

Hazards Consideration Finding as defined in 10 CFR 50.92.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Attorney for licensee:* Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201-1006.

*NRC Section Chief:* John A. Nakoski.

*Duke Energy Corporation, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Located in Mecklenburg County, North Carolina*

*Date of amendment request:* December 12, 2002.

*Description of amendment request:* The proposed amendments would revise the Technical Specifications (TS) for TS Table 3.3.2-1 Footnote (c) to correct an editorial error, TS 3.4.3 is revised to update the Reactor Coolant System Pressure-Temperature limits for use up to 34 Effective Full Power Years (EFPY) and TS 3.4.12 is revised to update the Low Temperature Over-Pressure limits for use up to 34 EFPY. Associated changes are also proposed for the TS Bases.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Duke has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes to the reactor coolant system (RCS) pressure and temperature (P-T) limits and low temperature overpressure protection (LTOP) limits are developed utilizing the methodology of American Society of Mechanical Engineers (ASME) Section XI, Appendix G, in conjunction with the methodology of ASME Code Case N-641. Usage of these methodologies provides compliance with the underlying intent of 10 CFR [Part] 50 Appendix G and provides operational limits established to prevent non-ductile failure of the reactor vessel. The Loss of Coolant Accident analysis and other accident analyses in the Updated Final Safety Analysis Report (UFSAR) do not assume

failure of the reactor vessel. The P-T and LTOP limits are not initiators or contributors to accident analyses addressed in the UFSAR. The proposed changes do not alter any assumption previously made in the radiological consequence evaluations nor affect the mitigation of the radiological consequences of an accident previously evaluated. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The changes to RCS P-T limits and LTOP limits are proposed to prevent non-ductile failure of the reactor vessel. The proposed changes do not modify the RCS pressure boundary, nor make any physical changes to the facility. The proposed changes do not introduce any new mode of system operation or failure mechanism. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed changes are developed utilizing the methodology of ASME Section XI, Appendix G, in conjunction with the methodology of ASME Code Case N-461. Usage of these methodologies provides compliance with the underlying intent of 10 CFR [Part] 50 Appendix G and provides operational limits established to prevent non-ductile failure of the reactor vessel. This Code case constitutes relaxation from the current requirements of 10 CFR [Part] 50 Appendix G. The alternate methodology allowed by the Code case is based on industry experience gained since the inception of the 10 CFR [Part] 50 Appendix G requirements and replaces some requirements that have now been determined to be excessively conservative. The more appropriate assumptions and provisions allowed by the Code case maintain a margin of safety that is consistent with the intent of 10 CFR [Part] 50 Appendix G. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Duke concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Attorney for licensee:* Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201-1006.

*NRC Section Chief:* John A. Nakoski.

*Entergy Nuclear Operations, Docket No. 50-247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York*

*Date of amendment request:* December 12, 2002.

*Description of amendment request:* The proposed amendment would revise the Facility Operating License and Technical Specifications (TSs) to increase the licensed core thermal power level to 3114.4 megawatts (MWt), which is a 1.4% increase above the currently authorized power level of 3071.4 MWt. The proposed power uprate involves the improvement in the core power uncertainty allowance originally required for the emergency core cooling system (ECCS) evaluations performed in accordance with Appendix K, "ECCS Evaluation Models," to Part 50 of Title 10 of the Code of Federal Regulations. In addition, changes would be made in TS Sections 1.1, 2.1, 2.3, 3.1, 3.4, 6.9, and the applicable TS Bases would be revised to account for the change in power level.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed 1.4% increase in maximum core thermal power is based on the use of instrumentation that supports a reduction in the measurement uncertainty value assumed in certain safety analyses. The affected analyses now use an uncertainty value of 2% which was required by 10 CFR [Part] 50 Appendix K at the time that the plant was originally licensed. At that time, measurement of feedwater flowrate in the plant secondary side used differential pressure-type flow venturists. The plant secondary side thermal calorimetric is used to determine reactor thermal power. A June 2000 revision to 10 CFR [Part] 50 Appendix K permitted the use of lower uncertainty values in the affected analyses, if the reduced value can be justified. Entergy Nuclear Operations (ENO) has implemented the use of Caldon, Inc. Leading Edge Flowmeter (LEFM) technology to measure feedwater flowrate. The LEFM measures fluid velocity by measuring the transit time of ultrasonic pulses introduced into the fluid stream. The LEFM Check System implemented at Indian Point 2 has a demonstrated measurement accuracy of 0.6%. Based on this measurement accuracy, the licensed thermal power can be increased 1.4% by reducing the assumed uncertainty used in safety analyses