Entergy

Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043
Tel 269 764 2000

Anthony J. Vitale
Site Vice President

PNP 2011-058

August 16, 2011

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

SUBJECT: License Amendment Request to Revise Calculated Peak Containment Internal Pressure

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc. (ENO) requests Nuclear Regulatory Commission (NRC) review and approval of a proposed license amendment to revise Renewed Facility Operating License DPR-20 for the Palisades Nuclear Plant (PNP). ENO proposes to revise Appendix A, Technical Specifications (TS), as they apply to TS Section 5.5.14, "Containment Leak Rate Testing Program."

The proposed license amendment revises the calculated peak containment internal pressure for the design basis loss of coolant accident (LOCA) described in TS Section 5.5.14. The calculated peak containment internal pressure, P_a, is increased from 53 psig to 54.2 psig. This increase in P_a is due to an increase in the calculated mass and energy released into containment during the blowdown phase of the design basis LOCA event.

This proposed change has been analyzed in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that this change involves no significant hazards consideration. The bases for this determination are included in Attachment 1, which provides a description of the proposed change, a background discussion, a technical analysis, a regulatory analysis, and an environmental review. Attachment 2 provides the revised TS pages reflecting the proposed changes. Attachment 3 provides the annotated TS pages showing the proposed changes.

Once approved, the amendment will be implemented within 60 days.

In accordance with 10 CFR 50.91(b), ENO is notifying the State of Michigan of this proposed license amendment by transmitting a copy of this letter to the designated state official.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 16, 2011.

Sincerely,

ajv/jse

Attachments:
1. Description and Evaluation of Requested Change
2. Renewed Operating License Page Change Instructions and Revised Technical Specifications Page
3. Mark-up of Technical Specifications Page

cc: Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC
State of Michigan

ATTACHMENT 1

DESCRIPTION AND EVALUATION OF REQUESTED CHANGE

1.0 DESCRIPTION

Entergy Nuclear Operations, Inc. (ENO) requests Nuclear Regulatory Commission (NRC) review and approval of a license amendment request (LAR) to revise Renewed Facility Operating License DPR-20 for the Palisades Nuclear Plant (PNP).

ENO proposes to revise Appendix A, Technical Specifications (TS), to change the calculated peak containment internal pressure for the design basis loss of coolant accident described in TS Section 5.5.14b. The calculated peak containment internal pressure for this event, P_a, would be increased from 53 psig to 54.2 psig. This increase in P_a is due to an increase in the calculated mass and energy released into containment during the blowdown phase in the loss of coolant accident (LOCA) containment response analysis.

2.0 PROPOSED CHANGE

TS Section 5.5.14b. currently states:

> "The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 53 psig. The containment design pressure is 55 psig."

The proposed change would revise TS 5.5.14b., by replacing the P_a value of 53 psig with a value of 54.2 psig.

The revised TS Section 5.5.14b. would read as follows:

> "The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 54.2 psig. The containment design pressure is 55 psig."

3.0 BACKGROUND

Peak containment internal pressure during a LOCA is calculated to determine, in part, whether the design pressure limit for the containment building would be exceeded during a design basis accident. At PNP, peak containment pressure for a LOCA bounds the peak containment pressure for a main steam line break.

A LOCA is initiated by the rupture of the primary coolant system piping. The primary coolant flashes to steam and escapes through the pipe break. As the steam is released to containment, containment atmosphere pressure and temperature quickly increase. The structures in containment will absorb energy and condense steam, counteracting the initial pressure and temperature increase. The containment air coolers and containment spray system, which are activated by the increase in containment pressure, act to reduce containment pressure and temperature by removing energy from the containment atmosphere as the event progresses.

During a LOCA event, the initial blowdown of the primary coolant system adds mass and energy to the containment atmosphere. Mass and energy release data is provided by Westinghouse as an input to the LOCA containment response analysis performed by ENO.

Westinghouse identified a non-conservative LOCA mass and energy release input for the containment response analysis. Westinghouse determined that the mass and energy generated by the thermal hydraulic response computer code during the blowdown phase of the event was not adequately detailed, with respect to time step data, during the early stages of the transient for use in downstream containment response calculations. This resulted in an under-prediction of the mass and energy released to the containment during the blowdown phase of the event, and consequently after inclusion of the additional mass and energy release led to an increase in the calculated peak containment internal pressure at PNP. The peak containment pressure remains below the PNP containment design pressure of 55 psig.

