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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-315

1.0 INTRODUCTION

By letter dated October 29, 1990, the Indiana Michigan Power Company (IMPC) proposed changes to the Technical Specifications (TS) for Unit 1 which reflect revisions to the pressure set point and enable temperature for the low-temperature overpressure (LTOP) protection system. In addition, the October 29, 1990, letter provided the response for the D. C. Cook Nuclear Power Plant (DCCNP) for Generic Letter (GL) 88-11. Supplementary information was submitted by letters dated June 18, 1991 and April 13, 1992 in response to staff requests. Reevaluation and revision of the LTOP set point was necessitated by the generation of more restrictive 10 CFR Part 50, Appendix G heat up and cooldown curves based on a reactor vessel Capsule U analysis, conducted in accordance with Revision 2 to Regulatory Guide 1.99. Results of the Capsule U vessel exposures up to 32 effective full power-years (EFPY) were provided to the NRC in a submittal dated June 22, 1990. The revised pressure set point of 435 psig will replace the current value of 400 psig in TS 3.4.9.3. The revised set point is larger than the original value because a reduction in the allowance for instrument error more than offsets the change in the 10 CFR Part 50, Appendix G pressure limit. The revised enable temperature of 152°F will replace the current value of 170°F in the following TS: 3.1.2.3, 4.1.2.3.2, 3.4.1.3, 3.4.9.1, 3.4.9.3, 3.5.3, 4.5.3.2, and Bases 3/4.1.2, 3/4.4.1, 3/4.4.9.

The function of LTOPs is to prevent reactor coolant system (RCS) pressure from exceeding Appendix G limits during low temperature operations such as normal plant heat up, cooldown, and cold shutdown; especially, when the RCS is in a water-solid condition. This function is accomplished by venting through the two power-operated relief valves (PORVs) located near the top of the pressurizer whenever RCS pressure exceeds the LTOPs pressure set point and the temperature in one or more of the cold legs is below the enable temperature. The LTOPs pressure relieving capability supplements that of safety relief valve in the residual heat removal (RHR) system which, as per TS, is in operation and open to the RCS at temperature below 350°F. Additionally, DCCNP operating procedures require that a steam bubble be maintained in the pressurizer during low temperature operations. The presence of a steam bubble precludes water solid operation and, thus, mitigates the effect of a potential pressure transient.

Each of the PORVs is designed with adequate relief capacity to prevent RCS pressure from exceeding Appendix G limits when the following limiting transients occur in a water-solid RCS:

(1) A mass input transient caused by a charging/letdown flow mismatch. In this scenario, a single charging pump is in operation (in accordance with TS when RCS temperature is below the enable temperature) and the letdown path through the RHR system is inadvertently isolated due to spurious closure of the RHR isolation valves. As a result, the RHR safety relief valve becomes isolated, leaving only the LTOP activated pressurizer relief valves available for pressure relief.

(2) A heat input transient involving the startup of a reactor coolant pump (RCP) with temperature asymmetry existing between the primary side water in the steam generator and the coolant in the remainder of the RCS. In this scenario, all RCPs have been removed from service and the RCS is being cooled by the RHR system. Because of the reduced RCS circulation, particularly within the steam generator, the relatively stagnant water there remains at a higher temperature than the coolant in the remainder of the RCS loop. Upon startup of an RCP, the warmer water is "washed" out of the steam generator tubes and replaced by the cooler water which is then heated by the secondary side fluid. The resulting expansion of primary side water causes a pressure increase. The maximum possible temperature asymmetry is considered to be 50°F.

Selection of an LTOPs pressure set point entails the consideration of various system parameters. DCCNP employs a constant pressure set point, independent of temperature, in comparison to some plants which employ a variable set point based on temperature. An acceptable set point pressure range is established such that the maximum and minimum pressures occurring during the limiting transients stay within specified bounds. The upper bound is chosen as the lower of the RCS pressure corresponding to Appendix G limits or the RCS pressure corresponding to PORV piping structural limit (based on a water hammer analysis of PORV operation under water-solid conditions). In the case of DCCNP, the upper bound corresponds to the Appendix G limits. The lower bound corresponds to the minimum RCS pressure required for an RCP start (i.e., the RCP No. 1 seal limits) taken as 325 psig. In the event the set point range cannot accommodate both the upper and the lower bounds, the upper (Appendix G) bound takes precedence. For a given set point, the maximum pressure overshoot and undershoot occurring during a transient is dependent on the PORV opening and closure times, respectively, as well as related factors such as RCS pressure signal transmission delay and PORV volumetric capacity vs opening position.

