UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 236 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 218 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated September 17, 1999, the Indiana Michigan Power Company (the licensee) requested approval to make changes to the Updated Final Safety Analysis Report (UFSAR) and applicable emergency operating procedures (EOPs) to credit the negative reactivity provided by insertion of the rod cluster control assemblies (RCCAs) following any design basis Loss-of-Coolant Accident (LOCA). The use of the methodology and associated changes to the UFSAR and EOPs, when evaluated by the licensee in accordance with 10 CFR 59.59, resulted in an unreviewed safety question that requires prior approval by the NRC staff in accordance with the provisions of 10 CFR 50.90 prior to implementation. The proposed amendment would also change the Bases for Technical Specification (T/S) 3/4.5.5, "Refueling Water Storage Tank (RWST)," which is affected by the application of the methodology.

By letter dated October 26, 1999, the NRC staff made a request for additional information (RAI) concerning the licensee's leak before break (LBB) analysis. By letters dated November 10, 1999, and November 19, 1999, the licensee provided the requested information. The information contained in the November 10 and November 19, 1999, letters supplemented the September 17, 1999, application and did not change the Commission's preliminary significant hazards determination.

2.0 EVALUATION

The concern addressed by taking credit for the negative reactivity provided by insertion of the RCCAs pertains to the post LOCA dilution of boron in the containment sump liquid due to the boron concentrating in the reactor vessel. Following a LOCA the potential exists for the reactor coolant collected in the recirculation sump to decrease in boron concentration to a value at, or below, the critical boron concentration for the reactor core. When the emergency core cooling system (ECCS) alignment is switched from cold leg injection alignment to a hot leg injection alignment following a LOCA, the potential exists for the reactor core to return to criticality during the relatively short time frame when the realignment first takes place. This occurs during the switchover. After the switchover is accomplished and the ECCS is aligned to the hot leg recirculation configuration, the core is provided with a back-flushing flow that aids in re-establishing an evenly distributed boric acid concentration within the reactor vessel. The switchover subcriticality analysis conservatively assumes that the diluted boron sump liquid completely displaces the more highly borated liquid in the vessel during the transition from cold leg recirculation to hot leg recirculation. It is this conservative assumption that results in the postulated return to criticality at the time of the switchover. Therefore, taking credit for insertion of the RCCAs provides a significant source of negative reactivity that can be used to offset the conservative assumption to ignore mixing, and can be used to demonstrate post-LOCA subcriticality at the time the switchover is performed.

The licensee's justification for use of the control rod insertion methodology following a LOCA is the ability of the rod cluster control assemblies inserting into the reactor core following design basis LOCAs. The licensee analyzed the following reactor coolant system (RCS) breaks.

