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September 22, 1982
EF2-59174

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

Dear Mr. Kintner:

References: (1) Enrico Fermi Atomic Power Plant, Unit 2 NRC Docket No. 50-341.

(2) Letter from Robert L. Tedesco to H. Tauber, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System," dated May 5, 1981.

(3) Letter from Glenn Sherwood to Darrell Eisenhut transmitting NEDO-24342, "GE Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Breaks," dated April 30, 1981.

(4) Letter from H. Tauber to R. Tedesco, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System, EF2-53,890, dated July 10, 1981.

(5) Letter from Darrell Eisenhut to all BWR Applicants for CP's, Holders of CP's, and Applicants for OL's, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System (Generic Letter 81-35)," dated Aug. 31, 1981.

Subject: Response to NUREG-0803

Mr. Tedesco's letter (Reference 2) requested that we submit a generic and a plant specific response for scram system pipe breaks. The report prepared by General Electric (Reference 3) was the basis for our generic response (Reference 4). Reference 5 forwarded NUREG-0803 which provided further NRC definition for the plant specific response. The Fermi 2 plant specific response to NUREG-0803 is enclosed.

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This completes our responses to the safety concerns associated with pipe breaks in the BWR Scram System. If you have any questions regarding the above, please contact Mr. Larry E. Schuerman, (313) 649-7562.

Sincerely,

Harry Tauber

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GED/106/3.1
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Enclosure to
EF2-59,174

Response to NUREG-0803

for the

Enrico Fermi Atomic Power Plant, Unit 2.

I. SUMMARY

Detroit Edison has reviewed NUREG-0803 and generally agrees with the generic conclusions contained therein. The Detroit Edison Fermi 2 specific positions with respect to the generic conclusions in Section 5 of NUREG-0803 are contained in Section III of this enclosure. The positions show that Fermi 2 is in essential compliance with the pertinent parts of the guidance. Detroit Edison takes exception to the NRC conclusion in NUREG-0803 that a sufficient data base does not exist to terminate the review of this issue on the basis of a quantitative risk assessment. A revised analysis performed by G.E. in behalf of the BWR Owners' Group (and verified applicable to Fermi 2) shows that this is not the case. As a consequence, no further action concerning this matter should be required.

II. DISCUSSION

The NRC developed a three-phase approach in NUREG-0803 for evaluating the safety concern associated with pipe breaks in the scram system. The first step was to determine the probability of a Scram Discharge Volume (SDV) pipe failure and to evaluate the contribution of such a failure to a core melt. The NRC's generic analysis resulted in a probability of 10^{-4} per plant year for a SDV pipe failure and less than 10^{-6} per plant year as the frequency of a core melt given the operability of mitigation equipment. The NRC continued in their evaluation of the consequences of the postulated scenario because these generic probabilities were not sufficiently small or substantiated.

Detroit Edison supported the Boiling Water Reactor Owners' Group (BWROG) in performing a plant specific probabilistic risk assessment (PRA) since both the NRC's NUREG-0803 and General Electric's (GE) NEDO-24342 generic analyses were overly conservative in assessing the plant specifics. A GE prepared report, NEDO-22209, August, 1982, "Analysis of Scram Discharge Volume System Piping Integrity", documents the results of this BWROG study to determine the probability of the loss of SDV piping integrity, and to evaluate the contribution of such a loss to a core melt. This report is attached to this submittal.

The results of the BWROG report show that the probability of an unisolatable loss of scram system piping integrity is 3×10^{-7} per plant year. As calculated in NEDO-24342, the probability of core damage given an unisolatable SDV rupture for Mark I and II plants is 6×10^{-5} per plant. Consequently, the combined probability for core damage for this event is 2×10^{-11} events per reactor year. This probability is sufficiently small to preclude further efforts related to this matter for Fermi 2.

BWROG Report Methodology

The BWROG report provides a more detailed analysis of the failure probability of the SDV piping, takes into account plant specific data, and evaluates the failure of valves as a loss of SDV integrity.

