NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

January 9, 2006

NRC 2005-0007
10 CFR 50.73

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

Point Beach Nuclear Plant Units 1 and 2
Docket Nos. 50-266 and 50-301
License Nos. DPR-24 and DPR-27

Licensee Event Report 266/301/2005-006-00
Calculation Errors in Model for ECCS Long Term Cooling

Enclosed is Licensee Event Report (LER) 266/301/2005-006-00 for the Point Beach Nuclear Plant Units 1 and 2. LER 266/301/2005-006-00 describes the discovery of errors in the calculations that were used as the basis for the response to NRC Generic Letter GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment." These errors impact the analytical basis for demonstrating compliance with the acceptance criteria in 10 CFR 50.46 (b)(5), "Long-Term Cooling." This condition is reportable in accordance with 10 CFR 50.73(a)(2)(ii)(B).

Summary of Commitments

NMC will supplement its response to NRC GL 98-04 and GL 2004-02 by April 15, 2006.

Dennis L. Koehl
Site Vice-President, Point Beach Nuclear Plant
Nuclear Management Company, LLC

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
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NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2007

# LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)



| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| POINT BEACH NUCLEAR PLANT UNIT 1 | 05000266 | 1 of 6 |

TITLE (4)

## CALCULATION ERRORS IN MODEL FOR ECCS LONG TERM COOLING

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVOLVED (8) | |
|---|---|---|---|---|---|---|---|---|---|---|
| MO | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REV NO | MO | DAY | YEAR | FACILITY NAME | DOCKET NUMBER |
| | | | | | | | | | PT BEACH UNIT 2 | 05000301 |
| 11 | 08 | 2005 | 2005 | -- 006 -- | 00 | 01 | 09 | 2006 | FACILITY NAME | DOCKET NUMBER |

| OPERATING MODE (9) | 5/1 |
|---|---|
| POWER LEVEL (10) | 0/100 |

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR : (Check all that apply) (11)

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| | 20.2201(b) | | 20.2203(a)(3)(ii) | X | 50.73(a)(2)(ii)(B) | | 50.73(a)(2)(ix)(A) |
| | 20.2201(d) | | 20.2203(a)(4) | | 50.73(a)(2)(iii) | | 50.73(a)(2)(x) |
| | 20.2203(a)(1) | | 50.36(c)(1)(i)(A) | | 50.73(a)(2)(iv)(A) | | 73.71(a)(4) |
| | 20.2203(a)(2)(i) | | 50.36(c)(1)(ii)(A) | | 50.73(a)(2)(v)(A) | | 73.71(a)(5) |
| | 20.2203(a)(2)(ii) | | 50.36(c)(2) | | 50.73(a)(2)(v)(B) | | OTHER Specify in Abstract below or in NRC Form 366A |
| | 20.2203(a)(2)(iii) | | 50.46(a)(3)(ii) | | 50.73(a)(2)(v)(C) | | |
| | 20.2203(a)(2)(iv) | | 50.73(a)(2)(i)(A) | | 50.73(a)(2)(v)(D) | | |
| | 20.2203(a)(2)(v) | | 50.73(a)(2)(i)(B) | | 50.73(a)(2)(vii) | | |
| | 20.2203(a)(2)(vi) | | 50.73(a)(2)(i)(C) | | 50.73(a)(2)(viii)(A) | | |
| | 20.2203(a)(3)(i) | | 50.73(a)(2)(ii)(A) | | 50.73(a)(2)(viii)(B) | | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| T. Kendall | 920-755-7661 |

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

| CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANU-FA CTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | |

| SUPPLEMENTAL REPORT EXPECTED (14) | | | | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
|---|---|---|---|---|---|---|---|
| | YES (If yes, complete EXPECTED SUBMISSION DATE). | X | NO | | | | |

ABSTRACT

This report describes the discovery of errors in the calculations that were used as the basis for the licensee's response to NRC Generic Letter GL 98-04. Two significant errors were involved:

1. A correlation for head loss across a mixed fiber and particulate debris bed on a screen was improperly applied to a debris bed consisting only of coatings chips, and
2. While interpreting the resulting calculated head loss, the total submergence depth of the screens was used rather than the average submergence depth. This resulted in an erroneous conclusion that the available submergence would be sufficient to ensure adequate flow to the residual heat removal (RHR) pumps. Since the screens would be only partially submerged, air intrusion and loss of RHR pumping function would have been the correct conclusion reached.

Further investigation found that the flow path created by a partially blocked strainer had not been considered and that the increase in expected head loss created an additional challenge to RHR pump operability. These errors and deficiencies in modeling and interpretation of results impact the analytical basis for demonstrating compliance with the acceptance criteria in 10 CFR 50.46 (b)(5), "Long-term Cooling." This condition was reported to the NRC via the Emergency Notification System on November 8, 2005 (EN# 42129).