60 in^2 Accumulator Line Break

98 in^2 Pressurizer Surge Line Break

144 in^2 Reactor Vessel Inlet Nozzle Break

144 in^2 Reactor Vessel Outlet Nozzle Break

594 in^2 Reactor Coolant loop Outlet Nozzle

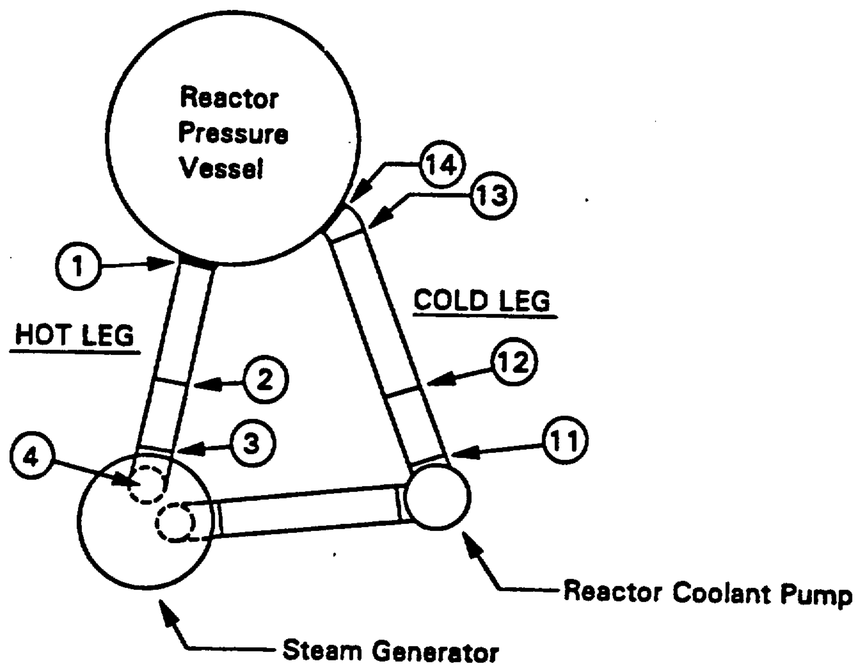
2.1 Main Coolant Loop Break Analysis

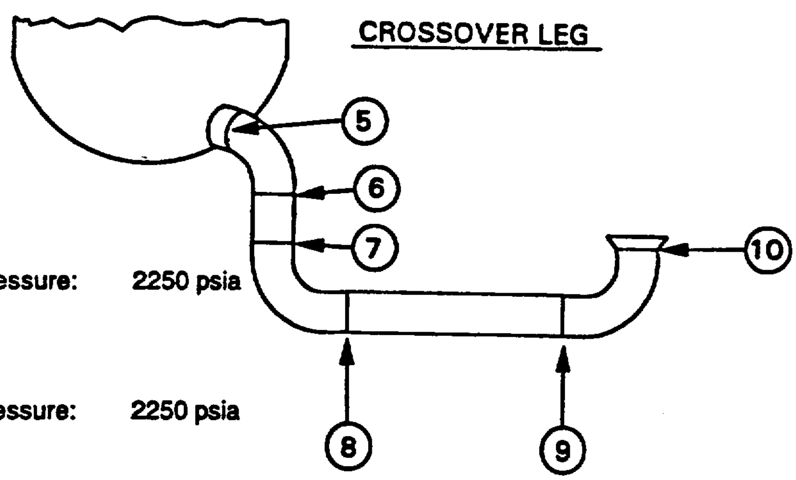
A part of the licensee's analysis concerned the of use of leak-before-break (LBB) technology to remove from consideration the dynamic effects (in this case the acoustic loads on the reactor internals generated by the depressurization associated with a "instantaneous" double-ended guillotine break (DEGB)) of a rupture of the D.C. Cook Unit 1 and 2 main coolant loops (MCLs). The NRC has previously permitted licensees to take credit for LBB piping behavior to address a similar issue, the resolution of Unresolved Safety Issue A-2 on asymmetric LOCA blowdown loads, and the licensee's submittal was consistent with the provision of Title 10 of the Code of Federal Regulations Part 50, Appendix A, General Design Criteria 4, which permits licensees to exclude the dynamic effects associated with postulated pipe ruptures from the facility's licensing basis if "analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping." NRC approval of the licensee's proposed application of LBB would result in the licensee needing only to evaluate the effect on the reactor internals of the acoustic loads developed by the DEGB of reactor coolant system auxiliary lines (along with SSE loads).

The MCLs of D.C. Cook Units 1 and 2 had been previously approved for the application of LBB technology by NRC letter dated, November 22, 1985. The licensee reanalyzed the applicability of LBB to this D.C. Cook Unit 1 and 2 MCL as a result of changes to the D.C. Cook Unit 1 and Unit 2 reactor coolant systems due to steam generator replacement (SG) activities. The following sections address the LBB review.

