Douglas R. Gipson
Plant Manager

Detroit Edison

Fermi 2
6400 North Dixie Highway
Newport, Michigan 48166
(313) 586-5326

April 9, 1990
NRC-90-0053

Nuclear Operations

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Reference: 1) Fermi 2
NRC Docket No. 50-341
NRC License no. NPF-43

Subject: Residual Heat Removal Piping Vibration Induced Fatigue Analysis

This letter documents the Engineering Functional Analysis of the Residual Heat Removal (RHR) piping vibration induced fatigue concerns and the subsequent system operability determination. Submittal of this information was agreed upon by Detroit Edison personnel and NRC Region III personnel during telecons on April 5 and 6, 1990.

BACKGROUND INFORMATION

During preoperational testing of the RHR System in 1982, piping and pump vibration concerns were noted. Detroit Edison, in cooperation with the pump manufacturer, pump supplier, and consultants, devised and executed field vibration testing. The RHR pump vibration data was used to establish an acceptance criteria for future vibration surveillance testing that was consistent with the design basis functions and life for the pumps. That criteria remains valid through the present time.

Piping vibration was evaluated under a separate program. The piping vibration was evaluated for fifteen (15) different flow modes under the Piping Vibration and Dynamic Effects Testing (PVDET) program described in the Updated Final Safety Analysis Report (UFSAR) Section 3.9-1. The program utilized the accumulation of acceleration and/or displacement data in multiple planes to confirm the ability of the piping to satisfactorily accommodate vibratory loading. The program included piping greater than four (4) inches in diameter, commonly referred to as large bore piping, and piping less than one (1) inch diameter in selected locations. The small bore piping locations selected were small diameter piping (less than one (1) inch diameter) that interface with large headers, typically at the sockolet fitting welds. Data accumulated from the large bore piping during preoperational testing was analyzed and the results considered acceptable for the RHR System prior to low power testing in 1985. The data for the small diameter branches was compared to acceptance criteria based on "simple beam" mathematical models. The system piping was also accepted for service. Further steady state vibrational testing was performed for the RHR System during the startup test program, as is documented in STUT.02A.033, STUT.03A.033 and STUT.06A.033. These confirmed the acceptability of the piping structure.

A061
1/0

Beginning in December of 1986, vibration related concerns began to become evident in the RHR pump rooms. A large bore piping hydraulic snubber oil reservoir experienced weld cracking and was separated from the main body of the snubber. In March of 1987, a three quarter (3/4) inch cantilevered drain connection off the twenty (20) inch diameter pump "A" discharge line experienced through wall cracking. This connection was one of the small diameter branch connections examined under the PVDET program. The connection was shortened and vibration readings taken on the repaired connection while in torus cooling mode were acceptable. Visual inspection of other small connections showed no unusual susceptability to vibration loading. The system and pump vibration levels were consistent with the levels measured during pre-operational testing and, therefore, piping excitation levels previously identified were unchanged.

A RHR vibration monitoring and analysis program was established in October of 1987 due to vibration-related fatigue failures of components not covered by the PVDET program. Field measurements of vibratory displacements or accelerations were taken as plant operation permitted, beginning in 1988. In April of 1988, a three quarter (3/4) inch diameter chemical injection tap experienced through wall cracking causing a minor leak. This occurred at the time the "A" RHR pump had inadvertantly run dead headed for one half (1/2) hour. This operation was at higher than normal vibration. The higher vibration combined with the dead head operation was considered the cause of the tap cracking. In August, the small seal water cooler, whose threaded three quarter (3/4) inch diameter discharge nipple had cracked in August of 1987, cracked again causing a small water leak. The corrective action taken was to replace the pipe nipple containing cut pipe threads with one that was manufactured with rolled threads. The return line to this cooler cracked in the threaded connection in January of 1989 and was also repaired using a spool with rolled threads. In November of 1989, the three quarter (3/4) inch diameter supply line screwed nipple cracked a third time. The small cooler was removed from the pump casing mounting and remounted on a free standing support near the pump. In December of 1989, a three quarter (3/4) inch chemical tap developed a pin-hole at the sockolet weld and a small stream of water was observed. The connection was shortened and the failed weld replaced with a weld designed to reduce stress concentration.

Data for the Division 2 RHR pump room had been obtained by February 1989. The Division 1 data was taken later during the first refueling outage. The vibratory data accumulated in 1988 and 1989 was evaluated by the Detroit Edison Engineering Research Department (ERD). A preliminary report was received in late March of 1990. The report indicated that several of the small bore branch connections were experiencing vibratory stress levels in excess of the ASME/ANSI OM-3 (1982) acceptance level of 10 Ksi. This was computed using measured accelerations and simple beam (finite element) mathematical models. Based on the ERD results received to date, five connections have calculated loadings as high as 20 Ksi, which include the chemical injection tap that experienced the pin-hole leak in December of 1989

which was modified as previously described. (The other four connections identified have been recently modified, based on ERD results, to reduce stress concentrations.)

