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February 22, 1990

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

Mr. Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects I/II
US Nuclear Regulatory Commission
Mail Stop 14B-2
Washington, DC 20555

SUBJECT: Application for License Amendment to Increase Authorized Power Level (TAC No. 69542).

On February 9, 1990 the Advisory Committee on Reactor Safeguards (ACRS) discussed Con Edison's application for a license amendment authorizing an Indian Point Unit No. 2 power level of 3071.4 MWt. The full ACRS was briefed by the Systematic Assessment of Experience Subcommittee which had met on February 6, 1990 to discuss the subject application. The following are responses to specific questions raised at the full committee meeting for which responses were not readily available.

o Question:

Given the current Indian Point Unit No. 2 extended fuel cycle duration and proposed core average temperature and 3071.4 MWt power level, what is the expected change in the moderator temperature coefficient (MTC)?

Response:

The core is designed so that the cycle specific values will always satisfy the appropriate safety limits. In response to the committee's specific question, a comparison of MTC values is provided below:

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MODERATOR TEMPERATURE COEFFICIENTS (PCM/°F)

	CYCLE 10 (2758 MWt)	STRETCH * (3071.4 MWt)
VESSEL AVERAGE MODERATOR TEMP. (°F)	549	579.7*
CORE AVERAGE MOD. TEMP. (HZP) (°F)	547	547
NOMINAL EOL BURNUP (MWD/MTU)	17050	16000
HZP, ARO, NOXE (0 MWD/MTU)	-0.58	-0.58
HFP, ARO, EQXE (0 MWD/MTU)	-4.17	-6.3*
HFP, ARO, EQXE (EOL)	-23.67	-32.28*

* NOTE: The stretch-rating information is based on evaluations and calculations for the cycle 10 stretch rating safety analysis. These data were not calculated for a specific cycle and are not limiting values but do represent typically expected values.

The vessel average moderator temperature (579.7°F) is the upper limit used in the analysis.

o Question:

What is the effect of Tave and power level increases sought by the proposed license amendment on the projected end of life Reference Nil-ductility Transition Temperature (RT_{NDT}) and Reference Pressurized Thermal Shock Temperature (RT_{PTS}), and what would be the remaining margin?

Response:

The limiting material for RT_{NDT} and RT_{PTS} for Indian Point Unit No. 2 Reactor Vessel is a plate at location 0 thickness from the inside surface for RT_{PTS} and at location 1/4 thickness for RT_{NDT} . The effect of uprating from 2758 MWt to 3071.4 MWt in core power level and vessel Tave increase from 549°F to 579.7°F on the projected end of life RT_{NDT} and RT_{PTS} for 32 effective full power years is summarized below. The methodology used to calculate RT_{NDT} is Regulatory Guide 1.99 Revision 2. The methodology of 10 CFR 50.61 is utilized to determine RT_{PTS} .

Power Level	Tavg	RT_{NDT}	$RT_{PTS}/MARGIN$
2758 MWt	549°F	233°F	237°F/33°F
3071.4 MWt	579.7°F	240°F	244°F/26°F

o Question:

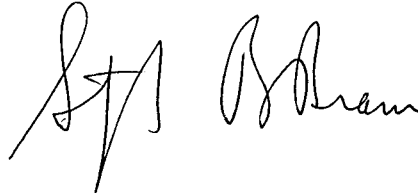
Has the lower girth weld of the steam generators(S/Gs) ever been inspected?

Response:

Per Section XI of the ASME Boiler and Pressure Vessel Code, Article IWC, a portion(136 inches) of the lower girth weld for S/G No. 21 was inspected on November 22, 1987. No indications were found.

Should you have any questions regarding this matter please contact Mr. Charles W. Jackson, Manager, Nuclear Safety and Licensing.

Very truly yours,



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