2.1.1 Identification of Analyzed Piping and Piping Material Properties

The licensee's submittal identified and analyzed the following sections of high energy piping for LBB behavior verification. For each D.C. Cook Unit 1 and 2 MCL, the submittal addressed the piping from the reactor vessel to the SG (the hot leg), from the SG to the reactor coolant pumps (RCPs) (the crossover leg) and the piping from the RCPs to the pressure vessel (the cold leg). Fourteen separate locations were analyzed around on loop of the piping for each unit, and these locations are identified in Figure 1.





HOT LEG

Temperature 620°F, Pressure: 2250 psia

CROSSOVER LEG

Temperature 548°F, Pressure: 2250 psia

COLD LEG

Temperature 548°F, Pressure: 2250 psia

FIGURE 1 SCHEMATIC DIAGRAM OF D. C. COOK UNITS 1&2 PRIMARY LOOP SHOWING WELD LOCATIONS

The analyzed piping was identified as having the following material components. The main piping sections were manufactured from American Society for Mechanical Engineers (ASME) SA-351, Grade CF8M molybdenum-bearing cast stainless steel (CSS). The welds were identified as being stainless steel shielded metal arc welds (SMAWs).

For the material properties used in the LBB analysis, the licensee's analysis used Certified Materials Test Report (CMTR) data for the tensile properties and adjusted the tensile data to the temperature required for the analysis by interpolating between the ASME Code tensile properties and applying an equivalent ratio to the actual tensile property data. For determining the fracture toughness properties of the CSS pieces, the licensee's analysis evaluated the effect of thermal aging, as required by NRC staff guidance on LBB evaluations.

2.1.2 General Aspects of the Licensee's LBB Analysis

In this analysis, the licensee sought to reaffirm the LBB behavior of the subject piping considering changes made to the piping and supports as a result of SG replacement activities. As such, the analysis directly examined the impact of the recalculated piping loads during normal operation (NOP) and SSE conditions on the critical flaw margin and leakage flaw stability criteria. The licensee's analysis made use of the Westinghouse proprietary evaluation codes for assessing the fracture mechanics behavior of the leakage flaw and critical flaw. A brief non-proprietary overview of the analysis and results is provided below.

2.1.3 Licensee Evaluation of the Main Coolant Loop Piping

The licensee's analysis in WCAP-15131, Revision 1, was initiated by an evaluation to determine if any atypical loading condition or degradation mechanism exists which could invalidate the assumptions of the LBB analysis. This evaluation included a review of pressurized water reactor operating history and the potential for water hammer events, intergranular stress corrosion cracking (IGSCC), stress corrosion cracking (SCC), and/or low or high cycle fatigue of the MCL piping. The licensee concluded that IGSCC and SCC were extremely unlikely based upon primary water chemistry control and monitoring. Water hammer events are unlikely due to pressurized-water reactor (PWR) operational characteristics which preclude voiding in these normally filled lines. Finally, low and high cycle fatigue is addressed by piping designs to meet ASME Code requirements and vibrational monitoring systems. In summary, the licensee determined that these loading conditions and degradation mechanisms had an extremely low probability of occurrence which did not affect the ability of the subject piping to be qualified for LBB.

Next, the licensee determined the appropriate loading conditions to be used for the analysis. The licensee calculated the total piping stress at the fourteen locations identified in Figure 1 from the algebraic sum of the forces and moments due to deadweight, thermal expansion, and pressure loads during normal 100 percent power operation. These loads, herein called the "NOP" loads, along with the corresponding pipe dimensions at the fourteen locations, are given in Table 1. The licensee then calculated the total piping stress at the fourteen locations from the absolute sum of the forces and moments due to deadweight, thermal expansion, pressure loads during normal 100 percent power operation, and SSE inertial and anchor motions. These loads, herein called the "NOP+SSE" loads, for each location are given in Table 2. Based on an evaluation of these loads, the licensee determined that the limiting locations for the LBB analysis would be locations 1, 10 , and 11, as shown in Figure 1.

