Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
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November 21, 2005

Re: Indian Point Unit 2
Docket No. 50-247
NL-05-132

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **Response to Request for Additional Information Regarding Relief Request RR 73 for RPV Weld Examination Schedule (TAC No. MC7306)**

References:

1. Entergy letter NL-05-072 to NRC, dated June 8, 2005; "Request for Relief to Extend the Third 10-year Inservice Inspection Interval for the Reactor Vessel Weld Examination".

2. NRC letter to Entergy, dated October 14, 2005 regarding request for additional information. (ML052840030)

Dear Sir:

Entergy Nuclear Operations, Inc. is submitting additional information to support NRC review of an Inservice Inspection Program relief request submitted by Reference 1. The response to NRC questions transmitted by Reference 2 is attached.

There are no new commitments identified in this submittal. If you have any questions or require additional information, please contact Mr. Patric W. Conroy, Licensing Manager at (914) 734-6668.

Very truly yours,

Patric W. Conroy

Patric W. Conroy

Licensing Manager
Indian Point Energy Center

cc: next page

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cc: Mr. John P. Boska, Senior Project Manager, NRC NRR
Mr. Samuel J. Collins, Regional Administrator, NRC Region 1
NRC Resident Inspector's Office, Indian Point Unit 2
Mr. Peter R. Smith, NYSERDA
Mr. Paul Eddy, NYS Department of Public Service

ATTACHMENT 1 TO NL-05-132

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

REGARDING

INDIAN POINT 2 RELIEF REQUEST RR-73

FOR

REACTOR PRESSURE VESSEL WELD EXAMINATION SCHEDULE

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

NRC QUESTION:

You stated that the technical justification for your request was consistent with the guidance provided in a January 27, 2005, letter from the NRC to Westinghouse Electric Company (Summary of Teleconference with the Westinghouse Owners Group Regarding Potential One Cycle Relief of Reactor Pressure Vessel Shell Weld Inspections at Pressurized-Water Reactors Related to WCAP-16168-NP, "Risk Informed Extension of Reactor Vessel In-Service Inspection Intervals"). Item number six of this guidance is repeated below:

> The licensee could then provide a discussion of how, based on its plant operational experience, fleet-wide operational experience, and plant characteristics, the likelihood of an event (in particular, a significant pressurized thermal shock event) over the next operating cycle which could challenge the integrity of the reactor vessel pressure vessel (RPV), if a flaw was present, is very low.

Section 5.5 of your submittal includes general statements indicating that the likelihood of pressurized thermal shock (PTS) events is small and briefly describes an Entergy operating procedure that provides actions to avoid, or limit thermal shock to the reactor pressure vessel.

The NRC staff is re-evaluating the risk from PTS events in a study done to develop a technical basis for revising Title 10 of the *Code of Federal Regulations*, Part 50, Section 61 (10 CFR 50.61). Although the NRC staff has not yet completed its evaluation, the current results indicate that the following three types of accident sequences cause the more severe PTS events and thereby dominate the risk. Please describe the characteristics of your plant (design and operating procedures) that provide assurance that the likelihood of a severe PTS event over the next operating cycle which could challenge the integrity of the RPV, if a flaw was present, is very low.

Sequence 1:
Any transient with reactor trip followed by one stuck-open pressurizer safety relief valve that re-closes after about 1 hour. Severe PTS events also require the failure to properly control high-head injection.

Sequence 2:
Large loss of secondary steam from steam line break or stuck-open atmospheric dump valves. Severe PTS events also require the failure to properly control auxiliary feedwater flow rate and destination (e.g., away from affected steam generators) and failure to properly control high pressure injection.

Sequence 3:
Four to nine-inch loss-of-coolant accidents. Severity of PTS event depends on break location (worst location appears to be in the pressurizer surge line) and primary injection systems flow rate and water temperature.

ENTERGY RESPONSE:

The IP2 operator response to each of the listed sequences would be in accordance with the IP2 Emergency Operating Procedures (EOPs). The procedural guidance for these conditions would limit the severity of a PTS event from the initiators. By taking the actions to limit the cooldown and subsequent repressurization of the RCS the proceduralized operator response will limit any potential challenge to the Reactor Pressure Vessel.

Sequence 1

Any transient with reactor trip followed by one stuck-open pressurizer safety relief valve that re-closes after about 1 hour. Severe PTS events also require the failure to properly control high-head injection.

During this event, the reactor trip and a Safety Injection would occur on Low Pressurizer Pressure. Based on this the Operators would address the condition per the applicable IP2 Emergency Operating procedures. The operators would then enter Emergency Operating Procedure (EOP) E-0 Reactor Trip or Safety Injection. The major actions taken in this procedure are to verify the reactor has tripped, a Steam Generator (SG) heat sink has been established via Auxiliary Feedwater and Safety Injection (SI) flow established. The operators would check for an uncontrolled Reactor Coolant system (RCS) cooldown and isolate any steam flow and limit AFW flow if a cooldown was occurring.

