November 8, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - REVIEW OF LEAK-BEFORE-BREAK FOR THE PRESSURIZER SURGE LINE PIPING AS PROVIDED BY 10 CFR PART 50, APPENDIX A, GDC 4 (TAC NOS. MA7834 AND MA7835)

Dear Mr. Powers:

By letters dated August 22, 2000, as supplemented September 5, October 7, and October 26, 2000, you submitted a request for the Nuclear Regulatory Commission (NRC) to review and approve the leak-before-break (LBB) evaluation for the Donald C. Cook Nuclear Power Plant, Units 1 and 2, pressurizer surge line piping. The submittal was made in accordance with the provisions 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." LBB evaluations utilizing the guidance of NRC NUREG-1061, Volume 3, have been previously approved by the NRC staff as a method for making such a demonstration.

The NRC staff has completed its evaluation of the Indiana Michigan Power Company (I&M) (the licensee) submittal. The information provided by the licensee in its original submittal and supplemented by information in follow up letters on September 5, October 7, and October 26, 2000, was sufficient to permit the NRC staff to independently evaluate the licensee's conclusions. While the detailed results of the NRC staff's evaluation do differ with I&M's, the NRC staff agrees with the licensee's conclusion that LBB behavior has been demonstrated for the pressurizer surge line piping. It should be noted that the NRC staff's conclusion was predicated on I&M's demonstration that the Donald C. Cook units' leakage detection system inside of containment was capable of reliably detecting 0.8 gallons per minute of primary system leakage.

Therefore, the NRC staff finds that I&M may remove consideration of the dynamic effects associated with the postulated rupture of the pressurizer surge line piping from the licensing basis of Units 1 and 2 of the Donald C. Cook Nuclear Power Plant. The safety evaluation which addresses the technical basis for the NRC staff's finding is enclosed. This action closes restart action matrix item 8.4.

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos.: 50-315, 50-316

Enclosure: As stated

Therefore, the NRC staff finds that I&M may remove consideration of the dynamic effects associated with the postulated rupture of the pressurizer surge line piping from the licensing basis of Units 1 and 2 of the Donald C. Cook Nuclear Power Plant. The safety evaluation which addresses the technical basis for the NRC staff's finding is enclosed. This action closes restart action matrix item 8.4.

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Donald C. Cook Nuclear Plant, Units 1 and 2

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST TO APPLY LEAK-BEFORE-BREAK STATUS TO THE

PRESSURIZER SURGE LINE PIPING AT

DONALD C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2

INDIANA MICHIGAN POWER COMPANY

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated August 22, 2000, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Power Plant, Units 1 and 2, requested that the Nuclear Regulatory Commission (NRC) review and approve their application to remove consideration of the dynamic effects of postulated ruptures of the D. C. Cook, Units 1 and 2 pressurizer surge line piping from the facility's licensing basis. I&M's submittal was based on an application of Title 10 of the Code of Federal Regulations Part 50, Appendix A, General Design Criteria (GDC) 4, which states:

> However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when 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.

For the purposes of this demonstration, I&M submitted a leak-before-break (LBB) analysis prepared by Westinghouse for the subject portions of the pressurizer surge line piping. LBB evaluations developed using the analysis methodology contained in NRC NUREG-1061, Volume 3[1] have been previously approved by the Commission as demonstration of an extremely low probability of piping system rupture.

2.0 REGULATORY REQUIREMENTS AND STAFF POSITIONS

Nuclear power plant licensees have, in general, been required to consider the dynamic effects which could result from the rupture of sections of high energy piping (fluid systems that during normal plant operations are at a maximum operating temperature in excess of 200 °F and/or a maximum operating pressure in excess of 275 psig). This requirement has been formally included in 10 CFR 50, Appendix A, General Design Criteria 4 which states, "structures, systems, and components important to safety....shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit."