Three different approaches were used to evaluate the SDV pipe break probability: the first approach (based on NEDO-24342) used the assessed probability for a LOCA per Wash-1400 and modified the probabilities by the ratio of SDV piping length to LOCA sensitive piping length; the second approach (based on NUREG-0803) estimated the SDV piping length and multiplied it by a failure rate of 3×10^{-7} per foot per year to obtain a break probability; the third approach, fracture mechanics, evaluated break probabilities by analyzing the mechanism of crack growth while under repeated stress.

II. DISCUSSION (Cont'd)

BWROG Report Methodology (Cont'd)

The first two experience approaches yielded a break probability of 1×10^{-7} per plant year. The fracture mechanics approach supported this low probability in that no failures for the cyclic stress for scram cases were predicted since the stresses were insufficient to increase cracks to a critical size.

A loss of SDV piping integrity is also evaluated in the BWROG report. This analysis includes the previously discussed SDV pipe break as well as the failure of a pressure relief valve or the failure of two (2) drain or two (2) vent valves. The evaluation is based upon plant specific data from 14 utilities with MK I or MK II containments. The plant specific data used in the PRA includes scrams per year, scram reset times, welded joint information, and piping diameters and lengths for the SDV configuration including post-Browns Ferry 3 modifications.

Applicability to Fermi 2

A study of a breach of SDV integrity for an "average" plant and a "limiting" plant is included in the BWROG report. The average plant refers to a plant having the average pipe lengths, number of scrams and scram duration of the plant specific data. Since Fermi 2 is not yet operating, the application of data from the operating plants is appropriate for the number and duration of scrams. The average plant data are also representative of Fermi 2. Thus, for Fermi 2 the probability for a loss of SDV integrity is 1×10^{-7} per reactor year as calculated in the BWROG PRA using plant specific data and the approach of NUREG-0803. Similarly, the probability of a SDV break leading to a core melt is approximately 2×10^{-11} per plant year.

The NRC, in NUREG-0803, stated that "it was agreed that if the probability of core damage from the postulated scenario was shown to be sufficiently small, no further review, beyond verification of plant-specific response applicability, would be necessary." They further noted that "as the review progressed, it became evident that a sufficient data base did not exist to conservatively terminate the generic review on the basis of a quantitative risk assessment." The BWROG analysis uses plant specific data and shows that the probability of core damage is sufficiently small.

It is thus concluded that the probability of core damage initiated by a failure of the SDV piping integrity is sufficiently low so as to preclude the necessity of qualification or design modification for equipment required to detect and/or mitigate the consequences of such an integrity loss.

III. DETROIT EDISON POSITIONS RELATIVE to SECTION 5 of NUREG-0803

Table 5.1 of NUREG-0803 summarizes NRC staff's guidance for an acceptable plant-specific resolution of the concerns with regards to the SDV pipe break. The Fermi 2 specific responses listed below demonstrate that the mechanical quality, maintenance procedures, operator actions and existing system performance are adequate to resolve these concerns. Information given in these responses includes data on SDV piping integrity but for which no credit was taken in the risk analysis previously done. This information plus using the existing provisions of NRC Standard Review Plan (SRP) 3.6.1 for moderate - energy fluid systems and 3.6.2 for flow from a crack demonstrate the conservatism of the NRC analysis.

The plant specific assurance of mechanical quality is in agreement wth the original NRC guidance of NUREG-0785, Safety Concerns Associated with Pipe Breaks in the BWR Scram System. This AEOD report states that, "If from these convolutions one were to conclude that the SDV pipe break is a significant contributor to BWR core uncovery risk, it is believed that the risk can best be reduced by decreasing the likelihood of a break in the SDV system piping by an appropriate upgrading of the SDV mechanical integrity assurance basis." Since by the G.E. study the SDV pipe break has been shown to be an insignificant contributor to core uncovery risk, Detroit Edison strongly believes in line with the above recommendation that further efforts should be in the area of insuring SDV system piping integrity. This is the fundamental basis for the Detroit Edison positions discussed below. For convenience, the Fermi 2 positions are listed in the same sequence as on Table 5.1 of NUREG-0803.

1. PERIODIC IN SERVICE INSPECTION

The SDV system is included in the Fermi 2 In-Service Inspection Program in conformance with the requirements of Class 2 piping under Section XI of the Boiler and Pressure Vessel Code" of the American Society of Mechanical Engineers (ASME Code). The pre-service inspection is based on the 1977 Edition of Section XI Code up to 1979, Summer Addenda. The in-service inspection is presently based on the 1980 Edition of Section XI Code up to 1980 Winter Addenda. The In-service and Inspection Testing Program has been previously submitted to the NRC.

