + temper
1345°F (6 hr. 45 min,) air cool + post weld
1200/1225°F (6 hrs. 30 min) air cool

Charpy V - Notch Impact Toughness

Test Temperature : +60°F

Ft-lb : 66, 56, 54

Mils: 53, 50, 53

% Shear : 40, 40, 40

- Not Available

GDC-51

Project Grand Gulf 1

Valve MSIV

Component Body

Applicable Code ASME Sect. III, 1974

Valve Vendor Atwood & Morrill, Co.

Material Vendor Quaker Alloy Casting Co.

Material Specification - ASME SA 216 Grade WCB

Heat No. - F6406

Chemical Composition (wt %) - C/.23 Mn/.89 Si/.53 P/.019 S/.012 Al/NA

Grain Size (ASTM No.) - NA

Heat Treatment 1680/1710°F (5 hrs. 30 min) air cool + temper
1350°F (5hr. 30 min) air cool + Post weld 1200 F (6 hr.) air cool

Charpy V - Notch Impact Toughness

Test Temperature +60°F

Ft-lb 32,31,34

Mils 33,32,31

% Shear 40,40,40

NA - Not Available

MTEB-3 response to NRC staff position

The Detroit Edison Company agrees to incorporate the Head Spray and Standby Liquid Control Systems into the Class 2 in-service inspection program. The selection criteria for welds in these systems will be the same as described in Class 2 System Weld Program, Enrico Fermi Atomic Power Plant Unit 2, 30, April 1981.

6/24/81