Then the leakage flaw size at each of these limiting locations was determined. The leakage flaw size was determined by applying the NOP loads to a postulated through-wall flaw and determining what size flaw would provide 10 times the leakage detectable by the D.C. Cook containment leakage monitoring system. This safety factor of 10 on the detectable leakage is included in the NRC guidance on LBB evaluations to account for uncertainties in the thermohydraulic calculations for the fluid flow through the crack. Since the D.C. Cook containment leakage monitoring system is capable of detecting 1 gallon per minute (gpm) of leakage (in the course of 4 hours), the leakage size flaw is 10 gpm. The leakage size flaw for locations 1, 10, and 11 are shown in column 2 of Table 3.

The licensee then determined the critical flaw size at each location. The NOP+SSE loads were applied to a postulated through-wall flaw, and the minimum flaw size which failed under the NOP+SSE loads was defined as the critical flaw size. In order to demonstrate that this piping met the margins on flaw size required by NUREG-1061, Volume 3, and DSRP 3.6.3, the critical flaw size at each location must be twice the length of the leakage flaw size. The licensee's analysis demonstrated this in two ways. First, the critical flaw size was determined by using a limit load analysis methodology. This assumes that the piping fails by plastic collapse when the net section of the piping as a stress level equals the flow stress of the material. The second method to demonstrate that a margin of two on flaw size was achieved involved an analysis using a J-integral approach to assessing fracture behavior. For this analysis, the licensee analyzed a flaw equivalent to twice the length of the leakage size flaw and demonstrated that it did not propagate unstably to failure under NOP+SSE loadings. This ensured that a margin of two was achieved without directly determining the critical flaw size. The results of the licensee's analysis are given in columns 3 and 4 of Table 3.

A final criteria that must be evaluated to demonstrate the LBB qualification of the piping is to show that the leakage size flaw is stable under loads potentially greater than the NOP+SSE load combination. If the NOP+SSE loads are summed algebraically, then they should be multiplied my a factor of $\sqrt{2}$ and the leakage flaw should still be found to be stable. In this evaluation, since the licensee chose to sum the NOP+SSE loads absolutely, no additional multiplier is required. Therefore, its demonstration that a flaw twice the size of the leakage flaw (i.e. the critical flaw) was stable also demonstrates that the leakage flaw will be stable under these loads.

2.1.4 <u>Leak Before Break Staff Summary</u>

Based on the information provided by the licensee regarding the materials comprising the D.C. Cook Unit 1 and 2 MCL piping and the loads under NOP and SSE conditions, the staff independently assessed the compliance of this system with the LBB criteria established in NUREG-1061, Volume 3. The staff has concluded that the analyses submitted by the licensee, along with additional information submitted regarding the torsional moments at each analysis location, were sufficient to demonstrate that LBB behavior would be expected from the subject piping following the installation of the replacement SGs. The staff's evaluation, which follows the guidance of NUREG-1061, Volume 3, is provided below.

2.1.5 Identification of Analyzed Piping and Piping Material Properties

The staff examined the list of materials identified for the MCL piping and concluded that it would be necessary to evaluate the material properties of both the CSS piping segments and the associated SMAWs at the limiting locations because of their susceptibility to thermal aging. NUREG-1061, Volume 3, specifies particular aspects which should be considered when developing materials property data for LBB analyses. First, data from the testing of the plant-specific piping materials is preferred. However, in the absence of such data, more generic data from the testing of samples having the same material specification may be used. More specifically, it was noted in Appendix A of the NUREG that "[m]aterial resistance to ductile crack extension should be based on a reasonable lower-bound estimate of the material's J-resistance curve," while section 5.2 of the NUREG stated that the materials data should include,"appropriate toughness and tensile data, long-term effects such as thermal aging and other limitations."