Revision 2 of Branch Position RSB 5-2 in the Standard Review Plan defines the LTOPs enable temperature as RT(NDT) + 90(F). However, the licensee's proposed enable temperature (as well as the current and previous values) does not conform to this definition. The criterion used by the licensee in determining the proposed value is that the enable temperature be greater than the temperature below which the plant may operate in a water-solid condition. As noted earlier, this value is 150°F for DCCNP. Additionally, to ensure

availability of the RHR safety valve while the RHR system is in service (and, thus, provide an alternative or supplementary means of pressure relief to LTOPs), the autoclosure interlock on the RHR isolation valves is administratively defeated.

2.0 EVALUATION

2.1 LTOP SET POINTS

The licensee utilized the LOFTRAN thermal-hydraulic code to model the limiting mass input and heat input transients described above. For the mass input transients, a series of cases was run at various LTOPs set points between 400 and 700 psig, over a range of mass injection rates and a range of PORV opening times varying from one to ten seconds. For conservatism, a low RCS temperature (85°F) corresponding to a high coolant bulk modulus was selected for the analyses. Valve closure time was fixed at four seconds. From these runs, a set of maximum and minimum pressures was determined for each set point value. Similarly, for the heat input transient, a series of cases was run at various set point values over the same range of PORV opening times and the same closure time. All heat input cases were run at RCS temperatures of both 85°F and 150°F, with an assumed asymmetry of 50°F with the steam generator. Again, a set of maximum and minimum pressure was computed for each set point value.

For both transients, only one of the two PORVs was assumed operable and no credit was taken for pressure relief through the RHR safety valve. A comparison of results from the mass input and heat input runs indicated that the most limiting temperature with respect to Appendix G criteria was 85°F and, at that temperature, the mass input transient represented the limiting case. Based on these analyses, a relationship between maximum allowable LTOPs set point and PORV opening time was developed for 32 EFPY.

The above analyses, as presented in the June 18, 1991 submittal, were originally prepared and previously submitted to support LTOPs set point TS changes for DCCNP Unit 2. However, because the LOFTRAN analyses were performed using the most conservative values of the relevant Unit 1 and Unit 2 plant parameters, the methodology and results are applicable to both units. The Appendix G limits for each unit, of course, are different and have been developed from the respective Capsule U analyses.

The proposed LTOPs set point pressure of 435 psig was selected to achieve consistency with the current Unit 2 set point value and is based on assumed PORV opening and closure times of 6.0 seconds and 4.0 seconds, respectively. Actual stroke times for the Unit 1 PORVs, as measured during past in-service testing, have been less than 5.0 seconds for opening and less than 2.0 seconds for closure. Since the maximum allowable LTOPs set point pressure decreases with increasing opening times, the assumed value of 6.0 seconds is, therefore, conservative. As a set point of 435 psig and a 6.0 second opening time, the maximum pressure overshoot was computed as approximately 506 psig. This value is below the Appendix G pressure list of around 515 psig at an RCS temperature of 85°F. The proposed pressure set point is, therefore, acceptable.

Using the criterion discussed earlier, the proposed enable temperature was selected as 152°F. This particular value was chosen to achieve consistency with the current Unit 2 value. As noted previously, the proposed enable temperature does not conform to the definition provided in Branch Position RSB 5-2 of the Standard Review Plan. However, the intent of the position is met because the availability of an alternative means of pressure relief, the RHR safety valve, is ensured because operating procedures preclude water-solid operation above the enable temperature. The proposed enable temperature is, therefore, acceptable.

2.2 GENERIC LETTER 88-11 RESPONSE

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," the IMPC requested permission to revise the pressure-temperature (P-T) limits for the Donald C. Cook Nuclear Power Plant Unit 1. The request was documented in a letter from the licensee dated October 29, 1990. This proposed P-T limits are valid for 32 effective full power years (EFPY). The proposed P-T limits were developed using Regulatory Guide (RG) 1.99, Rev. 2. Generic Letter 88-11 recommends RG 1.99, Rev. 2 be used in calculating P-T limits, unless the use of different methods can be justified. The P-T limits provide for the operation of the reactor coolants system during heat up, cooldown, criticality, and hydrotest.