An operability analysis of this condition demonstrated that adequate net positive suction head (NPSH) would be available to the emergency core cooling system (ECCS) pumps to ensure long-term cooling pending final resolution of generic PWR sump screen issues.

***Event Description:***

While investigating an issue related to containment coatings and their potential to clog the containment [NH] sump strainers [SCN], errors were discovered in the calculations that were used as the basis for responding to NRC Generic Letter GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment." The errors were in three distinct areas, and each error was non-conservative.

The first error was in the use of a correlation that was used to calculate head loss across a screen that was fouled with a debris bed made entirely of assumed coatings (paint) chips. The correlation had been taken from NUREG/CR-6224, and had been developed empirically by measuring head losses across a debris bed comprised of a mixture of fibrous material and fine particulates. No further or additional research or testing results were found that would support the use of the NUREG/CR-6224 correlation for a uniform debris bed of larger flakes or chips.

The second error was in the interpretation of the results of the head loss calculation. The calculation results showed that the head loss across the screens due to debris fouling would be less than the total submergence of the screens. However, it was more than the average submergence, and the screens would not be fully submerged. Since the available head to drive flow across the screens is actually the average submergence (only the bottom of the screen experiences the full submergence differential pressure ($\Delta P$), while the top of the screen has a zero $\Delta P$), these results indicated that under the postulated conditions, the screens could not supply adequate flow for the pumps and air intrusion would occur. The results were incorrectly interpreted as being acceptable.

The third error was a failure to recognize that the relatively small (and impervious) debris "pile" that was calculated to be deposited at the base of the screens would have the effect of restricting flow and increasing head losses in excess of those calculated.

The sump screens at Point Beach Nuclear Plant (PBNP) are right circular cylinders (one per train), approximately 13.5 inches in diameter and approximately 5 feet tall. The sump outlets are 10-inch diameter vertical stainless steel pipes whose open ends are flush with the containment floor (no depressed sump). The screens are arranged concentric with these pipe openings.

The outlet isolation valves are designated 1(2)-SI-850A(B), and are a design that is believed to be unique to PBNP. These "poppet" type valves have a flat circular disc that is attached to a push-rod running concentrically up the sump outlet pipes. The pushrods penetrate the piping through valve packing glands on 90 degree elbows located in the containment tendon gallery. When shut, the valve disc seats on a smooth seating surface flush with the pipe outlets. The valves are opened by a hydraulic cylinder pushing the valve pushrods upward from the tendon gallery. The disc outside diameter is approximately 12 inches, while the stroke length is approximately 2.5 inches. The valve discs have an O-ring retained in a circumferential dovetail groove on their undersides. This results in an O-ring face seal when the valves are shut. Containment pressure acts in the direction to seat the valve discs.

If the screen surface below the approximate 2.5-inch stroke height becomes fouled with debris, the required flow must be drawn through the relatively small (approximately 3/4 inch wide by approximately 12 3/8 inch

diameter) annular gap between the valve disc and the screen. This condition had not been previously recognized and the resulting head losses had not been calculated when determining the available NPSH to the RHR pumps.

The aggregate effects of these three errors impacted the basis used to demonstrate compliance with the acceptance criteria in 10 CFR 50.46 (b)(5), "Long-term Cooling."

Two separate operability evaluations and a supporting calculation were subsequently performed to demonstrate adequate net positive suction head (NPSH) would be available to the emergency core cooling system (ECCS) [BP] pumps to ensure long-term cooling, and that air entrainment would not occur.

The containment sump strainers are planned for modification as previously committed to in Nuclear Management Company, LLC (NMC) letter NRC 2005-0109, "Nuclear Management Company Response to Generic Letter 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors, for Point Beach Nuclear Plant," dated September 1, 2005. This modification will result in a larger strainer surface area and a greater clearance in the vicinity of the SI-850 valves. This modification will be supported by design analysis and testing that will demonstrate the strainers comply with the long-term cooling capability requirement of 10 CFR 50.46(b)(5).

10 CFR 50.46 requires that, "Any change or error correction that results in a calculated ECCS performance that does not conform to the criteria set forth in paragraph (b) of this section is a reportable event as described in 50.55(e), 50.72 and 50.73." This condition was reported to the NRC via the Emergency Notification System on November 8, 2005 (EN# 42129).

***Component and System Description:***

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss-of-coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are two phases of ECCS operation: injection and recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST)[TK] and injected into the Reactor Coolant System (RCS)[AB]. The residual heat removal (RHR)[BP] pumps [P] provide RCS injection directly into the upper reactor vessel [RPV] plenum via the core deluge injection lines, while the safety injection (SI)[BQ] pumps provide RCS injection via the cold legs. When sufficient water is removed from the RWST to ensure that enough boron

has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for recirculation.