The staff noted that although tensile test data had been provided by the licensee for the cast stainless steel heats in the D.C. Cook Unit 1 and 2 MCLs, this data did not account for thermal aging effects. Likewise, no heat-specific fracture toughness data for the D.C. Cook materials was provided in the aged condition. The staff's evaluation assumed that conservatively high amounts of δ-ferrite were present (greater than 15 percent, based on the highest value cited for a specific heat, 39344-2, of 22.92 percent) in the CSS pieces at the limiting locations. Results from work at Argonne National Laboratory (References 1 and 2), sponsored by the NRC, were used as the basis for developing generic J-R and stress-strain curves for the CSS material. The CSS material properties parameters used for the staff's evaluation are given in Table 5. Materials property parameters for the evaluation of the aged stainless steel SMAWs were also developed based on work by Argonne National Laboratory (Reference 3) and are given in Table 6. These generic material properties representations for CSS and SMAWs are consistent with those chosen by the staff in previous LBB reviews.

2.1.6 General Aspects of the Staff's LBB Analysis

The staff's analysis was performed in accordance with the guidance provided in NUREG-1061, Volume 3. Based on the information submitted by the licensee, the staff determined the critical flaw size at the bounding location for the MCL using the codes compiled in the NRC's Pipe Fracture Encyclopedia (Reference 4). For the purposes of the staff's evaluation, the critical location was defined by those locations at which materials with low postulated fracture toughness existed in combination with high ratios of SSE-to-NOP stresses. This was because high SSE stresses tend to reduce the allowable critical flaw size while low NOP stresses increase the size of the leakage flaw required to produce 10 gpm of leakage. In particular, when evaluating the critical flaw in thermally-aged CSS base materials, the staff used the LBB.ENG2 code developed by Brust and Gilles (Reference 5). When evaluating SS SMAWs, the staff used the LBB.ENG3 code developed by Battelle (Reference 5) for the express purpose to determine if a substantial difference in the tensile properties of the weld and base metal was expected.

The staff then compared the critical flaw at the bounding location to the leakage flaw which provided 10 gpm of leakage under NOP conditions to determine whether the margin of 2 defined in NUREG-1061, Volume 3, was achieved. The leakage flaw size calculation was carried out using the Pipe Crack Evaluation Program (PICEP, Revision 1) analytic code developed by the Electric Power Research Institute. The 10 gpm value was defined by noting that the D.C. Cook Unit 1 and 2 containment leakage detection systems would be able to detect a 1 gpm leak in the course of one hour and a factor of 10 is applied to this 1 gpm detection capability to account for thermohydraulic uncertainties in calculating the leakage through small cracks.

2.1.7 Staff Evaluation of the D.C. Cook Unit 1 and 2 Main Coolant Loop

First, the staff examined the licensee's evaluation regarding atypical loading conditions or degradation mechanisms which could invalidate the assumptions of the LBB analysis. The staff concurred that the evaluation of pressurized water reactor operating history and the potential for water hammer events, intergranular stress corrosion cracking (IGSCC), stress corrosion cracking (SCC), and/or low or high cycle fatigue of the MCL piping was appropriate. The staff agreed with the licensee's conclusion that IGSCC and SCC were extremely unlikely based upon primary water chemistry control and monitoring. The staff also agreed that water hammer events are unlikely due to PWR operational characteristics and that low and high cycle fatigue is addressed by piping designs that meet ASME Code requirements and vibrational monitoring systems. In summary, the staff concurred with licensee's determination that these loading conditions and degradation mechanisms had the extremely low probability of occurrence which did not affect the ability of the subject piping to be qualified for LBB.

The staff's evaluation then examined the loadings submitted by the licensee. It was noted that the summation methodology utilized by the licensee was not completely consistent with the guidance provided by the staff in NUREG-1061 Volume 3 or Draft Standard Review Plan (DSRP) 3.6.3 for determining loads for LBB analyses. The inconsistency in the licensee's analysis was that the licensee did not include the torsional moments (as directed to in NUREG-1061 Volume 3). However, the licensee subsequently provided those moment components for each piping location in a letter dated November 19, 1999, so that information was available for the staff's evaluation. Based on the staff's evaluation of the loadings supplied by the licensee, the staff concluded that the limiting locations for the MCL piping evaluation would be location 1 (at the hot leg nozzle connection to the reactor vessel), location 5 (at the SG outlet nozzle), location 10 (at suction nozzle of the RCP) and location 11 (at the discharge nozzle of the RCP) as shown in Figure 1.