The operators would transition from E-0 to E-1 Loss of Reactor or Secondary Coolant. In this procedure the operator would again verify the SG's as a heat sink, reset the SI signal to regain manual control of the equipment and check to see if the SI can be terminated. As long as the valve remained open at this time the expected transition would be to ES-1.2 Post LOCA Cooldown and Depressurization.

ES-1.2 will perform a RCS controlled cooldown, depressurization, and SI reduction sequence. This would continue until Residual Heat Removal system could be placed in service to cool the plant down into Mode 5.

As soon as the operators transition from E-0 to another procedure, in this case E-1, the STA would start to monitor the EOP Critical Safety Function Status Trees (CSFST's), the 4th highest priority one being RCS Integrity (PTS) behind, Subcriticality, Core Cooling, and Heat Sink. The normal procedural flow path would have the operators attempt to limit the cooldown and control SI flow. If a PTS concern developed the operators would be notified by the STA and then would transition to the applicable PTS mitigation procedure FR-P.1 or 2, Response to Imminent or Anticipated Pressurized Thermal Shock Condition. Directions are given to maintain RCS pressure less than the Cold Overpressure Limit.

After the initial transition from E-0, the EOP's have a "foldout" page associated with each procedure which has parameters to be monitored throughout that procedure set. In this case with E-1 and ES-1.2, one of the foldout page items is to monitor for SI termination criteria. This would be met when the failed valve closes in about the 1 hour time frame. The operators would verify that RCS pressure was greater than the shut off head of the SI system, a SG heat sink has been established, and the RCS is intact. The operators would then transition to ES-1.1 SI Termination. Here SI system pumps would be secured and pressure controlled.

If a PTS condition had occurred during this time (as evidenced by CSFST's) the operators would have direction from FR-P1 to not cooldown or increase pressure until an hour soak time was completed.

Sequence 2

Large loss of secondary steam from steam line break or stuck-open atmospheric dump valves. Severe PTS events also require the failure to properly control auxiliary feedwater flow rate and destination (e.g., away from affected steam generators), and failure to properly control high-pressure injection.

During this event, the operators would cause a manual reactor trip to occur if an automatic setpoint was not reached. This is in accordance with AOP-UC-1 Uncontrolled Cooldown. The Operators would also close the Main Steam Isolation Valves. This is done to isolate any downstream line break. In this case an Atmospheric Dump valve is failed open which would not isolate the steam flow. The operators would then transition to E-0 Reactor Trip or SI.

The operators would verify the reactor has tripped and a heat sink has been established and transition to ES-0.1 Reactor Trip Response. The first step in ES-0.1 checks for an uncontrolled cooldown. Again AFW is limited and steam flow is isolated. The operators can take action to isolate AFW flow to the SG with the failed valve in accordance with the EOP users guide (OAP-12).

CSFST's are monitored upon the transition from E-0. If a PTS condition is entered the operators would be directed to the applicable FRP procedure, FR-P.1 or 2.

If a SI signal occurs, the operators would transition to E-2 Faulted SG Isolation. This procedure would isolate the SG to limit the cooldown affects. Then a transition to E-1 occurs. From here the same procedural flow path as sequence 1 would occur. When the isolated SG dries out, SI termination criteria would soon be met.

Sequence 3

Four- to nine-inch loss-of-coolant accidents. Severity of PTS event depends on break location (worst location appears to be in the pressurizer line) and primary injection systems flowrate and water temperature.

The plant and operator response to this event would be the same as sequence 1 with the exception that SI termination would not be met. The operators would continue in ES-1.2 and place RHR in service.

In addition to the above EOP considerations, other factors that provide assurance of a low likelihood of a severe PTS event occurring during the next operating cycle include the following:

- Entergy operates IP2 as a base-load plant for the full operating cycle which reduces the likelihood of a PTS risk type transient occurring during load-follow power level maneuvers.
- The pressure-temperature (P-T) curves currently approved for IP2 (Amendment 224) are valid for 25 effective full power years (EFPYs) which provides margin to the projected value of 23.7 EFPY at the end of the next operating cycle. Current analyses also confirm that the reactor vessel RT-PTS values and upper shelf energies for plates and welds are within screening criteria at 32 EFPY which corresponds to the end of the current operating license. The conclusions were confirmed as part of the recent stretch power uprate (Amendment 241).
- A low-leakage core design strategy has been in place for many fuel cycles at IP2, which reduces the neutron flux on the inner vessel wall, particularly at the most highly exposed areas adjacent to the core corners.