As noted in Section 1.0 above, the NRC modified GDC 4 to permit the dynamic effects of some high energy piping ruptures to be excluded from facility licensing bases based upon the demonstration of an extremely low probability of piping system rupture. Consistent with this modification to GDC 4, the NRC accepted the “leak-before-break” analysis methodology as an acceptable means by which this extremely low probability of piping system rupture could be demonstrated. The philosophy of "leak-before-break" behavior for high energy piping systems was developed by the NRC in the early 1980s, used in certain evaluations stemming from Unresolved Safety Issue A-2, "Asymmetric Blowdown Loads on PWR Primary Systems," and then subsequently expanded for application toward resolving issues regarding defined dynamic effects from high energy piping system ruptures. The methodology developed by the NRC for performing LBB analyses was detailed in NUREG-1061, Volume 3 which was published in November 1984.

3.0 LICENSEE'S DETERMINATION

The following discussion contains information supplied by I&M in its August 22, 2000, letter forwarded to the NRC and the enclosures to that letter. These enclosures included the proprietary report prepared by Westinghouse for I&M: WCAP-15434, Revision 1, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for D. C. Cook, Units 1 and 2 Nuclear Power Plants.” The following discussion also includes information provided in the licensee’s follow up letters, dated September 5, October 7, and October 26, 2000. The licensee’s September 5, 2000, follow up letter provided information to support conclusions drawn in the original submittal and provide extended analyses. The licensee’s October 7 and October 26, 2000, follow up letters provided information to demonstrate the facilities’ leakage detection system inside of containment was capable of reliably detecting 0.8 gallons per minute (gpm) of primary system leakage.

3.1 Identification of Analyzed Piping and Piping Material Properties

I&M's submittal identified and analyzed the following sections of piping for LBB behavior verification. I&M addressed the pressurizer surge line for each unit from its connection to the reactor coolant system hot leg to the pressurizer as shown in Figure 3-1 of WCAP-15434, Revision 1.

The pressurizer surge line piping was identified as having the following material components. The piping and fittings of the D.C. Cook, Units 1 and 2 pressurizer surge lines were manufactured from wrought American Society for Mechanical Engineers (ASME) specification SA-376 Type 316 stainless steel (SS). The welds in this system were identified as having been fabricated from SS using gas tungsten arc and shielded metal arc welding (SMAW) processes. The line was manufactured from 14-inch nominal diameter, Schedule 160 (nominal wall thickness, 1.406 inches) pipe.

For the material properties used in the pressurizer surge line LBB evaluations, I&M/Westinghouse used minimum and average room temperature tensile properties based on Certified Materials Test Report data. The minimum and average tensile properties at temperatures of interest (205 °F, 617 °F, and 653 °F) were calculated using the ratio of the ASME Code, Section III properties at room temperature to the Code properties at the

temperatures of interest to scale the Certified Materials Test Report (CMTR)-based data. The modulus of elasticity variation with temperature was established based on ASME Code, Section III values. The minimum tensile properties were used by I&M/Westinghouse in the LBB critical flaw size determination, while the average tensile properties were used in the LBB leakage flaw size determination.

3.2 General Aspects of the Licensee's LBB Analysis

The analyses provided by I&M/Westinghouse sought to address four principal areas which were consistent with the criteria established for LBB analysis acceptability in NUREG-1061, Volume 3. One, demonstrate that the subject piping is a candidate for LBB analysis by showing that the piping is not particularly susceptible to active degradation mechanisms or atypical loading events. Two, establish the critical through-wall flaw size under which analyzed locations would be expected to fail under normal operating procedure (NOP) plus safe-shutdown earthquake (SSE) or startup/shutdown loading conditions. Three, establish the leakage behavior of smaller through-wall flaws under NOP loads alone for each location. Four, evaluate the margin between the critical through-wall flaw size and an appropriate leakage through-wall flaw size and the stability of the through-wall leakage flaw.

3.3 Evaluation of Pressurizer Surge Line Piping

The analysis of the pressurizer surge line piping that was submitted to the NRC staff as an enclosure to the licensee’s August 22, 2000, letter was prepared for the licensee by Westinghouse as report number WCAP-15434, Revision 1. Additional analyses prepared by Westinghouse were submitted in the licensee’s September 5, 2000, follow up letter. This section summarizes the results of the I&M/Westinghouse results for the four subject areas noted in Section 3.2 above.