To evaluate the P-T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev.2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P-T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U. S. Appendices G and H of 10 CFR Part 50 describe specific requirements for the fracture toughness and reactor vessel material surveillance that must be considered in setting P-T limits. An acceptable method for constructing the P-T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor

vessel materials. This guide defines the ART as the sum of unirradiation reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the DCCNP, Unit 1 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff determined that the material with the highest ART at 32 EFPY for Cook 1 was the intermediate shell plate (B4406-3) with 0.15% copper (Cu), 0.49% nickel (Ni), and an initial RT_{ndt} of 40°F.

The licensee has removed four surveillance capsules from DCCNP, Unit 1. The results from capsules T, Y, X, and U in Unit 1 were published in Southwest Research Institutes Reports 02-4770, SwRI-7244-001/1, SwRI 02-6159, and Westinghouse Report WCAP-12483, respectively. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the DCCNP, Unit 1 limiting beltline material, the intermediate shell plate (B4406-3), the staff calculated the ART to be 161.3°F at 1/4T (T = reactor vessel beltline thickness) and 130.8°F for 3/4T at 32 EFPY. The staff used a neutron fluence of 1.41E19 n/cm^2 at the inside diameter of the vessel, which reduced to 8.46E18 n/cm^2 at 1/4T and 3.05E18 n/cm^2 at 3/4T. The ART was determined by Section 2 of RG 1.99, Rev. 2, because the most limiting beltline material was in the surveillance capsules.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 171°F at 1/4T and 138°F at 3/4T at 32 EFPY for the same limiting plate. The staff judges that a difference of 9.7°F between the licensee's ART at 1/4T of 171°F and the staff's ART of 161.3°F is acceptable because the licenseee has used a more conservative Chemistry Factor (CF-6) which was presented without explanation in the Capsule U report. Substituting these ARTs into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P-T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the pre-service system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 60°F from Section IV.A.2 of Appendix G, the staff has determined that the proposed P-T limits satisfy Section IV.A.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. Unirradiated Charpy USE data are available for the beltline plates and the surveillance weld. This surveillance weld, is representative of beltline welds and, therefore, can be used to predict the end of life (EOL) USE of the beltline welds. The staff evaluated the USE issue and determined that both the beltline plates and the beltline welds meet the 50 ft-lb EOL USE requirements.

3.0 SUMMARY

The staff concludes that the proposed P-T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 32 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The P-T limits also satisfy Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2 to calculate the ART. Hence, the proposed P-T limits may be incorporated in the DCCNP, Unit 1 Technical Specifications.

The license demonstrated, through use of limiting analyses, that the proposed LTOPs set point pressure was conservatively selected such that LTOPs will provide the necessary overpressure protection to prevent the revised Appendix G pressure-temperature limits from being exceeded. In addition, the proposed enable temperature has been demonstrated to meet the intent of Branch Position RSB 5-2 of the Standard Review Plan.

4.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988.

2. NUREG-0800, Standard Review Plan, Section 5.3.2 Pressure-Temperature Limits.

3. November 7, 1977, Letter from J. Tillinghast (IMP) to E. G. Case (USNRC), Subject: Donald C. Cook Nuclear Plant Unit No. 1.

4. December 5, 1988, Letter from M. P. Alexich (IMP) to T. E. Murley (USNRC), Subject: Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations."

5. E. B. Norris, "Reactor Vessel Material Surveillance Program for Donald C. Cook Unit No. 1; Analysis of Capsule T." SwRI Report 02-4770, December 8, 1977.

6. E. B. Norris, "Reactor Vessel Material Surveillance Program for Donald C. Cook Unit No. 1; Analysis of Capsule Y," SwRI-7244-001/1, January 1984.

7. E. B. Norris, "Reactor Vessel Material Surveillance Program for Donald C. Cook Unit No. 1; Analysis of Capsule," SwRI Report 06-6159, June 22, 1981.

8 E. Terek, S. L. Anderson, L. Albertin, and N. K. Ray, "Analysis of Capsule U from American Electric Power Company D. C. Cook Unit 1 Reactor Vessel Radiation Surveillance Program," Westinghouse Report WCAP-12483, January 1990.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes in surveillance requirements. The staff has determined that the amendment involves 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 amendment involves no significant hazards consideration and there has been no public comment on such finding (55 FR 49452). Accordingly, the amendment meets 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 amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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