Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 1
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Buchanan, NY 10511-0249

April 15, 2002

Re: Indian Point Unit No. 2
Docket No. 50-247
NL-02-052

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: Response to Request for Additional Information Indian Point 2 License Amendment Request for Power Limits With Inoperable Steam Line Safety Valves (TAC No. MB3918)

Reference: 1. Letter from F. Dacimo, Entergy, to U. S. Nuclear Regulatory Commission, dated January 8, 2002, Subject: License Amendment Request (LAR No. 02-004) for Power Limits With Inoperable Steam Line Safety Valves

By letter dated January 8, 2002 (Ref. 1), Entergy Nuclear Operations, Inc. (ENO) submitted an application for an amendment to the Technical Specifications (TS) for Indian Point Unit No. 2 (IP2). The proposed amendment requested a revision to incorporate the use of a more conservative equation to calculate the Power Range Neutron Flux High Trip setpoint when one or more Main Steam Safety Valves are inoperable. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed this submittal, determined that additional information was required to complete its review, and requested that additional information in a telephone conference on March 6, 2002. As a result of the telephone conference, ENO is submitting the requested additional information in Attachment 1. The revised proposed TS pages are included in Attachment 2, which supercede those provided in Reference 1.

The assessment submitted in Reference 1 that concluded that the proposed TS did not involve a significant hazards consideration is not affected by the additional information submitted herein in support of the application. There are no commitments contained in this letter. Should you or your staff have any questions regarding this submittal, please contact Mr. John F. McCann, Manager, Nuclear Safety and Licensing at (914) 734-5074.

Very truly yours,

Fred Dacimo
Vice President – Operations
Indian Point 2

Attachments
cc: See page 2

A001

cc:

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of)
ENTERGY NUCLEAR OPERATIONS, INC.) Docket No. 50-247
Indian Point Nuclear Generating Unit No. 2)

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Pursuant to Section 50.90 of the Regulations of the Nuclear Regulatory Commission (NRC), Entergy Nuclear Operations, Inc., as holder of Facility Operating License No. DPR-26, hereby submits additional information in support of the January 8, 2002 application for amendment of the Technical Specifications. The specific additional information is provided in Attachment 1. The specific revised proposed Technical Specification pages are set forth in Attachment 2.

The assessment submitted on January 8, 2002 demonstrated that the proposed change does not involve a significant hazards consideration as defined in 10CFR50.92(c). That assessment is unchanged by the additional information.

As required by 10CFR50.91(b)(1), a copy of this submittal has been provided to the appropriate New York State official designated to receive such amendments.

BY: ______________________
Fred Dacimo
Vice President – Operations
Indian Point 2

Subscribed and sworn to
before me this 15 day
April, 2002

____
Notary Public

ERSILIA A. BOVIERO
Notary Public, State of New York
No. 01AM6038689
Qualified in Westchester County
Commission Expires March 20, 2004

ATTACHMENT 1 TO NL-02-052

Response to Request for Additional Information
Regarding Power Limits With Inoperable Steam Line Safety Valves

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

Request for Additional Information:

The instrumentation and channel uncertainties term "U" on page 2 of Attachment 1, "License Amendment Request," does not appear to be defined exactly the same as the calculation methodology proposed in NSAL 94-001, since the two separate sequential operations that are proposed in NSAL 94-001 are combined into one formula. The second operation specified in the NSAL is to adjust the values calculated from the first operation lower to account for instrument and channel uncertainties (typically 9% power). Please clarify the definition of the uncertainties term used in your calculation and how it compares to the calculation methodology proposed by NSAL 94-001.

Response to Request for Additional Information:

The instrumentation and channel uncertainties term, U, used in the calculation of the maximum allowable Power Range Neutron Flux High Trip setpoints when one or more of the Main Steam Safety Valves (MSSVs) are inoperable is the typical value of 9% given in NSAL 94-001 to account for Nuclear Instrumentation System trip channel uncertainties. This is conservative for IP2, since the present calculated uncertainties for these channels are closer to 7.5%. The multiplication factor, as used in the calculation, was actually 0.91 or (1-U) to reduce the calculated setpoint by 9%.

However, when reviewing the calculation methodology of combining the two separate sequential operations proposed in NSAL 94-001, it was determined that the reduction of the calculated Power Range Neutron Flux High Trip setpoints by using a multiplication factor was nonconservative. The calculated value should be reduced by 9% of full scale (using subtraction) not by 9% of setpoint value.