Initially, the I&M submittal addressed the issue of potential piping degradation mechanisms and atypical loading conditions. Per the discussion of the limitations of LBB analyses in NUREG-1061, Volume 3, the LBB approach should not be considered when operating experience has indicated particular susceptibility to failure from the effects of corrosion, water hammer, or fatigue. Such mechanisms could cause the development of complex or extensive flaws in piping which significantly degrade its load carrying capacity while not propagating through-wall over a sufficient length to be detectable, or provide loads which are difficult to bound analytically. I&M’s submittal concluded that pressurized-water reactor pressurizer surge line piping, like that at D.C. Cook, Units 1 and 2, has not been shown to be particularly susceptible to the effects of water hammer, intergranular stress corrosion cracking, or erosion-corrosion.

Regarding the potential for fatigue cracking from mechanical and thermal loadings, I&M/Westinghouse noted that low cycle fatigue considerations were accounted for in the design of this piping system through the fatigue usage factor evaluation to show compliance with the rules of the ASME Code, Section III (also in accordance with analyses in response to NRC Bulletin 88-11). Additionally, I&M/Westinghouse provided an analysis of the growth of postulated surface flaws based on design transient loading conditions and the analysis procedure suggested by the ASME Code, Section XI, Appendix A. I&M/Westinghouse showed that for semi-elliptic surface flaws with initial depths of up to one-tenth of the thickness of the

pipe wall, little or no growth was expected to occur. High cycle fatigue loads, primarily from pump vibrations, are managed through the monitoring of reactor coolant pump shaft vibration limits and inservice measurements have shown that the magnitude of the stresses associated with these vibrations is low and is not expected to raise a concern through the operating life of the facilities.

Next, the I&M/Westinghouse analysis evaluated the pressurizer surge line piping by developing the applied stresses under NOP, NOP plus SSE, forced cooldown, and the low probability event of startup/shutdown plus SSE loading conditions and determined the leakage and critical through-wall flaw size for various locations along the piping. In the determination of the NOP applied stresses, the analysis included the tensile and bending stresses resulting from the internal pressure, deadweight, and thermal expansion. The licensee also provided another set of NOP loads which also included the contribution of the thermal stratification stresses at NOP conditions. SSE loads were added to the NOP loads (with and without NOP thermal stratification loads) when determining loads to be considered for the critical flaw size evaluation. In addition, forced cooldown and startup/shutdown plus SSE loading conditions were developed separately and are of interest for evaluating the allowable critical flaw size since large thermal bending stresses due to thermal stratification are developed in this piping as a result of the temperature difference between the hot leg and pressurizer during these evolutions. It should, however, be noted that in the original submittal, load combinations in WCAP-15434, Revision 1 did not account for torsional moments on the pressurizer surge line piping for any of the loading conditions mentioned above. Loads that were provided in Enclosure 4 to the licensee's August 22, 2000, submittal and the loads submitted as part of the extended analyses in the licensee's follow up letter dated September 5, 2000, did include torsional moments as requested by the NRC staff.

In the load combination, the deadweight, thermal expansion and/or thermal stratification, pressure, and SSE stresses were summed absolutely for the critical flaw size determination. Likewise, when evaluating the loads for the startup/shutdown conditions (with or without the SSE), the deadweight, thermal expansion, thermal stratification, and pressure loads were summed absolutely. The deadweight, thermal expansion and/or thermal stratification, and pressure stresses for NOP conditions were summed algebraically for the leakage flaw size determination. Table 1a summarizes the significant load combination results provided by I&M/Westinghouse in WCAP-15434, Revision 1. Table 1b shows the load combinations of interest submitted by the licensee which include torsional moments.

For the purposes of LBB analyses, the critical flaw size can be defined as the longest preexisting through-wall flaw which could exist without growing unstably to double-ended pipe rupture under faulted or off-normal loading conditions. As explained above, for the pressurizer surge line piping this includes considering the NOP plus SSE and forced cooldown loading conditions. The analysis performed by Westinghouse to establish the critical flaw size at a nodal location was based on the use of a limit load analysis approach. This approach effectively predicts piping failure based on net section collapse of the cross-section which has been reduced by the through-wall cracked section. In the I&M/Westinghouse analysis of the pressurizer surge lines, the stainless steal (SS) welds were identified as the limiting material, i.e., the material for which the smallest margin between the critical and leakage flaw size exists. When analyzing SS welds using a limit load-based approach, an additional factor, the Z-factor, was incorporated to account for the generally lower toughness and lower load carrying capacity of SMAW welds. The I&M/Westinghouse analysis applied the Z-factor to increase the applied

loads and, thus, reduce the through-wall flaw size which could be withstood without piping failure.

