Entergy

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Palisades Nuclear Plant
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10 CFR 50, Appendix A

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

Response to Request for Additional Information – Request for Authorization to Extend Third 10-Year ISI Interval for Reactor Vessel Weld Examination (TAC No. MD3059)

Dear Sir or Madam:

By letter dated September 15, 2006, NMC (the former licensee for Palisades Nuclear Plant (PNP)) requested Nuclear Regulatory Commission (NRC) approval for the use of an alternative to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, paragraph IWB-2412, Inspection Program B, for the Palisades Nuclear Plant. PNP submitted this relief request because the Westinghouse Owners Group Topical Report, WCAP-16168-NP, "Risk-Informed Extension of Reactor Vessel Inservice Inspection Interval," dated October 2003, is currently being reviewed by the NRC and not yet approved.

By electronic email dated March 8, 2007, the NRC sent a request for additional information (RAI). On April 26, 2007, a teleconference was held with the NRC to discuss the RAI. Enclosure 1 provides the response to the RAI for PNP.

Summary of Commitments

This letter contains no new commitments and no revision to existing commitments.

Christopher J. Schwarz
Site Vice President
Palisades Nuclear Plant

CC Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC

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NRR

ENCLOSURE 1
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
PALISADES NUCLEAR PLANT

NRC Request

As discussed in a letter to Westinghouse dated January 27, 2005, the staff expects that one-time requests to extend the inspection interval of the reactor pressure vessel (RPV) welds by one cycle should include a discussion indicating that the likelihood of a significant pressurized thermal shock (PTS) event over the next operating cycle is very low.

Your submittals dated March 31 and October 11, 2005, described Palisades' response to three of the most significant PTS sequences identified in the ongoing PTS rulemaking work. To support the conclusion that the request for relief for this second one-cycle extension satisfies the risk-informed principal that any proposed increase in risk is small, please provide an estimate of the annual frequency of these more severe PTS sequences and describe the process used to evaluate the frequency of these events which could challenge the integrity of the RPV, if a flaw was present.

ENO Response

Palisades Nuclear Plant (PNP) was one of three pilot plants evaluated in the recent NRC effort to re-evaluate the risk of pressurized thermal shock. These efforts are summarized in NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61): Summary Report." As part of the NRC effort, probabilistic risk assessment (PRA) models were developed for each of the pilot plants using plant specific information. The PNP PRA model is discussed in an NRC letter report, "Palisades Pressurized Thermal Shock (PTS) Probabilistic Risk Assessment (PRA)," dated October 6, 2004 (ADAMS Accession number ML042880473).

The analyses documented in this report were developed by the PNP PRA staff, and therefore, accurately models the PNP. The PRA model included detailed event tree and fault tree analyses that defined both the sequences of events that are likely to produce a PTS challenge to RPV structural integrity, and the frequency with which such events can be expected to occur. Due to the large number of sequences identified, it was necessary to group/bin sequences with like characteristics into representatives that could later be analyzed using thermal-hydraulic codes. This resulted in 65 binned sequences for PNP. Thermal-hydraulic analyses were performed for each of these bins (i.e., representative transients) by Information Systems Laboratories, Inc. (ISL) to develop time histories of temperature, pressure, and reactor vessel wall heat transfer boundary conditions. The PNP staff assisted ISL in developing the appropriate RELAP boundary conditions, as well as providing a detailed design review of the developed model used in creating the transient histories.

These histories were then input into the probabilistic fracture mechanics (PFM) analysis to determine conditional probability of reactor vessel failure for each transient.

From this analysis, it was determined that only a portion of the transients contribute to the total risk of RPV failure, while the remainder have an insignificant or zero contribution. The transients that were identified to be contributors to PTS risk were then used for the PFM analysis in the PTS study and for the pilot plant studies in this report. Therefore, thirty transients were analyzed for PNP. After detailed PFM analyses, only eleven transients were identified to have a contribution to the frequency of reactor vessel failure greater than one percent of the total risk. The results of the PFM analyses are discussed in ORNL/NRC/LTR-04/18, "Electronic Archival of the Results of Pressurized Thermal Shock Analyses for Beaver Valley, Oconee, and Palisades Reactor Pressure Vessels Generated with the 04.1 version of FAVOR." Information from this report for these eleven sequences/transients (identified by "TH Case #" from ORNL/NRC/LTR-04/18) is provided in Table 1. The column at far right identifies to which sequence category, from the October 11, 2005, submittal, the transient applies. The following sequences were previously documented in a request for additional information (RAI) dated August 23, 2005. The August 23, 2005, RAI was sent in regards to the March 31, 2005, submittal mentioned above.

These sequence categories are defined as follows:

<u>Sequence 1</u>

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

<u>Sequence 2</u>

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.

<u>Sequence 3</u>

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.

Table 1 provides the eleven transients that were identified to have a contribution to the frequency of reactor vessel failure greater than one percent of the total risk.

Table 1: PTS Sequence/Transient Frequencies				
Sequence/ Transient TH Case #	System Failure	Operator Action	Sequence/ Transient Frequency (Events/yr)	Sequence Category
19	Reactor trip with 1 stuck-open ADV on SG-A.	None. Operator does not throttle HPI.	2.29E-03	2
40	40.64 cm (16 in) hot leg break. Containment sump recirculation included in the analysis.	None. Operator does not throttle HPI.	3.22E-05	3
48	Two stuck-open pressurizer SRVs that reclose at 6000 sec after initiation. Containment spray is assumed not to actuate.	None. Operator does not throttle HPI.	7.67E-07	1
54	Main steam line break with failure of both MSIVs to close. Break assumed to be inside containment causing containment spray actuation.	Operator does not isolate AFW on affected SG. Operator does not throttle HPI.	4.26E-06	2
55	Turbine/reactor trip with 2 stuck-open ADVs on SG-A combined with controller failure resulting in the flow from two AFW pumps into affected steam generator.	Operator starts second AFW pump.	2.74E-03	2
58	10.16 cm (4 in) cold leg break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	None. Operator does not throttle HPI.	2.66E-04	3
60	5.08 cm (2 in) surge line break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	None. Operator does not throttle HPI.	2.09E-04	3
62	20.32 cm (8 in) cold leg break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	None. Operator does not throttle HPI.	7.07E-06	3
63	14.37 cm (5.656 in) cold leg break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	None. Operator does not throttle HPI.	6.06E-06	3
64	10.16 cm (4 in) surge line break. Summer conditions assumed (HPI and LPI injection temp = 100 F, Accumulator temp = 90 F)	None. Operator does not throttle HPI.	7.07E-06	3
65	One stuck-open pressurizer SRV that recloses at 6000 sec after initiation. Containment spray is assumed not to actuate.	None. Operator does not throttle HPI.	1.24E-04	1

Notes:
TH ### – Thermal hydraulics run number ### from NRC PTS Risk Re-evaluation
IE – Initiating event
ADV – Atmospheric dump valve
SRV – Safety and relief valve
AFW – Auxiliary feedwater
HPI – High-pressure injection
LPI – Low-pressure injection
RCP – Reactor coolant pump
SG – Steam generator

The sequence/transient frequencies presented in Table 1 show that even if a flaw were present in the PNP reactor vessel beltline, the likelihood of having a PTS sequence/transient that could challenge the integrity of the reactor vessel is acceptably small.