The under-prediction of mass and energy released into containment occurred only for the LOCA event. The mass and energy release assumed in the main steam line break analysis was not affected.

4.0 TECHNICAL ANALYSIS

TS Section 5.5.14, "Containment Leak Rate Testing Program," describes the calculated peak containment internal pressure during the design basis LOCA, P_a. The containment leak rate testing program uses P_a when leak testing containment, containment isolation valves, and containment penetrations, including the containment airlock, in accordance with 10 CFR 50 Appendix J. Upon NRC approval, the increase in P_a would be reflected in PNP Appendix J containment leak rate testing procedures.

The change in P_a does not affect the offsite radiological consequences of a LOCA as previously analyzed in the PNP Updated Final Safety Analysis Report. The LOCA offsite radiological dose consequence analysis is based on the maximum allowable containment leakage rate of 0.1% of containment air weight per day. The analysis assumes that containment leakage during the first 24 hours of the event is 0.1% of containment atmosphere by weight and 0.05% of containment atmosphere by weight afterward. It also assumes that the release of radionuclides to containment is instantaneously mixed with containment air within the containment free air volume. Since the maximum allowable containment leakage rate is not being revised, containment leakage assumed in the LOCA analysis is not impacted. Therefore, the increase in the calculated peak containment internal pressure does not impact the offsite radiological consequences of the LOCA accident analysis.

The change in P_a does not affect the PNP analysis of radiological consequences of a LOCA with respect to radiological dose to the control room operators. Calculated control room operator dose during a LOCA is dependent on the maximum allowable containment atmosphere leakage rate and is unaffected by calculated peak containment internal pressure, as discussed above. Since the maximum allowable containment leakage rate is not being revised, dose to the control room operators is not affected by a change in peak containment pressure.

The change in P_a does not affect environmentally qualified equipment within containment. This equipment is qualified for the containment design pressure of 55 psig. Therefore, an increase in peak containment pressure to 54.2 psig does not affect the environmental qualification of equipment within containment.

Peak containment temperature is only minimally affected by the increase in mass and energy released to containment during the blowdown phase of a LOCA, and environmentally qualified equipment within containment continue to be qualified with respect to temperature.

5.0 REGULATORY SAFETY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met.

General Design Criterion 4, "Environmental and dynamic effects design bases," states that structures, systems and components important to safety shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents including LOCAs.

General Design Criterion 16, "Containment design," states that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

General Design Criterion 19, "Control room," states that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs, and that adequate radiation protection shall be provided.

General Design Criterion 38, "Containment heat removal," states that a system to remove heat from the reactor containment shall be provided that rapidly reduces, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptable low levels.

These General Design Criteria continue to be met by the change in calculated peak containment pressure. The environmental qualification of equipment within containment is not affected by the change in calculated peak containment pressure following a LOCA. The change in calculated peak containment pressure will be reflected in future 10 CFR 50 Appendix J, Type A containment integrated leakrate testing, so containment integrity is not impacted by the change. The change in calculated peak containment pressure does not impact the maximum allowable containment leakage rate and therefore does not impact control room operator dose. The calculated peak containment pressure remains below containment design pressure.

Based on the considerations above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will continue to be conducted in accordance with the site licensing basis, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

In conclusion, ENO has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the TS, and does not affect conformance with any regulatory requirements or criteria.

5.2 No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. (ENO) is proposing a license amendment to the Palisades Nuclear Plant (PNP), Technical Specifications (TS) Section 5.5.14, "Containment Leakage Rate Testing Program." The proposed amendment would increase the calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, from 53 psig to 54.2 psig.

ENO has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change to P_a does not alter the assumed initiators to any analyzed event. The probability of an accident previously evaluated will not be increased by this proposed change.