RHR OPERABILITY DETERMINATION

The RHR System has two loops, Division 1 and Division 2. Each loop can be operated in the following modes: (1) low pressure coolant injection (LPCI), (2) torus cooling, (3) shutdown cooling, (4) drywell or torus spray, (5) fuel pool cooling assistance, and (6) emergency core flooding.

Each loop of the RHR System is comprised of two (2) centrifugal pumps rated at 10,000 gpm, one (1) heat exchanger capable of removing heat from the reactor coolant at a flow rate of 10,000 gpm, and the interconnecting piping, valving, and controls required to permit the system to operate in any of the previously listed modes.

The design basis of the system is to mitigate the consequences of a design basis loss of coolant accident (LOCA) by either: 1) Providing cooling water flow under low pressure at a rate sufficient to prevent fuel cladding temperatures from reaching 2200 degrees Fahrenheit; or 2) by removing post-LOCA heat loads from the torus water at a rate sufficient to prevent safety related pump suction heads from falling below the required levels. Each mode of RHR operation is described in the UFSAR.

Detroit Edison considers the design of the RHR System and the ongoing vibration analysis program to be technically sound. The RHR System is currently considered operable for the following major reasons:

1) The welds on these small diameter connections were originally sized within the requirements of the ASME Code. Based upon the preliminary report from ERD, five small diameter connections were identified as possibly experiencing vibratory loadings in the 20 Ksi range versus the code allowable value of 10 Ksi. All of these connections plus two others (whose location and geometry indicate that, while not yet analyzed, could be susceptible to loadings of 20 Ksi magnitude) recently had their welds modified. The weld joining these nipples to the headers were "buttered" to a 2:1 slope and blended to reduce stress concentrations. This action provided an additional level of assurance that these connections will not fail.

2) The "endurance limit" of approximately 1,000,000 cycles for this material (the small diameter connections previously discussed) is indicative of infinite life. That is, material subjected to vibratory loadings which has not failed after 1,000,000 cycles is not considered likely to fail. These

connections, other than the one which failed in December, 1989, have experienced approximately 1,000,000,000 cycles and have not failed. This is based on the RHR System having been operated a total of approximately 500 days in various operating modes (especially shutdown cooling) and a fundamental frequency of approximately 80 Hz for these pressure taps.

3) The ASME fatigue curves, stress intensification factors, and the vibration measurement/evaluation methodology is conservative. The ASME/ANSI OM-3 standard and the material limiting stress values are based on conservative interpretations of empirical data. The fact that the remaining pipe nipples of concern have not failed, although the calculated field stresses are considered to be in excess of the required values in some cases, is indicative of proper initial design and installation configuration based on the state-of-the-art methodologies and criteria and data available at the time.

4) The leakage history of the nipples has been one of "leak before break". Catastrophic failures have not been experienced and are not expected to occur.

5) The small number of connection failures to date have been vibration induced fatigue failures caused by more aggressive RHR pump vibration than originally anticipated. However, the measurements taken to date and an analysis of this data do not indicate an increasing trend in these vibration levels.

6) Visual inspections of the RHR pumps and associated small diameter connections have confirmed acceptable tap configurations are present and that there is no further evidence of vibratory degradation or leakage at this time.

7) Previous failures have been corrected by weldment replacement/reinforcement and/or shortening the connections. No additional or repeat failures have occurred with the exception of a small cooler originally mounted on the "B" RHR pump casing. The small diameter threaded nipples experienced through wall cracking at the threads. The repeat failures occurred because the anticipated benefits in fatigue life by replacing the cooler mounting bracket and using rolled threads were not realized. This cooler was remounted on a free-standing support adjacent to the "B" RHR pump during the first refueling outage. This is fully expected to resolve the past problem. Similar coolers on the other three pumps satisfy OM-3 limits at the threaded connections, according to the recent analysis results.

8) Any failures which could be postulated would not prohibit the RHR System from performing its intended safety functions. This conclusion is based on (1) designed redundancy and separation of divisional pumps; (2) failures would be "leak

before break" and would be monitored or discovered without the need for unusual action; and (3) failure of three quarter inch connections under low pressure injection conditions is not expected to significantly reduce the RHR injection flow below 10,000 gpm for any RHR pump. Also, safety-related equipment in these corner rooms is qualified for the 100% humidity environment which would result.

CONCLUSIONS

The current RHR vibration analysis program was developed starting in 1987 in response to several vibration induced fatigue failures of components near the pumps. The type of failures which occurred have not (as expected) been catastrophic, such that the RHR System could not have performed its intended safety functions assuming this was the only failure. Operating history since the program was conceived indicates that the connections are acceptable, considering current vibration levels, especially with the weld modifications recently completed. When the analysis is completed in May, modifications will be developed to provide additional margin for future operation.

Based upon our recent discussions with Messers. T. Martin, M. Ring, R. DeFayette and W. Rogers of your staff, we will be meeting with you to further discuss this matter on April 16, 1990 at 1:00 p.m. central time in the Region III offices.

If there are any questions, please contact Michael Williams, Nuclear Engineering, at (313) 586-1066 or Patricia Anthony, Compliance Engineer, at (313) 586-1617.

Sincerely,

cc: A. B. Davis
R. W. DeFayette
W. G. Rogers
J. F. Stang
Region III