The ECCS flow paths consist of piping, valves, heat exchangers and pumps necessary to provide water from the RWST into the RCS during the injection phase and from the containment sump into the RCS during the recirculation phase following the accidents described in this LER. The major components of each subsystem are the RHR pumps, heat exchangers, and the SI pumps. Each of the two subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. ECCS Train interconnections could allow utilization of components from the opposite ECCS train to achieve the required ECCS flowpaths; however, cross train operation in the recirculation mode of operation requires local valve manipulations.

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment sump. The RHR pumps then supply the SI pumps.

**Safety Significance:**

Excessive degraded epoxy and unqualified coatings could potentially migrate to the containment sump screens during operations involving containment sump recirculation and clog the screens such that recirculation cooling water flow is unacceptably reduced. To determine whether the existing inventory of degraded or unqualified coatings could pose such a challenge, transport and head loss calculations were prepared by a contractor. These calculations were subsequently used as a reference when periodically assessing as-found degraded and unqualified coatings.

While reviewing the issues of concern, NMC determined that the unqualified coatings in containment would fail to a very small particulate size that would pass through without fouling the sump screens (approximately 10-1000 microns). This determination was based on the recently published testing results (EPRI Technical Report 1011753, "Design Basis Accident Testing of Pressurized Water Reactor Unqualified Original Equipment Manufacturer Coatings"). In addition, the volumetric fraction of the suspended solids attributable to these unqualified coatings is ~0.13%. This is judged to have an insignificant effect on the hydraulic characteristics of the ECCS system.

This had the effect of eliminating the large majority of postulated coatings "chips" or "flakes" from further consideration, leaving only degraded epoxy-based qualified ("acceptable") coatings as potentially fouling the sump screens.

As a conservative interim analytical measure, NMC adopted a "one-for-one" surface area assumption when assessing the degraded (i.e., delaminating) epoxy coatings. For every square foot of delaminating (but otherwise acceptable) coatings in close proximity of the ECCS sumps, it was assumed that one square foot of ECCS sump screen surface would be blocked. When the known inventory of delaminating epoxy coatings was reviewed, it was determined that there are no such coatings within the zone of transport that would result in their potential embedment on the sump screen surface.

In addition, it was found that the fluid velocities within containment (considerably less than 0.2 fps) are too low to facilitate sliding or tumbling transport of postulated chips (specific gravity of 1.6) across the floor toward the sump screens.

On the strength of these findings, it was determined that the existing inventory of unqualified and degraded coatings do not pose a safety significant risk to the existing sump screens.

As a further assurance of safety, a separate assessment was performed assuming that a debris pile of unspecified composition developed and resulted in an impervious obstruction of the lower several inches of the debris screen. That assessment determined that adequate flow would still be obtained through the upper regions of the debris screen without experiencing air ingestion, and that the frictional head losses through the small annulus (and downstream structures) would not result in either cavitation of the flow stream, or excessive reduction in NPSH available to the RHR pumps. To do so however, required increasing the containment pressure above atmospheric, though still well below the pressures anticipated to prevail during the corresponding LOCA transient.

In aggregate, these two approaches (first of demonstrating that coatings chips would not reach the strainers, and secondly, that even if the strainers did become blocked along their lower extremity that flow and pressure to the pumps would be acceptable) indicate that the calculational errors are considered to have a very low safety significance.

NMC concluded that since no sump screen clogging occurred, this condition did not constitute an actual loss of any safety function; therefore, this condition did not constitute a safety system functional failure.

**Cause:**

The unanalyzed condition was caused by deficiencies in the modeling of containment sump screen head losses, unfamiliarity with partially submerged screen flow phenomena, and failure to recognize the compounding effect of screen blockage in close proximity to the outlet valve disks.

**Corrective Action:**

Operability analyses and supporting evaluations were performed to demonstrate that adequate NPSH would be available to the ECCS pumps to ensure long-term cooling, and that air entrainment would not occur.

Actions were promptly taken to address the calculational deficiencies for the interim. The subject calculations are anticipated to be completely superseded by analyses currently in preparation to support resolution of issues identified in GL 2004-02.

Based on information identified during evaluation of this condition, NMC will supplement its response to NRC GL 98-04 and GL 2004-02 by April 15, 2006.

The containment sump strainers will be modified, as previously committed to in NMC letter NRC 2005-0109, "Nuclear Management Company Response to Generic Letter 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors, for Point Beach Nuclear Plant," dated September 1, 2005. This modification will result in a larger strainer surface area and a

greater clearance in the vicinity of valves 1(2)SI-850A(B.) This modification will be supported by design analysis and testing that will demonstrate the strainers comply with the long-term cooling capability requirement of 10 CFR 50.46(b)(5).

**Previous Similar Events**:

A review of recent LERs (past three years) did not identify any events that involved ECCS modeling errors or degraded epoxy or unanalyzed containment coatings.