

taken in accordance with the Federal Advisory Committee Act after consultation with the Committee Management Secretariat, General Services Administration.

FOR FURTHER INFORMATION CONTACT:

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NUCLEAR REGULATORY COMMISSION

[Docket Nos. 50-369 and 50-370]

Duke Power Company; McGuire Nuclear Station, Unit Nos. 1 and 2; Environmental Assessment and Finding of No Significant Impact

The U.S. Nuclear Regulatory Commission (NRC) is considering issuance of an exemption from Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix G, for Facility Operating License Nos. NPF-9 and NPF-17, issued to Duke Power Company (the licensee), for operation of the McGuire Nuclear Station, Unit Nos. 1 and 2 (McGuire), nuclear power plant, located in Mecklenburg County, North Carolina. Therefore, as required by 10 CFR 51.21, the NRC is issuing this environmental assessment and finding of no significant impact.

Environmental Assessment

Identification of the Proposed Action:

The proposed action would exempt the licensee from the requirements of 10 CFR part 50, appendix G, which would allow the use of American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code) Code Case N-641 as the basis for revised reactor vessel pressure (RVP) and temperature (P/T) curves, and low temperature overpressure protection system setpoints in the McGuire, Unit Nos. 1 and 2 Technical Specifications.

The regulation at 10 CFR part 50, section 50.60(a), requires, in part, that except where an exemption is granted by the Commission, all light-water nuclear power reactors must meet the fracture toughness requirements for the reactor coolant pressure boundary set forth in appendix G to 10 CFR part 50. Appendix G to 10 CFR part 50 requires that P/T limits be established for reactor pressure vessels (RPVs) during normal operating and hydrostatic or leak-rate

testing conditions. Specifically, 10 CFR part 50, Appendix G, states, "The appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions." Appendix G of 10 CFR part 50 specifies that the requirements for these limits are the ASME Code, Section XI, Appendix G, limits.

ASME Code Case N-641 permits the use of alternate reference fracture toughness for reactor vessel materials in determining the P/T curves and low temperature overpressure protection system setpoints for effective temperature and allowable pressure. The alternate reference fracture toughness involves the use of the " K_{IC} fracture toughness curve" instead of the " K_{IA} fracture toughness curve," where K_{IC} and K_{IA} are "Reference Stress Intensity Factors," as defined in ASME Code, Section XI, Appendices A and G, respectively. Since the K_{IC} fracture toughness curve shown in ASME Code, Section XI, Appendix A, Figure A-2200-1 (the K_{IC} fracture toughness curve), provides a higher fracture toughness value than the corresponding K_{IA} fracture toughness curve of ASME Code, Section XI, Appendix G, Figure G-2210-1 (the K_{IA} fracture toughness curve), using ASME Code Case N-641 to establish the P/T curves and low temperature overpressure protection system setpoints would be less conservative than the methodology currently endorsed by 10 CFR part 50, Appendix G. The provisions of ASME Code Case N-641 were incorporated into the Appendix G to Section XI of the ASME Code in the 1998 Edition through 2000 Addenda which is the Edition and Addenda of record in the 2003 edition of 10 CFR part 50. However, in this case, the McGuire licensing basis has only been updated to include the 1995 Edition through 1996 Addenda of the ASME Code. Therefore, an exemption to apply ASME Code Case N-641 is required.

The proposed action is in accordance with the licensee's application dated December 12, 2002, as supplemented by letters dated March 27 and April 23, 2003.

The Need for the Proposed Action

The proposed exemption is needed to allow the licensee to implement ASME Code Case N-641 in order to revise the method used to determine the P/T curves and because the continued use of the method specified by Appendix G to 10 CFR part 50, to develop low temperature overpressure protection system setpoints unnecessarily restricts the P/T operating window.

The underlying purpose of Appendix G, is to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. This is accomplished through regulations that, in part, specify fracture toughness requirements for ferritic materials of the RCPB. Pursuant to 10 CFR part 50, appendix G, it is required that P/T limits for the reactor coolant system (RCS) be at least as conservative as those obtained by applying the methodology of the ASME Code, Section XI, Appendix G. Current P/T limits produce operational constraints by limiting the P/T range available to the operator to heat up or cool down the plant. The operating window through which the operator heats up and cools down the RCS becomes more restrictive with continued reactor vessel service. Reducing this operating window could potentially have an adverse safety impact by increasing the possibility of inadvertent low temperature overpressure protection system actuation due to pressure surges associated with normal plant evolutions, such as reactor coolant pump start and swapping operating charging pumps with the RCS in a water-solid condition. P/T limits for an increased service period of operation of 34 effective full-power years for McGuire, Unit Nos. 1 and 2, based on ASME Code, Section XI, Appendix G requirements, would significantly restrict the ability to perform plant heatup and cooldown, and would create an unnecessary burden to plant operations, and challenge control of plant evolutions required with the Over Pressure Protection feature enabled. Continued operation of McGuire, Unit Nos. 1 and 2, with P/T curves developed to satisfy ASME Code, Section XI, Appendix G, requirements without the relief provided by ASME Code Case N-641 would unnecessarily restrict the P/T operating window, especially at low temperature conditions. Use of the K_{IC} curve in determining the lower bound fracture toughness of RPV steels is more technically correct than use of the K_{IA} curve, since the rate of loading during a heatup or cooldown is slow and is more representative of a static condition than a dynamic condition. The K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel. The staff has required use of the conservatism of the K_{IA} curve since 1974, when the curve was adopted by the ASME Code. This conservatism was initially necessary due to the limited knowledge of the fracture