Entergy

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PNP-2011-069

October 14, 2011

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

SUBJECT: Report of Changes, Tests and Experiments and Summary of Commitment Changes

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

Dear Sir or Madam:

Entergy Nuclear Operations, Inc. (ENO) is providing the Palisades Nuclear Plant (PNP) Report of Facility Changes, Tests and Experiments for the time period of September 30, 2009, through September 30, 2011. This report is being submitted in accordance with the requirements of 10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2).

Attachment 1 contains a description of each change to the facility, and a summary of the evaluation performed for each change, in accordance with 10 CFR 50.59. There were no changes made to the facility in accordance with 10 CFR 72.48 during this period.

Attachment 2 contains a summary of two regulatory commitment changes requiring NRC notification that were made from September 30, 2009, through September 30, 2011. The summary includes a justification for the change per Nuclear Energy Institute (NEI) Guideline NEI 99-04, "Guidelines for Managing NRC Commitment Changes," and NRC Regulatory Issue Summary 2000-17, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff."

This letter contains no new commitments.

Sincerely,

owg/jse

Attachment(s):
1. Report of Changes, Tests, and Experiments
2. Summary of Commitment Changes

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

# ATTACHMENT 1

# REPORT OF CHANGES, TESTS, AND EXPERIMENTS

4 Pages Follow

# Report of Facility Changes, Tests, and Experiments

**10 CFR 50.59 Evaluation Log Number:** 09-0601 **Evaluation Revision Number:** 0

**Document Number:** Not applicable.

**Title:** Core Operating Limits Report, Revision 18

**Activity Description:**

This Core Operating Limits Report (COLR) revision changes two of five criteria used to demonstrate adequate margin with respect to the "54/6.3"criterion of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

Section 2.4 of the COLR contains a requirement to perform an evaluation of the pin power and burnup of the core design against five criteria. The evaluation ensures that the design margin of safety is maintained with respect to the alternative source term radiological consequence analysis assumptions, as restricted by footnote 11 of Table 3 of RG 1.183. Footnote 11 restricts the Table 3 non-loss of coolant accident gap fractions (fraction of the fission product inventory residing in the gap between the fuel pellet and the fuel rod cladding) to currently approved light water reactor fuel with a peak burnup up to 62 GWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. This restriction is referred to as the "54/6.3" criterion.

The first two criteria are revised; the remaining three criteria are unchanged.

**Summary of 10 CFR 50.59 Evaluation:**

The NRC alternate source term (AST) safety evaluation addresses gap fraction validity through the evaluation of the five COLR criteria, and. The safety evaluation characterizes the five criteria differently. That is, criteria (1) and (2) are mentioned without qualification, criteria (3) and (4) are characterized as the basis for ensuring that the gap fractions for rods that exceed the "54/6.3" criterion remain less than a factor of two times the values given in Table 3, and criterion (5) is characterized as a fuel management restriction which would preserve the original assumptions within the AST dose calculations. The safety evaluation states that based upon the COLR fuel management criteria, the fission-product gap fractions used in the AST dose calculations are acceptable.

The proposed change does not revise criteria (3), (4) or (5). The revision of criteria (1) and (2) does not impact the margin contained in the doubling of the gap fractions for rods that exceed the "54/6.3" criterion, which is ensured via criteria (3) and (4). The revision of criteria (1) and (2) also does not impact the margin in the compensation process, which is ensured via criterion (5).

Therefore, the change does not result in a reduction of previously approved margins embedded in the AST analyses and the five criteria as revised continue to preserve the original assumptions in the dose analyses.

# Report of Facility Changes, Tests, and Experiments

**10 CFR 50.59 Evaluation Log Number:** 09-0602 **Evaluation Revision Number:** 2

**Document Number:** Engineering Change 8350

**Title:** Replace Containment Spray Isolation Valves per GSI-191 Resolution

**Activity Description:**

This Engineering Change (EC) proposed to recover engineering safeguards system pumps' net positive suction head margin during post-recirculation actuation signal (RAS) operation by throttling containment spray flow to the containment spray headers.

The 10 CFR 50.59 Evaluation for this EC was revised to eliminate a discussion of operator actions in the event that the containment spray isolation valve on each of the two containment spray headers fails open. Both containment spray isolation valves failing open would result in insufficient containment spray pump net positive suction head. With only one isolation valve failed open, sufficient net positive suction head would be available.

Assuming that both containment spray isolation valves fail open requires that two single failures be postulated. However, the Palisades licensing basis only requires postulating one single failure of safety related equipment. Since the licensing basis would require, in this instance, postulating only one containment spray valve failing open, the operator actions credited to address two failed open isolation valves address a situation that is outside of the site's licensing basis. Therefore, the discussion of the operator actions that address two failed open containment spray isolation valves were deleted in the Evaluation revision.

The conclusions of the Evaluation, which are summarized below, were not affected by the revision.

**Summary of 10 CFR 50.59 Evaluation:**

The 50.59 Evaluation concluded the new post-recirculation actuation signal (RAS) throttled design function for the valves is not an initiator of any accident and that no new failure modes that could initiate an accident are introduced. Additionally, there was no change in the likelihood of a malfunction since no new failure modes are created and the likelihood of malfunctions that exist for the existing failures will not be increased. The Evaluation also concluded that, at most, the dose impact of reducing spray system flow post-RAS was only 5% of the containment iodine source term and this portion of the source term could not result in a more than minimal increase in the consequences of an accident.