At each location, the staff evaluated the critical and leakage flaw sizes for the CSS material and the associated SMAW weld. The material properties assumed by the staff for the CSS and SMAW materials are shown in Tables 4 and 5 and the loads used in the staff's leakage flaw and critical flaw analysis (which include the torsional moments) are shown in Table 6. The staff applied the PICEP code using the nominal piping dimensions and a crack surface roughness of $\varepsilon = 0.0003$ inches. This procedure calculated a 10 gpm leakage flaw size for each material. The critical flaw size determined by using the LBB.ENG2 or LBB.ENG3 code as appropriate. The ratio of the critical-to-leakage-flaw size was then determined for comparison to the recommended margin of 2 in NUREG-1061, Volume 3. These results are summarized in Table 7. Since the margin of 2 on the crack sizes was achieved for each location, the leakage flaw was also shown to be stable given the absolute summation of the NOP and SSE loads (as calculated by the licensee plus the torsional loads included in the staff's analysis), and meets the margin on loading recommended by NUREG-1061, Volume 3.

2.1.8 Main Coolant Loop Break Summary

Based on the information and analysis supplied by the licensee, the staff was able to independently assess the LBB status of the D.C. Cook Unit 1 and 2 MCL piping. The staff has concluded that it has been demonstrated that the LBB behavior of the MCL is covered by the analysis submitted by the licensee and the independent evaluation by the staff presented in this SE. Furthermore, the licensee should be permitted to credit this conclusion for eliminating the dynamic effects associated with the postulated rupture of these sections of piping from the D.C. Cook Unit 1 and 2 facility licensing basis, consistent with the provisions of 10 CFR 50, Appendix A, General Design Criteria 4, including credit for removing from consideration the acoustic loads that would be generated by a DEGB of the MCL when evaluating the ability to insert the RCCAs.

Table 4: Parameters used in Staff Evaluation of Aged D.C. Cook CSS Piping

Parameter	Value
Young's Modulus	25500 ksi
Yield Strength	32.8 ksi
Ultimate Tensile Strength	78.8 ksi
Sigma-zero	32.2 ksi
Epsilon-zero	0.00129
Ramberg-Osgood Alpha	1.276
Ramberg-Osgood n	6.6
C	2599 in-lb / in^2
n	0.31

Note: $J = C(\Delta a)^n$

Table 5: Parameters used in Staff Evaluation of Aged D.C. Cook SS Piping Welds

Parameter	Value
Young's Modulus	25000 ksi
Yield Strength	49.4 ksi
Ultimate Tensile Strength	61.4 ksi
Sigma-zero	35.0 ksi
Epsilon-zero	0.00125
Ramberg-Osgood Alpha	9.0
Ramberg-Osgood n	9.8
A	228 in-lbs./in^2
C	476 in-lbs./in^2
n	0.643

Note: $J = A + C(\Delta a)^n$

Table 6: Loads Used in the Staff's Evaluation of Locations 1, 5, 10, and 11

	Location 1	Location 5	Location 10	Location 11
Normal Ops. Axial (Including Pressure)	1529 kips	1664 kips	1796 kips	1372 kips
Normal Ops. Bending	28495 in-kips	4557 in-kips	7313 in-kips	5116 in-kips
NOP + SSE Axial (Including Pressure)	1766 kips	1891 kips	1866 kips	1492 kips
NOP + SSE Bending	30033 in-kips	10790 in-kips	16837 in-kips	14347 in-kips