The revised calculation methodology is as proposed by NSAL 94-001. That is, the maximum allowable Power Range Neutron Flux High setpoint (SP) is calculated using the following simple, conservative heat balance equation:

$$SP = [(100/Q) * (w_s h_{fg} N/K)] - U$$

Where:

SP	=	Safety Analysis power range neutron flux setpoint, percent
Q	=	Nominal NSSS power rating of the plant (including reactor coolant pump heat), Mwt
K	=	Conversion factor, 947.82 (Btu/sec)/Mwt
w_s	=	Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, lb_m/sec
h_{fg}	=	Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, Btu/lb_m
N	=	Number of loops in the plant - 4
U	=	Instrument and channel uncertainties, percent of full scale

The ENO calculation FMX-00278-01, "High Neutron Flux Trip Setpoint Change Allowing Operation at Reduced Power Levels with Inoperable Main Steam Safety Valves (MSSVs)," was revised to calculate the values for the reduced setpoints and then reduce the calculated values by 9% of full scale to account for instrument and channel uncertainties.

ATTACHMENT 2 TO NL-02-052

TECHNICAL SPECIFICATION PAGES IN

STRIKEOUT/SHADED FORMAT

Deleted text is shown as ~~strikeout~~.

Added text is shown as shaded.

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

If these requirements cannot be met, then:

1. maintain the plant in a safe, stable mode which minimizes the potential for a reactor trip, and

2. continue efforts to restore water supply to the auxiliary feedwater system, and

3. notify the NRC within 24 hours regarding the planned corrective action.

Basis

Reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condensers. Thereafter, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to feed the steam generators is provided by operation of the turbine cycle feedwater system.

The operability of the twenty main steam line code safety valves ensure that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% Rated Thermal Power coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The total relieving capacity of the twenty main steam safety valves is 15,108,000 lbs/hr which is 114 percent of the total secondary steam flow of 13,310,000 lbs/hr at 100% NSSS Power (3083.4 Mwt). Startup and/or power operation is allowable with main steam safety valves inoperable within the limitations of Table 3.4-1 on the basis of the reduction in secondary system steam flow and thermal power required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are ~~derived on the following basis:~~ based on the heat removal capacity of the remaining operable steam line safety valves. The maximum thermal power corresponding to the heat removal capacity of the remaining operable steam line safety valves is determined via a conservative heat balance calculation as described in the attachment to Ref. 2 with an appropriate allowance for calorimetric power uncertainty.

$$\cancel{SP = \left[\frac{X - YV}{X}\right] \cdot 109}$$

~~Where:~~

~~SP = Reduced reactor trip setpoint in percent of rated thermal power~~

~~V = Maximum number of inoperable safety valves per steam line~~

~~109 = Power Range Neutron Flux High Trip Setpoint for (4) loop operation~~

~~X = Total relieving capacity of all safety valves per steam line (3,777,000 lbs/hr.)~~

~~Y = Maximum relieving capacity of any one safety valve (823,000 lbs/hr.)~~

In the unlikely event of complete loss of electrical power to the station, decay heat removal would continue to be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary steam generator feedwater pumps, and steam discharge to the atmosphere via the main steam safety valves and atmospheric relief valves. One motor-driven auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the plant. The minimum amount of water in the condensate storage tank is the amount needed for 24 hours at hot standby. When the condensate storage supply is exhausted, city water will be used.

The limit on secondary coolant total iodine activity of I-131 and I-133 is based on a postulated release of secondary coolant equivalent to the contents of four steam generators to the atmosphere due to a net load rejection with loss-of-offsite power. The limiting dose for this case would result from radioactive iodine in the secondary coolant. I-131 and I-133 are the dominant isotopes because of their low MPCs in air and because the other, shorter-lived isotopes cannot build up to significant concentrations in the secondary coolant under the limits of primary system leak rate and activity. One tenth of the iodine in the secondary coolant is assumed to reach the site boundary, making allowance for plate-out and retention in water droplets. The inhalation dose at the site boundary is then as follows:

$$\text{Dose(rem)} = \frac{C \bullet V}{10} \bullet B(t) \bullet X/Q \bullet DCF$$

where:

C = secondary coolant activity (0.15 μCi/cc = 0.15 Ci/m^3)

V = water volume in four steam generators (7416 ft^3 = 210 m^3)

$B(t)$ = breathing rate (3.47×10^{-4} m^3/sec)

X/Q = 7.5×10^{-4} sec/m^3

DCF = 1.00×10^{6} rem/Ci Iodine (131 and 133) inhaled

The resultant dose is less than 1.0 rem.

Reference

1. UFSAR - Chapter 10 and Section 14.1.9
2. NRC Information Notice 94-60: Potential Overpressurization of Main Steam System

TABLE 3.4-1

Maximum Allowable Power Range Neutron Flux High
Setpoint with Inoperable Steam Line Safety Valves
During 4-Loop Operation

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of Rated Thermal Power)
1	~~85~~ 64
2	~~61~~ 44
3	~~37~~ 24