2. THREADED JOINT INTEGRITY

Welded joints are utilized in the SDV system piping. A review of design and as-built drawings and a site inspection were conducted to verify the exclusion of threaded joints in the SDV system piping. The SDV system piping is designed, fabricated and installed to the requirements of the ASME Code, Section III, for Class 2 components.

3. SEISMIC DESIGN VERIFICATION

The SDV piping for Fermi 2 is Quality Class I and is designed for Seismic Category 1 loadings. A reanalysis of the seismic design for the SDV piping has been recently completed. While the original design and seismic analysis were completed during mid 70's a large portion of the installation did not occur until 1981/82. Reasons which necessitated a reevaluation included interference problems and modified requirements for seismic analysis from the NRC such as IE Bulletins 79-02 and 79-14. The redesign and reanalysis work was followed immediately by construction. The seismic design report is updated to reflect as-built conditions as stated under Item 6.

4. RCU - SDV EQUIPMENT PROCEDURE REVIEW

Work performed on safety related systems' equipment must be performed under the control of approved procedures and instructions. These provide assurance that the maintenance and surveillance activities are performed in a controlled and safe manner. SDV maintenance work, surveillance tests or modifications must have the approval of the Nuclear Shift Supervisor before starting. The Shift Supervisor evaluates the work to be performed on the equipment with regards to its potential impact on plant operations and personnel protection. Where isolation cannot be obtained, work is not permitted until proper plant conditions exist. HCU - SDV equipment maintenance affecting system integrity would require cold shutdown.

Documentation is provided to assure the Shift Supervisor that the work was completed and operationally verified. This documentation is in accordance with General Maintenance Procedure 12.000.15, Surveillance Program Procedure 12.000.18 and Tagging and Protective Barrier System Procedure 12.000.12. The requirements of these procedures are contained in the SDV piping/equipment maintenance and surveillance procedures and instructions.

5. ENVIRONMENTAL QUALIFICATION OF PROMPT DEPRESSURIZATION FUNCTION

The Automatic Depressurization System (ADS) accomplishes the reactor depressurization function. The equipment and instrumentation required for depressurization is located remote from the environment resulting from a SDV pipe break. The Automatic Depressurization System, as are others, is required for safe shutdown of the reactor. These systems are included for evaluation under the NUREG-0588 environmental qualification program. Further specific evaluation is not warranted considering our established qualification program and the low probability of the event as detailed in the Discussion.

6. AS-BUILT INSPECTION

The as-built conditions of the SDV piping and supports are incorporated in the design and seismic analysis for the system. The design, analysis and installation program is such that as-built drawings are completed and provided for input to an as-built revision of the stress reports. As-built drawings include small diameter piping.

The original design was performed and the start of installation was begun during the mid-70's. As stated in Item 3, the design rework occurred in 1980/81 with the installation being essentially complete as of May, 1982. The reconciliation of as-built field conditions to the seismic analysis is scheduled for completion in late 1982. There are nearly 200 SDV piping and support drawings and 25 stress reports that will reflect as-built conditions.

The designers and constructors for the SDV system have quality assurance programs that are approved by Detroit Edison. The requirements of these internal programs, as well as the audits and surveillance performed by Detroit Edison QA of these vendors, provide verification of the quality assurance for the SDV piping and supports with regard to as-built conditions.

7. IMPROVEMENT OF PROCEDURES

The operator actions necessary to mitigate the consequences of a potential rupture of the SDV are delineated in Fermi 2 Abnormal Operating Procedure 20.106.11, Scram Discharge Volume Failure. This procedure is executed concurrently with Abnormal Operating Procedure 20.106.01, Reactor Scram in the event the scram cannot be reset within five (5) minutes of initiation. The existing indication and instrumentation provides the operator with information to detect a SDV pipe break. Information available from the Control Rod Drive System includes multiple indications and alarms for drive temperature increases, rod insert overtravel and rod drift. Information to detect a leak in the Reactor Building includes sump level alarms, area radiation monitors, corner room level and temperature alarms and operator observation. If the scram cannot be reset within 25 minutes of the scram initiation signal AND the integrity of the SDV cannot be physically verified AND any of the multiple control rod indicators exist, the operator is directed to manually depressurize the reactor.