Table 7: Results of the Staff's Evaluation for Locations 1, 5, 10, and 11

Location	Leakage Flaw Size	Critical Flaw Size	Margin
1 - CSS	5.8 inches	28.9 inches	5.0
1 - SMAW	5.9 inches	30 inches	5.1
5 - CSS	9.8 inches	> 42 inches	> 4.3
5 - SMAW	9.9 inches	> 45 inches	> 4.5
10 - CSS	8.9 inches	> 42 inches	> 4.7
10 -SMAW	9.0 inches	> 45 inches	> 5
11 - CSS	8.1 inches	> 35 inches	> 4.3
11 - SMAW	8.2 inches	> 37 inches	> 4.5

2.2 OTHER RCS BREAK ANALYSIS

The relevant pipe breaks in the branch lines that were considered in the analysis are the 60-inch^2 accumulator line break and the 98-inch^2 pressurizer surge line break. These breaks are not covered by the application of the LBB technology at D. C. Cook. The licensee used the MULTIFLEX 3.0 computer code to calculate the blowdown loads on the reactor vessel and the reactor vessel internals, including the guide tubes and core barrel. MULTIFLEX has previously been used by Westinghouse to calculate blowdown loads. The version of MULTIFLEX used for this analysis is considered by Westinghouse as an improved version that was developed specifically for the Westinghouse Owner's Group Baffle Barrel Bolt Program (BBBP). Westinghouse stated that previous BBBP analyses performed using this version of the code were accepted by the NRC.

The NRC did not perform a detailed review of the MULTIFLEX 3.0 Code, but found the general methodology reasonable and acceptable. A conservative and previously accepted 1 millisecond break opening time was assumed. Therefore, these calculations of LOCA blowdown loads are judged to be conservative. Also, the blowdown loadings have been determined using a methodology which has been previously accepted by the NRC.

2.3 CONTROL ROD INSERTION

The ability to insert the control rods is a function of the guide tube's deflection during a LOCA transient. As the amount of deflection increases, control rod insertion time will first increase due to increased resistance and at sufficient deflection, control rod insertion will be precluded. Since the guide tube is a complex structure, and the motion of control rods are dependent on the amount of friction between the two components, it is difficult to determine control rod insertion through analytical means. For this reason, guide tube scram tests have been performed by Westinghouse in the past to experimentally determine the limits of control rod insertability. Guide tube scram tests have been performed on 96"-17x17, 150"-15x15 guide tubes, (References 6 and 7). Full-size guide tubes, with rod control clusters, were mechanically loaded at discrete elevations to simulate flow loads experienced during a postulated LOCA transient. The insertion for the control rods as a function of the guide tube deflection, which in turn is a function of the applied mechanical loads, were recorded during the tests. The allowable load is then determined as the limiting applied mechanical load corresponding to the guide tube's permanent loss of function. Westinghouse determined the total LOCA loads by combining the inertial acceleration and acoustic loads calculated by MULTIFLEX with the hydraulic cross flow loads, i.e., drag loads, which were estimated based upon scale model tests and plant strain measurements, together with information from the MULTIFLEX and other hydraulic calculations. A dynamic load factor was applied to account for the transient nature of the drag loading. This total LOCA load was added using the square root sum of squares (SRSS) method to the peak safe shutdown earthquake (seismic) load to obtain the total load. The staff finds the methodology used by Westinghouse to calculate the combined peak guide tube loading to be reasonable and consistent with industry practice. In most respects, this methodology is similar to NRC-accepted methodology for assuring control rod insertion during faulted LOCA and seismic conditions in other applications (Reference 8). Therefore, the staff finds it acceptable.

Westinghouse compared the calculated combined peak loads to the allowable values. Due to the differences in fuel assemblies between Unit 1 (15 x 15) and Unit 2 (17 x 17), the allowable loads and the peak combined loads differ between the units. For both units, the calculated peak combined load showed considerable margin to the allowable. Therefore, the maximum guide tube deflection which occurs under the limiting analyzed break size noted above plus deflection from seismic loading will not prevent the control rods from inserting.