The leakage flaw size for an LBB analysis is defined as the flaw size which, under NOP conditions, would leak 10 times the amount of fluid detectable by the facility's containment leakage detection system. The factor of 10 is established in the LBB guidance of NUREG-1061, Volume 3 as the safety factor on leakage to account for uncertainties in calculating leakage from a through-wall crack. As noted in Section 5.2.3 of WCAP-15434, Revision 1, the performance of the D.C. Cook containment leakage detection system is consistent with the guidelines of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," and is capable of detecting a gpm leak in one hour. Therefore, the leakage flaw calculated by I&M/Westinghouse at each nodal location was based on a leak rate of 10 gpm under NOP conditions. The leakage analysis performed by Westinghouse was based on the use of a Westinghouse proprietary methodology for calculating single or two-phase flow through cracks in light-water reactor piping. As will be discussed in Section 4.0, based on preliminary results from the NRC staff's evaluation of WCAP-15434, Revision 1, the licensee provided additional information in their October 7 and October 26, 2000, follow up letters to demonstrate that this leakage detection system was, in fact, capable of detecting 0.8 gpm of primary system leakage.

In WCAP-15434, Revision 1, I&M/Westinghouse identified the limiting location to be node 1020 in the D.C. Cook, Unit 2 surge line piping. In their September 5, 2000, letter, I&M/Westinghouse extended their analyses to include torsional loads for all analyzed nodes and also evaluated D.C. Cook, Unit 2 pressurizer surge line piping nodes 1030 and 1060. The results of I&M/Westinghouse's extended analyses are discussed below.

Based on the results of the analyses documented in their September 5, 2000, letter, I&M/Westinghouse again concluded that node 1020 for D.C. Cook, Unit 2 would be the limiting node for the LBB analysis. For node 1020, I&M/Westinghouse determined that the leakage flaw size which would provide 10 gpm of leakage at NOP conditions would be 6.30 inches if NOP thermal stratification loads were not considered, and 6.28 inches if NOP thermal stratification loads were included in the analysis. For the forced cooldown and NOP plus SSE loading conditions, the critical flaw sizes ranged from 15.70 to 16.12 inches. Hence, for any combination of leakage flaw size and critical flaw size, I&M/Westinghouse concluded that the margin on flaw size would be at least 2.5. For the low probability event of an SSE during startup/shutdown, the critical flaw size was likewise determined to be 12.40 inches, providing a margin of (12.40/6.28) = 1.97, which they rounded off to a value of 2. The licensee, therefore, concluded that the NUREG-1061, Volume 3, recommended that a margin of 2 on length between the leakage and critical flaws sizes had been adequately demonstrated. In addition, since the load summation for the critical flaw size analysis was done on an absolute basis, the stability of the leakage size flaw under forced cooldown or NOP plus SSE conditions with a safety factor on the loads of unity was also demonstrated.

4.0 STAFF EVALUATION

Based on the information provided by the licensee regarding the materials comprising the D.C. Cook, Units 1 and 2 pressurizer surge line piping and the loads under NOP, NOP plus SSE, and forced cooldown conditions, the NRC staff independently assessed the compliance of these systems with the LBB criteria established in NUREG-1061, Volume 3. The NRC staff has

concluded that the analysis submitted by the licensee, including the additional information supplied in the licensee's follow up letters of September 5, October 7, and October 26, 2000, was sufficient to demonstrate that LBB behavior would be expected from the subject piping. The following sections will focus on the differences between the details of the NRC staff's analysis, conducted per NUREG-1061, Volume 3, and the licensee's evaluation.

4.1 Identification of Analyzed Piping and Piping Material Properties

The NRC staff examined the list of materials identified for the pressurizer surge line piping and concluded that the materials of primary interest for the LBB analysis would be the SS welds because of their susceptibility to thermal aging. However, in evaluating the fracture behavior of the SS welds, the stress-strain properties of the surrounding wrought SS piping would also be used, as addressed below. NUREG-1061, Volume 3, specifies particular aspects which should be considered when developing materials property data for LBB analyses. 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 "material 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."