# Report of Facility Changes, Tests, and Experiments

**10 CFR 50.59 Evaluation Log Number:** 09-0680 **Evaluation Revision Number:** 1

**Document Number:** Engineering Change 10816

**Title:** Radiological Calculations for Alternate Source Term Implementation

**Activity Description:**

Engineering Change 10816 issued the radiological design basis calculations necessary to support the alternative source term (AST) radiological analysis methodology. These calculations required review in a 10 CFR 50.59 Evaluation because of changes made to input parameters in calculation. The input parameter changes impact radiological doses, so the calculations had to be re-performed to confirm that dose requirements continue to be met.

The 50.59 Evaluation described above was subsequently revised to address a couple of administrative deficiencies identified during a review of the Evaluation during a site self-assessment. To address the deficiencies, corrections were made to the responses to Evaluation questions 2 and 5.

The corrections made to the responses to Evaluation questions 2 and 5 had no affect on the conclusions of the Evaluation, which are repeated below.

**Summary of 10 CFR 50.59 Evaluation:**

The revised input assumptions do not result in more than a minimal increase in the consequences of an accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). The design basis radiological calculations were revised to incorporate the changed input parameters, and conclude that the dose limits for the exclusion area boundary, the low population zone, and control room access and occupancy continue to be met. The calculated doses are either unchanged from the previous revisions of these calculations or are changed slightly from the previously calculated doses, but remain within 10% of the difference between the current calculated dose and the regulatory requirement.

Therefore, the proposed activity does not result in more than a minimal increase in the consequences of an accident or malfunction of an system, structure, or component (SSC) important to safety previously evaluated in the UFSAR. This activity also does not increase the probability of any UFSAR described accidents or malfunctions, does not create the possibility of an accident of a different type, does not create the possibility of a malfunction of a SSC important to safety with a different result, and does not affect any design basis limits for fission product barriers.

# Report of Facility Changes, Tests, and Experiments

**10 CFR 50.59 Evaluation Log Number:** 10-0137 **Evaluation Revision Number:** 0

**Document Number:** EA-GOTHIC-04-08, Revision 3

**Title:** Containment Response to a LOCA using Gothic 7.2a

**Activity Description:**

This engineering analysis calculates the containment transient response to a large break loss of cooling accident (LBLOCA) for the purpose of ensuring that the resulting temperatures and pressures remain within design values and ensuring that electrical components are not exposed to conditions exceeding those evaluated by the environmental qualification (EEQ) program. The analysis is performed using GOTHIC 7.2a (Generation of Thermal-Hydraulic Information for Containment) software, which is described in section 14.18.1.2 of the Updated Final Safety Analysis Report (UFSAR).

Revision 3 of this analysis changes the manner in which the containment air coolers are modeled in GOTHIC. Previously, the containment air coolers were modeled using NUCK software to create a table correlating heat removal versus containment atmosphere saturation temperature, which was then used to define air cooler performance in the GOTHIC code by looking values up in the table. This revision of the analysis instead models the containment coolers using fan cooler components built into the GOTHIC software.

**Summary of 10 CFR 50.59 Evaluation:**

The change from using the NUCK table look-up fan cooler modeling technique to using the GOTHIC fan cooler models is not considered a departure from an approved method of evaluation. The GOTHIC methodology has been approved for this application via a NRC safety evaluation (SE) issued to Virginia Electric and Power Company (Dominion), and Palisades complied with the applicable terms, conditions, and limitations associated with the use of the method documented in the SE.

The proposed GOTHIC fan cooler model has been adjusted to yield analysis results which are, at all times during the event simulation, either slightly conservative with respect to the NUCK approach or essentially the same as the NUCK approach.

Because the GOTHIC fan cooler modeling approach was approved for use in containment response analysis calculations in the Dominion SE, and the Palisades model complied with applicable terms, conditions, and limitations regarding the use of the methodology as described in the SE, the change from the NUCK modeling approach to the GOTHIC methodology is not considered to be a departure from an approved method of evaluation. Moreover, the change from the NUCK output table to the GOTHIC model approach involves a change to an element of an approved methodology described in the UFSAR that yields results that are conservative or essentially the same.

# ATTACHMENT 2
# SUMMARY OF COMMITMENT CHANGES

| COMMITMENT NUMBER | SOURCE DOCUMENT/DATE | COMMITMENT DESCRIPTION | REVISED COMMITMENT | JUSTIFICATION |
|---|---|---|---|---|
| 2000318 | Letter, "Response to NRC Request for Additional Information on Proposed Administrative Changes (TAC No. 75882)" / April 30, 1990. | Revise General Operating Procedure 3 to require a signoff that the Plant Review Committee (PRC) has reviewed any upward operational condition changes made under Technical Specification 3.0.4 by June 1, 1990. | Cancel Commitment | Since the commitment was made, Technical Specification 3.0.4 has been revised to require a risk assessment before the Technical Specification may be applied. The risk assessment is a requirement of Standard Technical Specifications and is not unique to Palisades. The risk assessment assures that entry into Technical Specification 3.0.4 will not be made without appropriate administrative controls, which is what the NRC had requested in their request for additional information dated July 31, 1989 (TAC No. 72752). |
| 2011316 | Letter, "Response Providing Information Regarding Implementation Details for the Phase 2 and 3 Mitigation Strategies" / February 22, 2007 | Palisades Nuclear Plant will implement the viable site-specific reactor/containment mitigation strategies described in Enclosure 1 of the letter that could be used by emergency response organization or plant personnel in appropriate procedures/guidelines (Table A.6-1). | Revise Commitment | Remove "Provide guidance to supply an alternate 125V DC power source to the power operated relief valves (PORVs)" from the viable site-specific reactor/containment mitigation strategies listed in Table A.6-1 of Enclosure 1, in letter dated February 22, 2007. Replacement of the PORVs made this strategy no longer viable. |