

significantly between surveillance intervals (31 days). The seven day allowance provides an appropriate backstop to ensure the particulate level is restored to within limits in a reasonable time period. Since the diesel is still capable of performing its function the margin to safety is not reduced.

(6) An allowance for the person who is satisfying the requirement of the radiation protection staff position and for the person filling the Shift Technical Advisor (STA) position to be vacant for not more than two hours in order to provide for unexpected absences is proposed. This is consistent with the allowance permitted for the control room operator as reflected in existing TSs.

The margin of safety is not dependent on the presence of the STA or the radiation protection technician. Therefore, this change does not involve a significant reduction in a margin of safety.

(7) The STA will be allowed to support the shift crew rather than only the shift supervisor. This provides more flexibility and does not dilute the function of the STA.

The margin of safety is not dependent upon who the STA supports. Therefore, this change does not involve a significant reduction in a margin of safety.

(8) The Occupational Radiation Exposure Report will be submitted by April 30 of each calendar year instead of prior to March 1.

The margin of safety is not dependent on the submittal of information. Therefore, this change does not involve a significant reduction in a margin of safety.

(9) An allowance is proposed that will revise the high radiation areas to include additional previously approved methods for implementation of alternatives to the "control device" or "alarm signal" requirements of 10 CFR [Part] 20. These alternatives provide adequate control of personnel in high radiation areas as evidenced by NRC issuance of NUREG-1432.

The requirements for control of high radiation areas provide for the use of alternates to the "control device" or "alarm signal" requirements of 10 CFR 20.1601. This change provides such alternative methods for controlling access. These methods and additional administrative requirements have been determined to provide adequate controls to prevent unauthorized and inadvertent access to such areas. Therefore, this change does not involve a significant reduction in a margin of safety.

(10) An allowance to require periodic testing of stored fuel for the particulates only is proposed.

The testing of stored diesel generator fuel oil is revised to require the periodic testing of the stored fuel oil only for particulates (replacing the periodic testing per ASTM-D975) once every 31 days. The change reflects industry-standard acceptable DG fuel oil testing programs. Over the storage life of ANO-2 DG fuel oil, the properties tested by ASTM-D975 are not expected to change and performing these tests once on the new fuel oil provides adequate assurance of the proper initial quality of fuel oil. The periodic testing for particulates monitors a parameter that reflects degradation of fuel oil and can be trended to provide increased confidence that the stored DG fuel oil will support DG

operability. Therefore, this change does not involve a significant reduction in a margin of safety.

(11) The removal of the requirement to notify the Vice President, Operations ANO within 24 hours of violating a safety limit.

The margin of safety is not dependent upon notification of the Vice President, Operations ANO upon the violation of a TS safety limit. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(12) The Radioactive Effluent Release Report will be submitted by May 1 of each calendar year instead of prior to March 1.

The margin of safety is not dependent on the submittal of information. Therefore, this change does not involve a significant reduction in a margin of safety.

(13) A change to frequency of the integrated leak tests for each system outside containment that could contain highly radioactive fluids from "at a frequency not to exceed refueling cycle intervals" to "at least once per 18 months."

The current and proposed frequencies of this test are equivalent for all practical purposes. Therefore, this change does not involve a significant reduction in a margin of safety.

(14) A change that allows a 25% extension of the frequency in accordance with SR 4.0.2 for the integrated leak tests of each system outside containment that could contain highly radioactive fluids.

The proposed allowance allows a possible increase in performance interval. However, the test will still be performed at reasonable intervals to ensure the intent of the surveillance is maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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.

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Entergy Nuclear Operations, Inc., Docket No. 50-286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: May 28, 2003, as supplemented on June 24, 2003

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and Section 3.4.12, "Low Temperature Overpressure Protection (LTOP)," to incorporate revised reactor pressure vessel P/T limits and

overpressure protection system limits to allow operation up to 20 effective full-power years. Specifically, the proposed amendment would revise TS Figures 3.4.3-1 to 3.4.3-3 and TS Figures 3.4.12-1 to 3.4.12-4.

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 probability of occurrence of an accident previously evaluated for Indian Point 3 is not altered by the proposed amendment to the technical specifications (TSs). The accidents remain the same as currently analyzed in Final Safety Analysis Report (FSAR) as a result of changes to the P/T and LTOP limits. The new P/T and LTOP limits were based on the NRC [Nuclear Regulatory Commission] approved, for Indian Point 3, Westinghouse/Combustion Engineering methodology along with American Society of Mechanical Engineers (ASME) Code (Boiler and Pressure Vessel Code) alternatives including Code Case N-640. Code Case N-640 has been accepted for use by the NRC but has not been incorporated into Reg. [Regulatory] Guide 1.147, Rev. 12, at this time. An exemption is being submitted separately for the use of Code Case N-640. The proposed changes do not impact the integrity of the reactor coolant system pressure boundary (RCPB) as a result of this change. In addition there is no increase in the potential for the occurrence of a loss of coolant accident. The probability of any design basis accident is not affected by the change, nor are the consequences of any design basis accident affected by the proposed change. The proposed P/T limit curves and LTOP limits are not considered to be an initiator or contributor to any accident currently evaluated in the Indian Point 3 FSAR. These new limits ensure the long term integrity of the RCPB.

Fracture toughness test data are obtained from material specimens contained in capsules that are periodically withdrawn from the reactor vessel. These data permit determination of the conditions under which the vessel can be operated with adequate safety margins against non-ductile fracture throughout its service life. A new reactor vessel specimen was withdrawn at the most recent refueling outage and will be analyzed over the next year to enhance the database used to predict the fracture toughness requirements using projected neutron fluence calculations. For each analyzed transient and steady state condition, the allowable pressure is determined as a function of reactor coolant temperature considering postulated flaws in the reactor vessel beltline, inlet nozzle, outlet nozzle, and closure head.

The predicted radiation induced ΔT_{NDT} (shift in reference temperature nil-ductility