Given the above, the NRC staff did not concur with the I&M/Westinghouse methodology for evaluating the SS weld materials. Westinghouse's use of a Z-factor modified limit load approach is consistent with guidance in Draft Standard Review Plan, Section 3.6.3[2] (published for comment in 1987) on LBB and the technical bases on which some of the flaw evaluation criteria in ASME, Section XI were developed. However, since the mid-to-late 1980s time frame, additional evidence regarding the effects of thermal aging on SMAW SS pipe welds has been collected. When comparing the J-R data cited as the basis for the flaw evaluation criteria of ASME, Section XI and the Z-factor approach [3,4], it appears that the thermal aging of SS weld materials may not be adequately accounted for. It is the NRC staff's position that an LBB analysis is significantly different from a flaw evaluation and that the thermal aging of SS weld materials must be explicitly addressed. An additional study from Argonne National Laboratory[3] was the NRC staff's reference for this information and the NRC staff's characterization of the J-R curve is given in Table 2. The mean minus one standard deviation lower bound J-R curve used by the NRC staff was actually developed by Wilkowski and Ghaliadi at Battelle Columbus Laboratory as a fit to unaged SS weld data, but the conclusions of Reference 3 noted that there was little observed change in the fracture toughness behavior with thermal aging for those welds that began with inferior fracture toughness properties. The stress-strain properties, developed from Reference 3, for aged SS weld material for this evaluation are also given in Table 2. For the wrought austenitic SS piping, the NRC staff accepted the tensile properties provided by the licensee for use in the NRC staff's analysis.

In addition, the NRC staff did not concur with the original I&M/Westinghouse position in WCAP-15434, Revision 1, to exclude torsional moments in the load summations for determining both the critical and leakage flaw sizes. In a pre-submittal meeting with I&M/Westinghouse on August 15, 2000, the NRC staff noted that the guidance provided in NUREG-1061, Volume 3, and Draft Standard Review Plan 3.6.3 were clear on this subject: in an LBB evaluation, torsional moments shall be included in a square-root-sum-of-the-squares

(SRSS) summation with the other bending moments. While assessment in this manner may be conservative, excluding torsional moments from the analysis outright would certainly be non-conservative. Hence, unless an alternate methodology was provided to "more accurately" assess the impact of torsional moments (and assess the fracture toughness of the subject materials under combined Mode I and Mode II loadings), the SRSS summation is necessary to ensure all loads are adequately accounted.

4.2 General Aspects of the NRC staff's LBB Analysis

The NRC 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 NRC staff determined the critical flaw size at potential bounding locations for each piping system using the codes compiled in the NRC's Pipe Fracture Encyclopedia.[5] For the purposes of the NRC staff's evaluation, the list of potential bounding locations was defined by those locations at which materials with low postulated fracture toughness existed in combination with high ratios of SSE-to-NOP or forced cooldown-to-NOP stresses. This was because high SSE or forced cooldown stresses tend to reduce the allowable critical flaw size, while low NOP stresses increase the size of the leakage flaw. When evaluating pipe welds, the NRC staff used the LBB.ENG3 code developed by Battelle[6] for that express purpose. The LBB.ENG3 methodology is significantly different from the other codes in Reference 5 (and from the licensee's analysis) in that LBB.ENG3 explicitly incorporates a J-R based approach and accounts for the differences in the stress-strain properties of the weld and an adjoining base material when determining the effective energy release from the structure with crack extension. Criteria regarding the applied J exceeding the material J_{IC} and the applied dJ/da exceeding the material's d(J-R)/da were used to identify the critical crack size.

The NRC staff then compared the critical flaw at the bounding location to the leakage flaw which provided ten times the leakage detectable by the D.C. Cook, Units 1 and 2 containment leakage detection systems 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.[7] As discussed in Section 4.4 below, an 8 gpm leakage flaw value was used given: (1) the licensee's demonstration that the D.C. Cook, Units 1 and 2 containment leakage detection systems would be able to reliably detect a 0.8 gpm leak and, (2) the multiplicative factor of 10 applied to account for thermohydraulic uncertainties in calculating the leakage through small cracks. The stability of the leakage flaw under NOP plus SSE loads was subsequently evaluated to check the final acceptance criteria of NUREG-1061, Volume 3.