7. IMPROVEMENT OF PROCEDURES (Cont'd)

The Emergency Operating Procedures for Fermi 2 have not been revised for a SDV pipe break pending the activities of the BWR Owners' Group. Per its charter, the BWR Owners' Group cannot respond to NRC requests for utility action except at the discretion of its members. Neither can Detroit Edison commit the BWR Owners' Group to a specific course of action except by its participation in Owners' Group decisions by vote. Thus, Detroit Edison can only provide a response to the Staff's guidance to the BWR Owner's Group decisions by vote. Thus, Detroit Edison can only provide a response to the Staff's guidance to the BWR Owners' Group in NUREG 0803 as if it were addressed to Detroit Edison directly.

However, the BWR Owners' Group has discussed the guidance of NUREG-0803 regarding modifications of the Emergency Operating Procedures Guidelines and acknowledges the benefits of treating the subject generically. The BWR Owners' Group is in the process of completing an extension of the guidelines to include steps for reactivity control, and certain other modifications to the guidelines which have been discussed with your staff. It is Detroit Edison's judgment that completion of these modifications outweighs, in immediate importance, the NUREG-0803 guidance for other guideline modifications. After current activities of the guidelines are substantially complete, Detroit Edison will support a preliminary study by the BWR Owners' Group to determine the best approach to fulfilling the intent of the guidance provided in NUREG-0803. It is not clear that the best approach will involve modification of the guidelines. When the study is complete, the Owners' Group will determine whether to authorize specific actions to modify the Emergency Procedure Guidelines. Detroit Edison will advise the NRC of the result of that decision and the Owners's Group plan at the time.

8. VERIFICATION OF EQUIPMENT DESIGNED FOR WATER IMPINGEMENT

See Item 5 and DISCUSSION

9. VERIFICATION OF EQUIPMENT QUALIFIED FOR WETDOWN BY 212°F WATER

See Item 5 and DISCUSSION

10. FEEDWATER AND CONDENSATE SYSTEM OPERATION

The feedwater system is dependent on reactor building equipment for steam to drive the reactor feed pumps via the MSIVs and for level control instrumentation. This instrumentation is at another elevation and is not

10. FEEDWATER AND CONDENSATE SYSTEM OPERATION (Cont'd)

within the harsh environment for this event. Even if the reactor feed pumps were unavailable, there are multiple motor driven condenser and heater feed pumps which provide sufficient capacity to maintain reactor level without full depressurization.

11. EVALUATION OF AVAILABILITY OF HPCI-LPCI TURBINES DUE TO HIGH AMBIENT TEMPERATURE TRIPS

Sufficient high pressure pump capacity exists for coolant inventory makeup requirements resulting from a postulated SDV pipe break. The HPCI and RCIC pumps are both turbine driven pumps with isolation of a specific pump upon detection of a local steam leak. Thermocouples sensing high ambient temperature are located in the HPCI room and the Northeast corner room (RCIC). The physical separation of these rooms and the flood control and separation within the equipment and floor drain systems limit the harsh environment effects such that simultaneous isolation of HPCI and RCIC will not occur. The HPCI and RCIC system equipment is being evaluated within our environmental qualification program. In addition to the HPCI and RCIC systems two (2) Standby Feedwater System pumps are designed to provide high pressure injection. Each non-Quality Class I backup pump provides 600 GPM at reactor pressure and is located in the Turbine Building. Finally, the LPCI system is a subsystem of the RHR system and uses motor driven pumps which do not isolate for any high ambient room temperature.

12. VERIFICATION OF ESSENTIAL COMPONENTS QUALIFIED FOR SERVICE AT 212° AND 100% HUMIDITY

See Item 5 and DISCUSSION

13. LIMITATION OF COOLANT IODINE CONCENTRATION TO STANDARD TECHNICAL SPECIFICATION VALUES

The reactor water concentrations stated in the General Electric Standard Technical Specifications are applicable to Fermi 2. The NRC Staff concluded in NUREG-0803 that "the STS would provide a reasonably conservative upper limit of reactor water iodine concentration."