2.4 FUEL ASSEMBLY GRID LOADING CONSIDERATIONS

The general analytical procedure for evaluating fuel assembly transient response to seismic and LOCA transients was provided schematically by the licensee, outlining the main steps in the analytical sequence. Forcing functions for the reactor internals model are based on postulated LOCA and seismic conditions. The hydraulic forces and loop mechanical loads resulting from a postulated LOCA pipe rupture are prescribed at appropriate locations of the reactor pressure vessel (RPV) model. For the seismic analysis, the plant-specific design acceleration spectra are specified, based upon the plant site characteristics. For the current analysis, the synthesized seismic time histories are calculated from the D. C. Cook plant-specific acceleration response spectra envelope. These spectra are for the containment buildings at the appropriate elevation and the design-basis damping. Both the LOCA and seismic time histories are applied to the RPV system model. The core plate motions from the dynamic analysis of this model are obtained and are then input to the reactor core model.

The limiting LOCA and seismic grid impact loads for 15x15 and 17x17 assembly cores have been summarized. The maximum grid loads obtained from SSE and LOCA loading analyses, were combined as required using the SRSS method. The results of the seismic and LOCA analyses of the maximum impact forces for the 15x15 and 17x17 structural grids are compared to allowable grid distortion loads. These allowable grid loads are experimentally established as the 95-percent confidence level on the mean from the distribution of grid distortion data at normal plant operating temperature. Acceptability of the fuel assembly grid performance for RCCAs control rod insertion is verified by demonstrating that no grid deformation occurs in assemblies directly beneath control rod locations. For both Units 1 and 2, no fuel assembly grid distortion was calculated and thus control rod insertion will not be impeded under limiting break plus seismic loadings.

2.5 REACTIVITY CONTROL

The negative reactivity associated with the RCCAs being available will provide adequate negative reactivity to ensure that following a design-basis LOCA, the realignment from a cold leg recirculation configuration to a hot leg recirculation configuration will not result in core re-criticality. The amount of negative reactivity available is verified every fuel cycle and is shown to be sufficient to prevent re-criticality.

3.0 SUMMARY

Based on the review of the structural analysis methodology and results as discussed above, the staff finds that, for both Units 1 and 2, the maximum fuel assembly guide tube deflections which occur during limiting LOCA breaks plus seismic loadings will not prevent the control rod from inserting. In addition, no fuel assembly grid distortion will occur and thus control rod insertion will not be impeded for these loads. Therefore the staff finds that it is acceptable for the licensee to revise the UFSAR and EOPs to allow credit for the negative reactivity provided by the insertion of the rod cluster control assemblies into the reactor core following a design basis LOCA.

4.0 REFERENCES

1. Chopra, O.K., "Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems," NUREG/CR-4513, ANL-93/22, Rev.1.

2. Michaud, W.F., et al., "Tensile-Property Characterization of Thermally Aged Cast Stainless Steels," NUREG/CR-6142, ANL-93/35.

3. Gavenda, D.J., et al., "Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds," NUREG/CR-6428, ANL-95/47.

4. Pipe Fracture Encyclopedia, produced on CD-ROM by Battelle-Columbus Laboratory for the U.S. Nuclear Regulatory Commission, 1997.

5. Brust, F.W., et. al., "Assessment of Short Through-Wall Circumferential Cracks in Pipes," NUREG/CR-6235, BMI-2179.

6. "Scram Deflection Test Report 17x17 Guide Tubes, 96 Inch and 150 Inch," WCAP 9251, December 1977

7. "Summary Report on PGE Scrammability Test," CE-RD-233, February 17, 1969

8. D. Beaumont, et. al, "Verification Testing and Analysis of the 17x17 Optimized Fuel Assembly," WCAP 9401-P-A, August 1981, Westinghouse Report

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (64 FR 56531). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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