4.3 Evaluation of the D.C. Cook, Units 1 and 2 Pressurizer Surge Line Piping

Based on the loadings supplied by the licensee in their September 5, 2000, follow up letter, the NRC staff confirmed that node 1020 for D.C. Cook, Unit 2 was the limiting location for this pressurizer surge line evaluation. It should be noted that the NRC staff's analysis did not include a consideration of the loads from the event in which an SSE is postulated to occur during startup or shutdown of the unit. For the purpose for the LBB evaluation, the NRC staff has consistently considered this to be a sufficiently low probability event that using this load combination to establish the critical flaw size would be overly conservative given the other margins explicitly included in the analysis. Since the weld at node 1020 was identified as the

critical material, the LBB.ENG3 code was used for the evaluation, permitting the stress-strain properties of both the weld and the base metal to be included in the detailed analysis. The NRC staff's analysis also considered the nominal pipe wall thickness (1.406 inches) and minimum pipe wall thickness (1.251 inches) based on the weld counterbore. The NRC staff's results, below, reflect the assumption of a 1.406 inch wall thickness which was determined to be a conservative assumption for the overall LBB analysis.

Using base material properties as submitted by the licensee for appropriate temperature (617 °F when analyzing the NOP plus SSE condition, 205 °F when evaluating the forced cooldown event), the aged SS weld properties cited in Table 2, the J-R curve based on the information from Wilkowski and Ghaliadi, and the loads from Table 1b, the NRC staff calculated the limiting critical flaw size for D.C. Cook, Unit 2, node 1020. The critical flaw size at node 1020 for the NOP plus SSE condition (including NOP thermal stratification stresses) was determined to be 13.3 inches. In comparison, the critical flaw size at node 1020 for the forced cooldown condition was determined to be 14.1 inches. This result may, from the loads shown in Table 1b, appear to be contradictory. If the overall stresses from the forced cooldown condition loads are calculated versus the overall stresses for the NOP + SSE condition, the forced cooldown stresses will be shown to be higher. However, the increase in material strength at 205 °F (the temperature at node 1020 when the largest stresses occur in the startup/shutdown transient) increases the pipe's load carrying capacity during this transient relative to the capacity at NOP conditions. This increase in load carrying capacity makes the flaw size which can be withstood without failure longer and is sufficient to mitigate the effects of the increased loads and cause the NOP plus SSE evaluation to be limiting.

The NRC staff then used the PICEP code to evaluate the leakage flaw size for node 1020 under NOP plus thermal stratification conditions. Using the surface roughness value that the NRC staff has used in previous LBB evaluations of $\varepsilon = 0.003$ inch, the NRC staff determined that 8 gpm of leakage would be expected from a 7.47 inch through-wall flaw. Therefore, the minimum factor of safety between the length of critical and leakage size flaws using this approach would be (13.3/7.47) = 1.78. In previous LBB evaluations, the NRC staff has concluded that margins of slightly less than 2 on the critical-to-leakage flaw size are acceptable, provided that a full margin of 10 is maintained on the leakage uncertainty. The NRC staff concluded that for this evaluation, a margin of 1.78 provides adequate assurance that the D.C. Cook, Units 1 and 2 pressurizer surge line piping will exhibit LBB behavior. Finally, the 7.47 inch leakage flaw was shown to be stable under a combination of NOP plus SSE loads. Therefore, both LBB criteria were demonstrated for the bounding location and LBB behavior had been demonstrated for the D.C. Cook, Units 1 and 2 pressurizer surge line piping.

4.4 Evaluation of the D.C. Cook, Units 1 and 2 Containment Leakage Detection System

D. C. Cook, Units 1 and 2 Technical Specification (TS) 3/4.4.6 requires that the reactor coolant systems (RCS) leak detection system consisting of one containment atmosphere particulate radioactivity monitoring channel, a containment sump level and flow monitoring system, and either a containment humidity monitor or one containment atmosphere gaseous radioactivity monitoring channel be operable when the units are in Modes 1, 2, 3, and 4. The leakage detection system has been credited by the licensee with the capability to detect a 1-gpm leak in four hours to meet Generic Letter (GL) 84-04 requirements for applying the LBB methodology

to the facilities' main coolant loops. The NRC staff previously reviewed the licensee's leak detection in relationship to GL 84-04, and found it to be acceptable in a letter from NRC to I&M, dated November 22, 1985.