The change in P_a will not affect radiological dose consequence analyses. PNP radiological dose consequence analyses assume a certain containment atmosphere leak rate based on the maximum allowable containment leakage rate, which is not affected by the change in calculated peak containment internal pressure . The Appendix J containment leak rate testing program will continue to ensure that containment leakage remains within the leakage assumed in the offsite dose consequence analyses. The consequences of an accident previously evaluated will not be increased by this proposed change.

Therefore, operation of the facility in accordance with the proposed change to P_a will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change provides a higher P_a than currently described in the TS. This change is a result of an increase in the mass and energy release input for the loss of coolant accident containment response analysis. The calculated peak containment pressure remains below the containment design pressure of 55 psig. This change does not involve any alteration in the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Therefore, operation of the facility in accordance with the proposed change to TS Section 5.5.14 would not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The calculated peak containment pressure remains below the containment design pressure of 55 psig. Since PNP radiological consequence analyses are based on the maximum allowable containment leakage rate, which is not being revised, the change in the calculated peak containment pressure does not represent a significant change in the margin of safety.

Therefore, operation of the facility in accordance with the proposed change to TS Section 5.5.14 does not involve a significant reduction in the margin of safety.

6.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment would change a requirement with respect to installed facility components located within the restricted area of the plant as defined in 10 CFR 20. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 PRECEDENCE

This request is similar to the license amendment authorized by the NRC on November 15, 1999, for the Palisades Nuclear Plant (TAC No. MA7000, ADAMS Accession Number ML993320086).

ATTACHMENT 2

Renewed Operating License Page Change Instructions

and

Revised Technical Specifications Page

5.0-18

Two pages follow

ATTACHMENT TO LICENSE AMENDMENT NO.

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Remove the following page of Appendix A, Technical Specifications, and replace with the attached revised page. The revised page is identified by amendment number and contains a line in the margin indicating the area of change.

REMOVE	INSERT
Page 5.0-18	Page 5.0-18

5.5 Programs and Manuals

5.5.13 Safety Functions Determination Program (SFDP) (continued)

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.14 Containment Leak Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines of Regulatory Guide 1.163, "Performance-Based Containment Leakage-Test Program," dated September 1995, except that the next Type A test performed after the May 3, 2001, Type A test shall be performed no later than August 3, 2012, as modified by the following exceptions:

1. Leakage rate testing is not necessary after opening the Emergency Escape Air Lock doors for post-test restoration or post-test adjustment of the air lock door seals. However, a seal contact check shall be performed instead.

   Emergency Escape Airlock door opening, solely for the purpose of strongback removal and performance of the seal contact check, does not necessitate additional pressure testing.

2. Leakage rate testing at P_a is not necessary after adjustment of the Personnel Air Lock door seals. However, a between-the-seals test shall be performed at ≥10 psig instead.

3. Leakage rate testing frequency for the Containment 4 inch purge exhaust valves, the 8 inch purge exhaust valves, and the 12 inch air room supply valves may be extended up to 60 months based on component performance.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 54.2 psig. The containment design pressure is 55 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

ATTACHMENT 3

Mark-up of Technical Specifications Page

5.0-18

One page follows

5.5 Programs and Manuals

5.5.13 Safety Functions Determination Program (SFDP) (continued)

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.14 Containment Leak Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines of Regulatory Guide 1.163, "Performance-Based Containment Leakage-Test Program," dated September 1995, except that the next Type A test performed after the May 3, 2001, Type A test shall be performed no later than August 3, 2012, as modified by the following exceptions:

1. Leakage rate testing is not necessary after opening the Emergency Escape Air Lock doors for post-test restoration or post-test adjustment of the air lock door seals. However, a seal contact check shall be performed instead.

   Emergency Escape Airlock door opening, solely for the purpose of strongback removal and performance of the seal contact check, does not necessitate additional pressure testing.

2. Leakage rate testing at P_a is not necessary after adjustment of the Personnel Air Lock door seals. However, a between-the-seals test shall be performed at ≥10 psig instead.

3. Leakage rate testing frequency for the Containment 4 inch purge exhaust valves, the 8 inch purge exhaust valves, and the 12 inch air room supply valves may be extended up to 60 months based on component performance.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is ~~53~~ 54.2 psig. The containment design pressure is 55 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.