In a letter dated October 7, 2000, the licensee provided additional information regarding the capability of the leakage detection system at D.C. Cook, Units 1 and 2 to support the pressurizer surge line LBB application. The licensee stated that the particulate radiation monitoring system has the highest sensitivity among those leakage detection systems specified in the TS. The particulate radiation monitoring system has redundant channels, one of which is required to be operable in TS 3/4.4.6. In accordance with the TS, the channels are subject to twice daily channel checks, a quarterly channel functional test, and an eighteen-month channel calibration. Notification of an adverse condition, such as a system leak or a monitor failure, is accomplished via the radiation monitoring control terminals and the plant process computer. These devices create the radiation monitoring system alarm.

The licensee evaluated the annual availability of the radiation monitoring system in their October 7, 2000, letter (based on operating logs and condition reports) as follows: 1992 - 99.65%, 1993 - 99.83%, 1994 - 99.82%, 1995 - 99.80%, 1996 - 99.89%, 1997 - 99.85%. Furthermore, the licensee evaluated the sensitivity of the lower containment particulate radiation monitoring channel to detect RCS leakage. The licensee concluded that the particulate radiation monitor is capable of detecting a 0.5 gpm leak within one hour. The licensee considered that the systems' redundancy, the frequent surveillance, the past reliability, and the detection capability evaluation provide reasonable assurance that the system will be available and capable of detecting RCS leakage of 0.5 gpm in one hour.

The NRC staff reviewed the above letter and licensee's evaluation of the 0.5 gpm leakage detection capability and requested additional information from the licensee during a telephone conference held on October 16, 2000. In a letter dated October 26, 2000, the licensee responded to the NRC staff's request for additional information (RAI) and provided additional commitments as discussed below.

The NRC staff requested the licensee address the uncertainties of the leakage detection capability introduced by factors such as the variation of reactor coolant radioactivity, proximity of the particulate radiation leak detection sensor to the pressurizer surge line, and effects of surge line insulation. The licensee performed a new calculation, accounting for all the above factors, and concluded that the containment particulate monitors were capable of detecting a 0.8 gpm leak within one hour throughout the operating cycle. The NRC staff reviewed this new calculation including the conservatism in the assumptions used to address the uncertainties, and found it acceptable. Based on the results of this calculation, the NRC staff concluded that the containment particulate monitors at D. C. Cook, Units 1 and 2 are capable of detecting a RCS leak of 0.8 gpm or less within one hour.

The NRC staff requested the licensee implement compensatory actions should the containment particulate monitors become unavailable. The licensee responded that, in accordance with the provisions of NRC Administrative Letter (AL) 98-10, the licensee would implement administrative controls requiring a containment atmosphere grab sample or a RCS water inventory balance every 12 hours if both containment particulate monitor channels became inoperable. As required by AL 98-10, the licensee will submit a license amendment request to replace the administrative controls in a timely manner. For Unit 1, the administrative controls

will be implemented prior to the unit entering Mode 4. For Unit 2, the administrative controls will be implemented if the LBB methodology is to be credited as the basis for reducing any requirements on the installed surge line pipe whip restraints. The NRC staff finds the above commitments consistent with those made for the R.E. Ginna Nuclear Power Plant concerning LBB of its residual heat removal system piping, which was previously reviewed and found acceptable by the NRC staff. Therefore, the NRC staff concluded that implementation of the administrative controls mentioned above along with the licensee's commitment to submit a follow up license amendment, provided acceptable assurance that appropriate actions would be taken to maintain the leakage detection capability required to support this safety evaluation (SE).

The NRC staff also requested the licensee implement compensatory actions should RCS leakage reach a value of 0.8 gpm, the value above which the conclusions in Section 4.3, above, may no longer be valid if the leakage is a result of a through-wall flaw in the pressurizer surge line. The licensee responded that in accordance with the provisions of NRC AL 98-10, the licensee would implement administrative controls requiring that, if unidentified RCS leakage was determined to be greater than or equal to 0.8 gpm and if it cannot be demonstrated that the pressurizer surge line is not the source of the leakage, the Action specified in Technical Specification 3.4.6.2.a for unidentified RCS leakage will be followed. This Action requires that unidentified RCS leakage be reduced within the stated limits (which the NRC staff understands, based on this SE, to be the 0.8 gpm limit discussed herein) within 4 hours, or the unit be placed in hot standby within the next 6 hours and cold shutdown within the following 30 hours. For Unit 1, the administrative controls will be implemented prior to the unit entering Mode 4 from the current extended outage. For Unit 2, the administrative controls will be implemented if the LBB methodology is to be credited as the basis for reducing any requirements on the installed surge line pipe whip restraints. As required by AL 98-10, the licensee will submit a license amendment request to replace the administrative controls in a timely manner. I&M also described additional actions that would be taken if RCS unidentified leakage is below the 0.8 gpm value, but increasing. The NRC staff concluded that implementation of the administrative controls mentioned above, along with the licensee's commitment to submit a follow up license amendment provided reasonable assurance that appropriate actions would be taken in response to unidentified RCS leakage above the level permitted to support the basis for this SE.

5.0 CONCLUSION

Based on the information and analysis supplied by the licensee, the NRC staff was able to independently assess the LBB status of the analyzed portions of the D.C. Cook, Units 1 and 2 pressurizer surge line piping. The NRC staff has concluded that, because acceptable margins on leakage and crack size have been demonstrated, these sections of piping will exhibit LBB behavior. 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, Units 1 and 2 facility licensing basis, consistent with the provisions of 10 CFR 50, Appendix A, General Design Criteria 4.

TABLE 1a: D.C. Cook, Units 1 and 2 Pressurizer Surge Line Loads Used in the Licensee's Analysis (Not Including Torsional Moments)

Unit	Node	Loading Condition	Loads	
			Axial Force (lb)	Moment (in-lb)
2	1020	NOP (Deadweight + Thermal Expansion + Pressure)	205305	591740
2	1020	NOP + Thermal Stratification	207388	553661
2	1020	NOP + SSE	239024	1743082
2	1020	NOP + Thermal Stratification + SSE	236941	1916268
2	1020	Forced Cooldown	47953	2983880
2	1020	Startup/Shutdown + SSE	51704	4467141

TABLE 1b: D.C. Cook, Units 1 and 2 Pressurizer Surge Line Loads Used in the NRC staff's Analysis (Including Torsional Loads)

Unit	Node	Loading Condition	Loads	
			Axial Force (lb)	Moment (in-lb)
2	1020	NOP (Deadweight + Thermal Expansion + Pressure)	205305	614958
2	1020	NOP + Thermal Stratification	207388	566105
2	1020	NOP + SSE	239024	1784816
2	1020	NOP + Thermal Stratification + SSE	236941	1945222
2	1020	Forced Cooldown	47953	2987835
2	1020	Startup/Shutdown + SSE	51704	4486479

TABLE 2: Parameters Used in Staff Evaluation of D.C. Cook, Units 1 And 2 Aged SS Pipe Welds

Parameter	Value
Young's Modulus (205 °F/617 °F)	27600 ksi / 25215 ksi
Yield Strength (205 °F/617 °F)	60 ksi / 47 ksi
Ultimate Tensile Strength (205 °F/617 °F)	77 ksi / 60 ksi
Sigma-zero (205 °F/617 °F)	60 ksi / 47 ksi
Epsilon-zero (205 °F/617 °F)	0.00217 / 0.00238
Ramberg-Osgood Alpha	9.0
Ramberg-Osgood n	9.8
J_{IC}	73.4 KJ / m^2
C	83.5 KJ / m^2 mm
n	0.643

Note: $J = J_{IC} + C(\Delta a)^n$ and a point-by-point representation was converted to English System units after the calculation was completed in metric units.

6.0 REFERENCES

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[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] EPRI Report NP-4768, “Toughness of Austenitic Stainless Steel Pipe Welds,” October 1986.

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

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